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Docket Nos. 50-327/328

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Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. White:

SUBJECT: REVISED SAFETY EVALUATION ON THE TENNESSEE VALLEY AUTHORITY (TVA) SEQUOYAH NUCLEAR PERFORMANCE PLAN

By letters dated January 21 and February 17, 1988, the Nuclear Regulatory Commission (NRC) forwarded its preliminary safety evaluation and a first revised version of this evaluation regarding TVA's Sequoyah Nuclear Performance Plan (SNPP).

The enclosure to this letter forwards a second revised version of this evaluation which corrects some editorial errors and adds new or modified sections concerning: (1) Design Control, (2) Civil and Electrical Calculations, (3) Appendix R, (4) Environmental Qualification, (5), Piece Part Qualification, and (6) Moderate Energy Line Breaks. Of the program elements in the SNPP, all restart items are complete. However, the staff has not yet completed documentation of its review of civil calculations. Inspections in the civil area were conducted during February 1988 and final reports covering the inspections are currently being prepared. Following issuance of these reports, additional material will be included in the final issuance of this evaluation.

As previously noted, the staff plans to issue this evaluation, in final form, as NUREG-1232, Volume 2, Part 1, with the evaluations of employee concern element reports as Part 2.

Sincerely,

Original Signed By
Jane A. Axelrad

Stewart D. Ebneter, Director
Office of Special Projects

8804050313 880325
PDR ADOCK 05000327
P PDR

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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NUREG-1232
Vol. 2, Part 1

REVISED PRELIMINARY SAFETY EVALUATION REPORT ON TENNESSEE VALLEY AUTHORITY: SEQUOYAH NUCLEAR PERFORMANCE PLAN

PART 1: PROGRAMMATIC EVALUATION

U.S. Nuclear Regulatory Commission
Office of Special Projects

March 1988

Changes from February 17, 1988 Evaluation

Abstract		text revised
Table of Contents		page numbers, new sections added
Introduction	(pp. 1-1 to 3)	text added, text revised
Section 2.0	(p. 2-1)	references added
Section 2.1.1	(pp. 2-1 to 3)	text revised
Section 2.1.2	(pp. 2-4 to 5)	text added
Section 2.1.3	(p. 2-5)	text added
Section 2.2.2	(pp. 2-9 to 10)	text revised
Section 2.3.2	(p. 2-13)	text added
Section 2.3.3.1	(pp. 2-13 to 16)	section revised
Section 2.3.3.2.1	(pp. 2-17 to 28)	section added
Section 2.3.3.2.2	(pp. 2-28 to 30)	section added
Section 2.3.3.2.3	(pp. 2-30 to 33)	text added
Section 2.3.3.2.4	(pp. 2-33 to 38)	text revised
Section 2.3.3.2.5	(pp. 2-38 to 44)	section added
Section 2.3.3.3	(p. 2-45)	section revised
Section 2.3.4	(pp. 2-45 to 54)	section added
Section 2.4	(p. 2-56)	typographical correction
Section 2.5.1.2	(p. 2-65)	dates clarified
Section 2.5.1.3	(p. 2-66)	dates clarified
Section 2.8.2	(pp. 2-73 to 76)	typographical corrections
Section 2.8.3	(p. 2-78)	text updated and deleted
Section 3.1	(pp. 3-1 to 20)	section revised
Section 3.2.1	(p. 3-20)	text revised
Section 3.2.2	(p. 3-27)	text revised
Section 3.3.2	(p. 3-32)	text updated
Section 3.3.3	(p. 3-33)	text deleted
Section 3.7	(p. 3-62)	typographical correction
Section 3.7	(p. 3-63)	test revised
Section 3.8.1	(p. 3-64)	text revised
Section 3.8.2	(p. 3-66)	test revised
Section 4.1	(pp. 4-1 to 2)	text revised
Section 4.1.3	(p. 4-4)	text revised
Section 4.2.2.7	(p. 4-12)	text revised
Section 4.4.4.4	(p. 4-19)	text updated
Section 4.5	(p. 4-20)	text revised
Section 4.9.1	(pp. 4-32 to 34)	text revised
Section 4.9.2	(pp. 4-34 to 36)	text revised
Section 5.0	(p. 5-2)	text updated
Appendix A		list updated

ABSTRACT

This Safety Evaluation Report (SER) on the information submitted by the Tennessee Valley Authority (TVA) in its Sequoyah Nuclear Performance Plan, through Revision 2, and supporting documents has been prepared by the U.S. Nuclear Regulatory Commission staff. The plan addresses the plant-specific concerns requiring resolution before startup of either of the Sequoyah units. In particular, the SER addresses required actions for Unit 2 restart. In many cases, the programmatic aspects for Unit 1 are identical to those for Unit 2; the staff will conduct inspections of implementation for those programs. Where the Unit 1 program is different, the staff evaluation will be provided in a supplement to this SER.

On the basis of its review, the staff concludes that Sequoyah-specific issues have been resolved to the extent that would support restart of Sequoyah Unit 2.

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PART 1 PROGRAMMATIC EVALUATION

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APPENDICES

ACRONYMS

AA	alternate analysis
ABGS	auxiliary building general supply
ABGT	auxiliary building gas treatment
	alternating current
AEC	Atomic Energy Commission
AFW	auxiliary feedwater
AHU	air handling unit
AISC	American Institute of Steel Construction
ANI	American Nuclear Insurers
ANSI	American National Standards Institute
AOI	Abnormal Operating Instruction
APS	auxiliary power supply/system
ARF	air return fan
ASME	American Society of Mechanical Engineers
ASTM	American Society of Test Methods
AWS	American Welding Society
BNL	Brookhaven National Laboratory
BOP	balance of plant
BTP	branch technical position
CAQ	condition adverse to quality
CAQR	condition adverse to quality reports
CAR	Sequoyah corrective action report
CATD	corrective action tracking document
CCP	centrifugal charging pump
CCRIS	calculation cross-reference information system (computer data base)
CCW	component cooling water
CECC	central emergency control center (TVA)
CFR	Code of Federal Regulations
CNPP	Corporate Nuclear Performance Plan
CPS	control power system
CPS	cycles per second
CSP	containment spray pump
CSSC	critical safety system components
CSST	common station service transformer
CVCS	chemical and volume control system
DBA	design-basis accident
DBE	design-basis event
DBVP	design baseline and verification program
dc	direct current
DCN	design change notice
DG	diesel generator
DNE	Division of Nuclear Engineering (TVA)
DNQA	Division of Nuclear Quality Assurance
DOR	Division of Operating Reactors
DR	discrepancy report
EA	Engineering Assurance
ECCS	emergency core cooling system
ECN	engineering change notice

ECP	employee concern program
ECSP	employee concern special program
ECTG	employee concern task group
EGTS	effluent gas treatment system
EHC	electrical hydraulic control
EOI	emergency operating instruction
EP	emergency preparedness
EQ	equipment qualification
EQE	Earthquake Engineering Inc.
ERCW	essential raw cooling water
ESF	engineering safety features
FAR	function analysis reports
FHA	fuel handling accident
FLR	full load rejection
FRC	Franklin Research Center
FRM	function review matrix
FSAR	Final Safety Analysis Report
G/C	Gilbert/Commonwealth
GDC	general design criteria
GOI	general operating instruction
HCTTG	Heat Code Traceability Task Group
HELB	high-energy line break
HIV	hydraulic initiated valves
hp	horsepower
HVAC	heating, ventilation and air conditioning
I&C	instrumentation and control
I&CS	instrumentation and control system
IDI	integrated design inspection
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronic Engineers
IMI	instrument maintenance instruction
INPO	Institute of Nuclear Power Operations
IPCEA	Insulated Power Cable Engineers Association
IR	Inspection Report
IRG	independent review group (TVA)
ISA	Instrument Society of America
JD	job description
JTG	joint test group (TVA)
kV	kilovolt
KVA	kilovolt-amperes
kW	kilowatt
LER	licensee event report
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
M-K	Morrison-Knudsen
M&TE	measuring and test equipment
MCC	motor control center
MEB	Mechanical Engineering Branch (TVA)
MELB	moderate-energy line break
MI	maintenance instruction
MOV	motor-operated valve
MOVATS	motor-operated valve automated test system
MS	military standard
MSIV	main steam isolation valve

MSLB	main steam line break
MSVV	main steam valve vaults
NCR	nonconformance report
NDE	non-destructive examination
NEB	Nuclear Engineering Branch (TVA)
NERP	nuclear experience review program (TVA)
NMRG	Nuclear Manager Review Group
NO	Nuclear Operations (TVA)
NQAM	Nuclear Quality Assurance Manual
NRC	Nuclear Regulatory Commission
NSRB	Nuclear Safety Review Board
NSRS	Nuclear Safety Review Staff
OBE	operating-basis earthquake
OC	Office of Construction (TVA)
OL	operating license
ONP	Office of Nuclear Power
P&ID	piping and instrument drawing
PAM	post-accident monitoring
PAR	protective action recommendations
PD	position description
PHMS	permanent hydrogen mitigation system
PIR	problem identification report
PMT	post-modification testing
PNL	Pacific Northwest Laboratory
PORC	Plant Operation Review Committee
PORV	power-operated relief valves
POTC	plant operations training center
PRO	potentially reportable occurrence
QA	quality assurance
QC	quality control
QTC	Quality Technology Company
RC	radiological controls
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDA	radiological dose assessment
REP	radiological emergency plan
RG	regulatory guide
RHR	residual heat removal
RIP	Replacement Items Project
RO	reactor operator
RTG	restart test group (TVA)
RTP	restart test program (TVA)
S&L	Sargent & Lundy
SAL	Sequoyah Activities List
SAM	seismic anchor movement
SALP	systematic assessment of licensee performance
SCR	significant condition report
SCV	steel containment vessel
SDL	shutdown logic
SER	safety evaluation report
SGTR	steam generator tube rupture
SI	surveillance instruction
SIA	Structural Integrity Associates
SIAS	safety injection actuation signal

SNPP	Sequoyah Nuclear Performance Plan
SPDS	safety parameter display system
SPS	safety procedures staff
SQN	Sequoyah Nuclear Plant
SRO	senior reactor operator
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSD	self-drilling
SSE	safe-shutdown earthquake
STA	shift technical advisor
SWEC	Stone & Webster Engineering Corporation
SYSTER	system evaluation report
TACF	temporary alteration control forms
TAM	thermal anchor movement
TAR	test analysis report
TER	technical evaluation report
TSC	technical support center
TSS	Technical Support Section (TVA)
TVA	Tennessee Valley Authority
UHI	upper-head injection
USQD	unreviewed safety question determination
USST	unit station service transformer
UT	ultrasonic testing
VCPS	vital instrumentation and control power system
WB	wedge bolt
WGDTR	waste gas decay tank rupture
ZPA	zero period acceleration

1 INTRODUCTION

On September 17, 1985, the Nuclear Regulatory Commission (NRC) Executive Director for Operations issued a letter to the Chairman of the Board of Directors of the Tennessee Valley Authority (TVA) pursuant to Title 10 of the Code of Federal Regulations Part 50.54(f) (10 CFR 50.54(f)). This letter requested information on the actions TVA was taking to resolve NRC's concerns about TVA's nuclear program. These concerns were divided into four categories: (1) corporate activities, (2) the Sequoyah Nuclear Plant (SQN), (3) the Browns Ferry Nuclear Plant and (4) the Watts Bar Nuclear Plant.

TVA's Corporate Nuclear Performance Plan (CNPP), which was prepared in response to the NRC letter, was originally submitted to the NRC on November 1, 1985. The revised plan was submitted on March 10, 1986, subsequent revisions were submitted to the NRC on July 17, July 31 and December 4, 1986, March 26 and December 10, 1987. The NRC staff safety evaluation on the revised CNPP, through Revision 4, was issued as NUREG-1232, Volume 1, in July 1987.

In addition to its corporate plan, TVA is preparing separate plans to address site-specific problems at each of its nuclear plants. This NRC Safety Evaluation Report (SER) documents the staff's review of the corrective actions implemented by TVA to resolve problems at SQN, particularly for Unit 2 restart. In many cases, long-term corrective actions, extending beyond startup, are required to fully resolve these issues. The Sequoyah Nuclear Performance Plan (SNPP) was submitted on November 1, 1985. Revisions 1 and 2 to the plan were provided to the NRC by TVA on April 1 and July 2, 1987, respectively. Separate staff evaluations will be issued for Sequoyah Unit 1, Browns Ferry and Watts Bar at a later date.

TVA has established functional nuclear divisions and departments at its headquarters to provide technical direction to its nuclear facilities. The plant Site Director at each site plans, schedules, and coordinates the directives of the headquarters staff. Corrective initiatives started at the corporate level are being implemented at Sequoyah through the Sequoyah Site Director as well as through TVA offsite organizations responsible for direct support to Sequoyah. TVA established a Sequoyah Task Force on March 19, 1986, to review implementation of the corrective actions applicable to Sequoyah, to initiate specific actions to address Sequoyah problems, to monitor and ensure that a list of all known work items has been compiled, and to review the process and identification of those items required to be completed before restart of Sequoyah Units 1 and 2, which were shut down by TVA in August 1985. This task force examined the distribution of Sequoyah-related issues that had been identified by the corporate level team of industry advisors, to confirm that the actions taken at Sequoyah suitably address the root causes of problems. Sequoyah site-specific issues deal primarily with operations, maintenance, design control, and management system implementation. The SNPP describes the programs and activities planned by TVA to improve performance in each of these areas.

To complete its assignment, the Sequoyah Task Force developed a list of Sequoyah plant activities (except for those of a routine nature) to be completed before restart (Section IV.3.0 of the SNPP). The Sequoyah Activities List (SAL) was based on issues identified by NRC inspections, TVA quality assurance (QA) audits, American Nuclear Insurers (ANI) audits, Institute of Nuclear Power Operations (INPO) inspection reports, Sequoyah corrective action reports (CAR) and discrepancy reports (DR), TVA Nuclear Safety Review Staff (NSRS) and Nuclear Safety Review Board (NSRB) reports, employee concerns, Sequoyah reactor trip reports and licensee event reports (LERs), and technical issues identified by TVA's Division of Nuclear Engineering (DNE).

The task force had established criteria (Section IV.2.0 of the SNPP) to determine which items were required to be resolved for restart. The staff has reviewed and accepted this criteria by letter dated June 9, 1987. The task force reviewed the process the line organization used to identify, evaluate, disposition, and close out items and reviewed the adequacy of planned actions to be taken before Sequoyah Unit 2 restart. As new issues arise and work activities are developed, they are reviewed by Sequoyah management to determine their importance to restart. The Site Director must approve all new items added to the restart list; however, only the Manager of the Office of Nuclear Power (ONP) can delete items that have been designated for restart.

By letter dated March 11, 1988(a), NRC staff gave its approval for TVA to transfer from the restart criteria to use of the Technical Specifications for Sequoyah Unit 2 only. All issues previously identified as restart issues remained restart items. New issues must be evaluated against Technical Specification operability requirements.

TVA described a number of special programs to ensure integrated corrective actions dealing with problems created by deficiencies in the past conduct of activities. Section III of the original SNPP identified special programs that needed to be resolved before restart of Sequoyah Unit 2. These include programs to:

° complete the documentation and resolve electrical equipment environmental qualification questions initially raised at the time Sequoyah was shut down

° verify the adequacy, with regard to safe plant restart, of past selected safety-related design modifications keeping in mind the weaknesses in past design control programs

° reexamine cable tray support analysis for weaknesses in the analytical basis

° complete system analyses where proper design documentation did not exist in the past

° verify the adequacy of piping and supports that were not rigorously analyzed and where alternate analysis methodology has been poorly applied in the past

- resolve any differences in the effects of increased temperatures during main steam line breaks engendered by revised vendor analysis

- resolve identified areas of noncompliance with 10 CFR 50, Appendix R, fire protection requirements

- assess the adequacy of the welding program at Sequoyah, an issue raised through the employee concern program

- examine issues with regard to instrumentation sense lines

Since the original issuance of the SNPP, TVA has added other special programs to Section III of the plan. These include programs to:

- determine if a problem exists with regard to pipe wall thinning, similar to that which led to a pipe rupture at the Surry Nuclear Plant

- establish a Restart Test Program

- review replacement components and parts and resolve those that do not meet the same quality requirements as the installed equipment

- assess the adequacy of cable ampacity design calculations

- resolve cable pulling concerns such as sidewall pressure, bend radius, jamming, and overpulling

- correct a misapplication of actuator fuses

- resolve an apparent nonconformance with 10 CFR 50, Appendix A, involving containment penetrations

There are other programs as well to consider miscellaneous civil engineering issues, moderate energy line break flooding, containment coatings, ECCS water loss outside the crane wall, platform thermal growth, and heat code traceability. Many of these programs are applicable to Units 1 and 2 although actual implementation for Unit 1 may not be complete until after Unit 2 restart.

The programs mentioned above are evaluated in Sections 2 through 4 of this evaluation. They have been grouped into three sections: adequacy of design, special programs and restart readiness.

Another major problem area included the concerns expressed by TVA employees regarding the quality of TVA's nuclear activities. The programs relating to employee concerns are briefly described in Section 5 of this evaluation. The staff performed individual safety evaluations for the resolution of specific concerns; these will be addressed in Part 2 to this SER.

The NRC plans for handling allegations is discussed in Section 6 of this evaluation.

2 ADEQUACY OF DESIGN

One of the root causes of the problems at Sequoyah was the failure to consistently document any changes to the plant's design basis and to maintain the plant's configuration in accordance with that basis. TVA's efforts to strengthen its design control programs and to assess the effects of past weaknesses on the plant are discussed below.

In addition to TVA's efforts, the staff also conducted an integrated design inspection (IDI) of the Sequoyah essential raw cooling water system. The IDI was to provide added assurance to the NRC that all major design and construction problems had been identified and resolved before Sequoyah Unit 2 restart. The review focused on interfaces throughout design, engineering, construction, and operations. The inspection indicated the need for the licensee to pursue further corrective actions, most notably in the area of civil engineering.

The IDI is further discussed in Inspection Reports (IR) 50-327, 328/87-52 (IDI As-Built Walkdown) 87-48 and 87-74. Further information is also provided in TVA letters of October 29 and December 29, 1987, and March 2, 1988.

2.1 Plant Modification and Design Control

2.1.1 Introduction

In June 1985, TVA's Office of Engineering initiated a major restructuring of its design control program to replace a confusing array of redundant and overlapping procedures with an Engineering Program Directives Manual and a site-specific Project Manual. TVA had an independent contractor, Gilbert/Commonwealth (G/C) assess the adequacy of the new Sequoyah design control program.

NRC concerns regarding the generic implications of the design control process were detailed in the 10 CFR 50.54(f) letter dated September 17, 1985. In this letter, the NRC also requested that TVA provide a detailed description of the design control survey being conducted by G/C for TVA, including a discussion of any generic implications on plant design. In response to this request, TVA submitted a report of the status of the design control program as Part V of the original SNPP. In this document, TVA stated that the design process conformed to then-existing guidance, standards, and regulations.

The G/C survey was completed during October 1985 and submitted to the NRC on June 27, 1986. The survey determined that the then-current design control program was adequate, with three exceptions: (1) the need for reliable information on plant configuration for engineering personnel, (2) the need for increased emphasis on the documentation of design inputs, and (3) the requirement for completed design work to be reviewed for potential unreviewed safety questions.

In its review of the survey, the staff noted that the scope of the G/C review was limited to the Sequoyah design control program implemented after June 1985.

Thus, the survey did not assess the completeness of the previous design control program, nor the adequacy of designs developed under that program. The staff, therefore, asked TVA to describe more completely the basis for its conclusion that Sequoyah design controls were adequate. TVA subsequently contracted G/C to review the engineering change notices (ECNs) that had been implemented from the date of plant licensing to verify that modifications made under the old procedures adhere to original design inputs and conform to applicable codes, standards, and regulatory requirements.

During a meeting on December 12, 1985, the staff raised concerns about the adequacy of the controls on plant configuration with a "two-drawing" (as-designed and as-constructed) design control system. TVA committed to initiate a survey to assess the plant's current configuration to ensure that unreviewed safety questions did not exist. This survey was performed on a representative sample of three plant systems. The survey showed that unreviewed safety questions would result with two modifications if they were not completed or analyzed before restart. Additional weaknesses found in the configuration control program included inaccurate status of engineering change notices, poor control of as-constructed drawings in the control room, and a large backlog of changes that had not been implemented and changes that had been implemented but not administratively closed out.

The staff inspected the second G/C review and the TVA survey (see IR 50-327, 328/86-27) during the final stages of these efforts. TVA submitted the reports of these reviews to the NRC in a letter dated June 27, 1986. The inspections confirmed the inadequacies identified in the reviews and the TVA survey and raised the following additional issues:

- In several cases, standard industry codes and practices were not followed in the samples of original design examined by the NRC staff in conjunction with the review of the G/C effort.

- Some disciplines did not have calculations available to support the original design basis.

- Temporary alteration procedures had been used for permanent design modifications and management controls did not provide for engineering review and closure.

- There was not adequate design evaluation and documentation of seismic requirements in some instances.

- In five cases, design modifications violated the assumptions or the statements contained in unreviewed safety question determinations.

In addition to the above reviews and inspections, TVA's Corporate Division of Nuclear Engineering assessed an evaluation conducted by INPO and an internal evaluation of Sequoyah design control problems. TVA concluded that design control problems did exist and that the primary cause of these problems related to a lack of a comprehensive and integrated program to control design configurations during plant operations. Since licensing of Sequoyah, TVA had gone to an owner/operator concept where operations, rather than a centralized engineering organization, controlled plant modifications--including design work--to the extent of selecting the modifications to be implemented and the engineering organization to use and releasing funds for the engineering design work.

The Sequoyah Nuclear Performance Plan (SNPP), Part II, Section 3, addresses problems with the control of design changes and plant modifications and provides an action plan for improvements in the design control program. According to TVA, the weaknesses in this area, including the failure (1) to thoroughly document engineering work for design changes and (2) to maintain consistency between "as-designed" and "as-constructed" information, were attributed to the following:

- organizational problems (addressed separately in the revised Corporate Nuclear Performance Plan and Section II.1.2.5 of the SNPP)
- lack of adequate design controls and coordination of plant modifications that were done on a drawing-by-drawing basis
- the inability of Sequoyah personnel to follow through in a timely manner with the paperwork associated with changes
- a two-drawing system, where the as-constructed drawings were maintained at the plant and as-designed drawings were maintained by the Division of Nuclear Engineering at TVA headquarters
- the failure to maintain current design criteria and design basis information
- the large scope of some modifications and the associated work plans needed to implement the changes

To correct these weaknesses in the design control area, TVA proposed the following actions:

- revise the design control process to provide improved control of future design changes and plant modifications
- improve plant drawings to properly reflect past changes in a legible manner
- establish the design baseline and verification program (DBVP) to assess the adequacy of past modification work and to correct deficiencies
- review essential design calculations to provide definitive design basis

The DBVP and calculations review programs are discussed in more detail in Sections 2.2 and 2.3, respectively. The remaining aspects are discussed below.

2.1.2 Evaluation

TVA has acknowledged problems with control of plant design changes and is implementing an improved design change control program at Sequoyah. Design control problems identified through employee concerns, external reviews such as those performed by G/C and the Institute of Nuclear Power Operations (INPO), and NRC inspections are being individually addressed and corrected. TVA's action plan represents a significant enhancement to the design control process. Adequate controls appear to be in place for any modifications performed during the transition phase as discussed in IR 50-327, 328/87-42.

TVA's improved design change control program will be implemented in two phases for current and future plant modifications.

The first phase is to be implemented before restart of SQN Unit 2 and includes a change control board and a transitional design control system. The change control board consists of senior Sequoyah personnel who will provide overall management control during the transition period. The board will (1) evaluate existing and proposed modifications to minimize changes, (2) review plant modifications to ensure that line managers are accomplishing the changes in accordance with adequate design and configuration controls, (3) ensure that necessary interface and control procedures exist to maintain design integrity, and (4) ensure that the status of design and plant implementing documents associated with modifications is kept current. The transitional design control system will be based on modified TVA design control procedures. This process will require that design changes that are to be implemented be contained in complete packages specific to the appropriate unit. This will facilitate the reviews required to ensure that each change has been quality engineered, that it can be installed and tested, and that documentation and safety analyses are complete and based on actual plant configuration. A task engineer will coordinate these efforts.

In SNPP Section II.3.3.2, TVA indicates that one of the major keys in maintaining design control is a single, stand-alone plant modification package. This modification package will include a unique modification number, a description of the change and the reason for it, an unreviewed safety question determination (USQD), and installation and testing requirements.

TVA noted in Appendix 2 to the SNPP that many configuration markings on as-constructed drawings in the main control room were ambiguous, illegible, and incorrect. TVA established a program to: (1) check all configuration markings for accuracy, (2) correct legibility problems, and (3) develop an improved drawing system. This effort complemented the first phase of the new design control programs. However, during its inspection in April 1987, the staff identified two items of concern in the area of drawing control: the adequacy of primary and critical drawing lists and the adequacy of the temporary change process. The first item was resolved in IR 50-327, 328/87-65; the latter is a violation (87-65-03). TVA responded to the violation by letter dated February 16, 1988, and committed to second-party verification of changes to control room drawings. The staff has evaluated TVA's response and has found it acceptable (see IR 50-327, 328/88-19).

The second phase in the development of the improved design control program will be to establish a permanent design control system based on the plant modification package concept. A procedure will be developed to ensure a comprehensive and focused evaluation of modifications and proper implementation and follow through. Enhanced aspects of this program include the use of the actual plant configuration for design, updated design criteria, accurate reflection of the modification in licensing documents, and an integrated, project-oriented approach to handle changes to the plant, as opposed to the fragmented work-plan approach used in the past.

The permanent design control system will provide additional enhancement to the design control process. However, the staff recognizes that timeliness of the implementation of the permanent design change program is of concern to plant safety. TVA submitted additional information regarding its schedule for implementation of the permanent design control system in letters dated December 11, 1986, and February 27, 1987(a). In the December 11, 1986 letter, TVA committed to consolidation of the "as-constructed" and "as-designed" information on DBVP primary drawings before the end of the second refueling outage after restart of Unit 2. The staff finds this commitment acceptable because (1) the first refueling is presently planned for several months after restart and (2) in the interim, the actual configuration will be depicted on marked-up drawings available for engineering and operational purposes. By letter dated December 15, 1987, TVA stated that Division of Nuclear Engineering procedures, which were needed to establish the process for preparing Sequoyah implementing procedures, have been implemented. Site level procedures and training will be completed by March 31, 1988. The staff finds this schedule for transition acceptable.

TVA has not committed to implement a single drawing system for drawings other than DBVP drawings which are used by operations to operate the plant (primary drawings such as P&IDs). Other drawings will apparently be produced only as needed to support modifications. The staff believes that a more comprehensive approach, which includes scheduling details and identification of all other drawings to be maintained as configured, is needed. In a letter dated April 1, 1987(a), TVA stated that the details regarding comprehensive scheduling of drawings to be maintained as-configured is still being developed. The staff considers this item to be a post-restart issue.

2.1.3 Conclusions

On the basis of the findings as documented in IR 50-327, 328/87-24, 87-65, and 88-19, the staff concludes that TVA has taken the appropriate steps to correct design control problems at Sequoyah for restart.

2.2 Design Baseline and Verification Program

2.2.1 Introduction

TVA's special design baseline and verification program (DBVP) to assess the effect of past weaknesses in design and configuration control and to identify any corrective actions that may be required is addressed in SNPP Section III.2.

TVA forwarded the original documentation for this program as an enclosure to a June 27, 1986 letter to the NRC. In addition to this submittal, TVA presented an overview of the DBVP to the staff at a public meeting in Bethesda, Maryland on July 17, 1986. The description of the program was subsequently revised and supplemented by a TVA letter dated December 31, 1986(a).

The intent of this program is to provide additional confidence that the plant meets its original licensing bases. The program includes (1) verifying and establishing plant configuration; (2) reconstructing the design basis; (3) reviewing and evaluating, against the design basis, those modifications made since the operating license was issued; and (4) performing required tests or modifications developed from this review.

This program has four major areas:

- The development (or updating) of design criteria for both systems and generic plant design required for the pre-restart phase. This will include an evaluation of the inclusion of licensing commitments in design-basis documents.

- System walkdowns and/or test reviews, within the program boundaries, to verify the configuration and proper functional arrangements as depicted on primary control room drawings are correct.

- The evaluation of facility modifications that have been implemented or proposed since the operating license was issued to determine the technical adequacy of the modifications against the (updated) design-basis documents. Additionally, the status of engineering change notices (ECNs) were assessed to ensure that those notices that have been partially implemented, or not implemented at all, do not reduce the system's ability to perform its designated safety-related function or violate a licensing commitment.

- System evaluations, on the basis of results produced from the modification evaluation and walkdowns, to determine whether the systems, as modified, fulfill their functional design requirements (relative to FSAR Chapter 15 accidents and safe shutdown) and licensing commitments.

TVA also plans to extend its assessment of ECNs outside the scope of the program to verify that an unreviewed safety question has not resulted from a failure to implement or complete such changes.

2.2.2 Evaluation

The DBVP is being implemented in two phases. The pre-restart phase is limited to those systems, or portions of systems, required to mitigate accidents addressed in Chapter 15 of the Final Safety Analysis Report (FSAR) or to provide for safe shutdown. (This defined scope does not include all safety-related components and systems.) The post-restart phase continues engineering activities within the pre-restart phase that TVA considered not essential to safe restart but are necessary to correct identified design control problems. This phase will also extend portions of the program to other safety-related systems.

Scope of Pre-restart Phase

The staff evaluated the adequacy of the scope of the pre-restart phase of this program as presented in the June 27, 1986 submittal. Phase I applies to Unit 2 and common portions of the required systems.

During this initial review, it was not clear to the staff as to why:

(1) TVA chose to include only that portion of the ice condenser required for containment isolation

(2) the hydrogen analyzer and the permanent hydrogen mitigation system (PHMS) were not included as part of the hydrogen mitigation system

(3) the auxiliary feedwater suction and recirculation piping from the condensate storage tank were not included

In addition, on the basis of the system descriptions submitted by TVA, the staff could not verify that the main steam isolation valves (MSIVs) were included in the program scope.

In its response dated December 11, 1986, TVA adequately clarified the staff's concern relating to the auxiliary feedwater system in that the essential raw cooling water provides a safety-grade supply of water to the system and minimum flow requirements are provided through a branch line containing a flow restricting orifice. These features were examined under the DBVP. In addition, TVA confirmed that the main steam system from the steam generators through the MSIVs and the main steam check valves were included in the DBVP.

While TVA identified the ice condenser as a system to be addressed in Phase I of the DBVP, only that portion required for containment isolation was included. It was the staff's position that the portion of the ice condenser system in the DBVP Phase I should include all elements and components of the ice condenser that, in concert, enable the system to perform its safety function (e.g., doors, drains, seals, baskets, structural members, isolation barriers). With regard to the hydrogen analyzers and the PHMS, TVA had excluded those items from the pre-restart portion (Phase I) of the DBVP on the grounds that they are not needed to mitigate FSAR Chapter 15 design-basis accidents, which was the selection criterion developed by TVA. Although the staff concurred with the TVA conclusion that the hydrogen analyzers and the PHMS are not needed to mitigate FSAR Chapter 15 design-basis accidents, it was the staff's position that, in view of the ice condenser containment design vulnerability to hydrogen, design features related to hydrogen are sufficiently important to warrant review as part of the DBVP Phase I. Furthermore, since various independent reviews of TVA design programs had concluded that design control at Sequoyah was particularly weak after the operating license was issued, it is prudent to include it in the pre-restart phase because the PHMS was designed and installed after the license was issued. In its response dated February 27, 1987(a), TVA proposed additional technical assessment of thes items (the ice condenser, PHMS, and hydrogen analyzers). With the additi f these items in the restart portion of the DBVP, the staff concluded that t scope of systems being reviewed is sufficient to ensure the design adequac f requisite safety systems.

The staff had noted in its January 20, 1987 evaluation that TVA was conside d safe shutdown to be defined as hot standby for Sequoyah. The staff considered this inconsistent with its earlier position taken in NUREG-0011 and its Supplement 1. These NRC documents discussed compliance with Branch Technical Position (BTP) RSB 5-1 (NUREG-0800) for reaching cold shutdown with safety grade systems. TVA responded in a letter dated February 27, 1987(a) that Sequoyah's RHR system does not meet the requirement for achieving cold shutdown with safety-grade equipment and that this was recognized by the staff in NUREG-0011. Based on further review of NUREG-0011, the staff agrees with TVA's interpretation that Sequoyah's design basis is hot standby. The staff, therefore, considers that the pre-restart scope of the program is acceptable.

During its inspection, the staff identified an open item relating to whether proper function of logic and instrumentation could be verified during walkdowns. In response to this concern, TVA noted that electrical and instrumentation and control attributes were verified through various other methods, including; verification of terminations by a review of post modification test plans, verified work plans, or walkdowns; reviews of cables and junction boxes through the EQ program; a separate fuse verification program, and a sampling walkdown of instrument sensing lines. These activities satisfactorily resolve the staff's concern regarding the scope of the electrical walkdowns.

The staff had also identified a concern regarding the inclusion of only plant modifications made since licensing and not extending the review to include the original plant design. These observations were considered open issues and were sent to TVA for resolution in a staff letter dated September 9, 1986. In a response dated December 11, 1986, TVA presented the basis for the DBVP scope. As stated by TVA, other programs in the SNPP address specific pre-OL program weaknesses. In addition, the NRC conducted an integrated design inspection at Sequoyah as discussed previously. Based on these considerations the staff has concluded that the scope and system selection for Phase I of the DBVP are acceptable.

TVA defined the scope of the post-restart (Phase II) portion of the DBVP in a May 12, 1987 letter. The staff has not completed its review of the Phase II program; however, this review by the staff is not essential to issuing an SER that addresses the acceptability of TVA's programs to support restart of Sequoyah Unit 2. An evaluation of the Phase II program will be issued by the staff at a later date.

TVA Independent Oversight Review

As an integral part of its DBVP, TVA had the Engineering Assurance (EA) group of the Division of Nuclear Engineering perform an independent oversight review. This independent review effort is staffed on a full-time basis throughout Phase I and is comprised of a multidiscipline team of senior experienced technical personnel (EA team). An in-depth description of the independent oversight review process and its results is contained in TVA Report EA-OR-001, "Engineering Assurance Oversight Review Report, SQN Unit 2 DBVP," which was forwarded to the NRC by a letter dated May 15, 1987.

The objectives of this independent review are listed below.

- Confirm and validate that engineering activities are being conducted in accordance with the overall approved program plan, in accordance with the approved procedures established for the DBVP, and by personnel trained for the specific activity being confirmed/validated.
- Confirm the functional and technical adequacy of the system evaluations and the completeness/correctness of the supporting documentation.
- Verify that the corrective actions resulting from the TVA evaluations have been implemented and documented.
- Verify the adequacy and effectiveness of the transitional design change control methodologies and procedures.

A supplemental report by EA team was forwarded to the NRC by letter dated October 23, 1987. The team's overall conclusions are given below.

° The DBVP procedures were complete and adequate and met the objectives of the program and the activities conducted by the DBVP were correct, adequate, and in accordance with program procedures.

° The DBVP project demonstrated the functional and technical adequacy of modifications by providing and/or identifying supporting documentation and justification to establish that modifications comply with the re-established restart design-basis requirements.

° Reconciliation of the corrective actions and restart decisions for punchlist items was adequate. The identified corrective action documents provided appropriate resolutions for the punchlist item concerns; the justifications to support post-restart decisions were adequately documented; and the changes made to corrective actions and/or restart decisions that were different from what was reported in the system evaluation reports were justified and appropriately documented in the system closeout statements.

° The transitional design change control process is being implemented in a satisfactory manner. Organizational interfaces, responsibilities, and review/approval authorities have been satisfactorily addressed procedurally. Although there were occasional violations noted in the implementation of the procedures, the results were technically acceptable and an adequate level of supporting documentation was made available in the process without additional rework. Tighter project management controls will be required to ensure procedure compliance. The EA team will continue to monitor this area as part of the DBV Phase II oversight activities.

The team concluded that there are no apparent programmatic weaknesses remaining to be resolved with the program as a result of their findings and project action to address these findings. The team verified that actions had taken place to correct its findings; team concluded that the pre-restart phase of the DBVP has been fully and effectively implemented.

NRC review and inspection of the EA oversight has revealed an effective and thorough effort. The EA oversight resulted in both programmatic improvements and identification of technical shortcomings in various aspects of the DBVP implementation. TVA has taken action to correct these issues, and the EA team adequately monitored the corrective actions and enhancements. The staff considers that the EA oversight has provided significant additional assurance regarding the overall adequacy of the DBVP.

NRC Inspection Findings

Five NRC inspections have been conducted to assess the adequacy of TVA's DBVP to support restart of Sequoyah.

NRC IR 50-327, 328/86-38 summarizes the NRC's review of TVA's overall DBVP plan and scope, TVA's procedures for DBVP project review and EA oversight, TVA's preparation of system walkdown packages within the DBVP scope, and the

NRC's preliminary review of TVA's design criteria for FSAR Chapter 15 safety-related systems within the scope of the DBVP.

NRC IR 50-327, 328/86-45 summarizes the NRC's review of TVA's compilation and implementation of the commitment/requirement data base, the design criteria which TVA prepared to support SQN restart, and the adequacy of EA's independent oversight review of commitments/requirements and design criteria.

NRC IR 50-327, 328/86-55 summarizes the NRC's review of the DBVP's ECN review, the adequacy of the associated EA oversight, and the adequacy of TVA's actions regarding findings identified during previous inspections of the DBVP and during inspection of the G/C and TVA "3-system" design control reviews (see IR 50-327, 328/86-27).

NRC IR 50-327, 328/87-14 summarizes the staff evaluation of the System Evaluation Reports (SYSTERs) reflecting the DBVP's integrated assessment of the individual systems within the scope of the program.

Additional inspections (IR 50-327, 328/87-31) of the DBVP also were conducted to assess the adequacy of the corrective phase of the DBVP and corrective actions for related design control inspection findings.

Related NRC inspections (IRs 50-327, 328/87-06 and 50-327, 328/87-27) were conducted to evaluate TVA's assessment of the technical adequacy of calculations, since this aspect was not evaluated by the DBVP. The calculation review program is further discussed in Section 2.3 of this evaluation.

Through these inspections, the NRC has had direct and continual involvement in the monitoring and overview of TVA's design control programs, including the DBVP. NRC inspections have been performed at the corporate engineering offices, contract engineering offices, site engineering offices, and the plant site. All phases of the DBVP program have been monitored through a sampling inspection program including preparation and implementation of reviews, resolution of DBVP and EA findings, implementation of corrective and preventive actions, and verification of corrective and preventive actions. NRC observations and conclusions from these inspections as well as the staff's review of TVA's corrective actions for previous inspection findings have been published in the NRC inspection reports.

2.2.3 Conclusions

TVA initiated the DBVP and EA independent oversight review as part of its effort to correct past design control deficiencies identified by employee concerns and design control reviews, including those identified by G/C, TVA, and NRC. These programs provided substantial additional information that has allowed the staff to conclude that design control problems at Sequoyah are being corrected and that once the defined corrective actions are completed, the plant will conform to its licensing basis. Moreover, the staff agrees with the EA team in that the pre-restart phase of the DBVP has been fully and effectively implemented. However, the staff will review the transitional design control system during its review of the Phase II portion of the DBVP.

2.3 Design Calculations Program

TVA and the NRC have conducted several reviews in the past that have shown inadequate documentation of the calculations supporting the design basis for TVA's nuclear plants. Calculations have been determined to be missing, incomplete, or outdated. TVA's engineering disciplines (nuclear, mechanical, civil, and electrical) have each developed programs to resolve these problems. These efforts include (1) identifying essential calculations; (2) verifying the existence of, or regenerating, essential calculations; (3) ensuring the technical adequacy of these calculations; and (4) ensuring the calculations are current.

Essential calculations are those which address existing plant systems or features whose failure could (1) result in a loss of integrity of the reactor coolant system, (2) result in the loss of ability to place the plant in a safe shutdown condition, or (3) result in a release of radioactivity off site in excess of a significant fraction of the 10 CFR 100 guidelines.

The sections below discuss the calculations review efforts for the various disciplines. The NRC has conducted inspections in this area in coordination with the review of the DBVP. These inspection activities are discussed in IRs 50-327, 328/87-06, 87-27, and 87-64.

2.3.1 Nuclear and Mechanical Calculations

TVA's Nuclear Engineering Branch (NEB) and the Mechanical Engineering Branch (MEB) reviews implemented each of the objectives of the DNE calculation review effort.

To establish the list of essential calculations, NEB developed a list of calculations necessary to support the nuclear design and compared this list to the files of existing Sequoyah calculations. The existing calculations were identified as essential, desirable, file only, or superseded. All classification information was captured and verified in the calculation cross-reference information system (CCRIS) computer data base.

As a result of this effort, NEB identified a total of 395 essential calculations. Of these, four were identified as missing. Two of the missing calculations were required for plant restart and were regenerated.

To assess technical adequacy of the essential calculations, NEB initially took a sampling approach except for the calculations performed by the Safety Systems Section, which are primarily calculations used to support FSAR Chapter 15 accident analyses. The critical safety evaluations performed by Safety Systems Section received a 100-percent review. As a result of a random sample in the other sections, NEB determined that there were numerous errors in the pre-1985 calculations performed by the Radiation Protection Section. Additional samples were taken in this area as a result. The scope of the review program also was expanded when it was found that the initial sample selection did not address calculations supporting modifications reviewed by the DBVP nor those calculations performed by the NEB located at the site. As a result of deficiencies identified during these reviews, NEB decided to perform a technical adequacy review of the remaining essential calculations.

NRC inspections monitored the implementation of the nuclear calculation review effort. These inspections noted that the NEB calculation review had identified 30 unacceptable calculations (of which 21 were essential). These have been corrected with no effect on hardware. The staff considers that there is a high confidence that essential nuclear calculations needed to support the Sequoyah design are in place.

To establish the list of MEB essential calculations, a general list of calculations necessary to support the mechanical design of a nuclear power plant was developed. MEB determined that 111 calculations were "missing" from the total set of 397 calculations determined as essential to the Sequoyah design. The staff noted that several calculations listed in the calculation log were obsolete or superseded. Therefore, MEB had to regenerate the missing calculations and identify the controlling calculations. The missing calculations were all regenerated. No equipment or hardware changes were required as a result of regenerating these calculations.

MEB initially sampled 55 previously existing essential calculations to assess their technical adequacy. Six of these were determined to be unacceptable: three in the heating, ventilating, and air conditioning area involving improper heat load input and three in the area of heat exchanger analysis involving inadequate use of vendor data for calculations involving "off-design" conditions. These calculations were identified as common-cause deficiencies and the subject calculations were revised. As a result of the number of unacceptable calculations and a lack of examination of calculations associated with the DBVP, an additional set of 22 calculations was reviewed for technical adequacy. Seven additional calculations were identified as unacceptable (these calculations were then revised). TVA then decided to perform a technical adequacy review of the remaining essential calculations.

TVA contracted with Stone & Webster Engineering Corporation (SWEC) to perform this additional review. Results of this review were provided in TVA's Task Completion Report SQTCR 008-1, Revision 0, "MEB Calculation Technical Adequacy Review." This report was reviewed by the staff in IR 50-327, 328/87-64. Of the 335 calculations reviewed, all but five were considered acceptable. The five remaining calculations were in the process of being corrected pursuant to TVA's condition adverse to quality process, with no anticipated impact on Sequoyah restart. SWEC concluded that the MEB calculations that were reviewed were generally of high quality and supported the Sequoyah design basis.

The essential mechanical calculations have been entered into the CCRIS to data base to establish a consolidated calculation and cross-reference log.

NRC inspections monitored the implementation of the mechanical calculation review effort. Although one additional calculation regarding HVAC adequacy during a loss of all ac power was considered missing, the staff considers that there is a high confidence that calculations needed to support the Sequoyah design are in place.

TVA's engineering assurance organization conducted in-process technical reviews of the calculation reviews. NRC inspections observed this oversight and considered it to be effective in monitoring and controlling the calculation review.

Deficiencies, which were identified during the calculation review efforts, are being tracked for resolution by TVA's condition adverse to quality (CAQ) process. The staff determined that TVA was appropriately applying the documented restart criteria for scheduling necessary corrective actions.

The staff concluded that the nuclear and mechanical engineering calculation review effort has been adequately defined and implemented to identify the necessary essential calculations for the operation of Sequoyah; that the technical adequacy of the calculations has been adequately demonstrated; and that necessary corrective actions are being scheduled in accordance with the documented restart criteria. Therefore, the staff finds the TVA actions for resolution of NEB and MEB concerns acceptable.

2.3.2 Civil Calculations

During its review of civil engineering calculations, TVA determined that a large number of rigorously analyzed pipe support calculations were not retrievable. Accordingly, TVA initiated a program to regenerate these calculations. In support of this program, TVA developed a criteria document, SQN-DC-V-24.2, to define in detail the FSAR requirements to which all safety-related pipe supports will eventually be upgraded. The staff has evaluated these criteria and determined that they are acceptable for restart (February 23, 1988 letter). The staff will be performing additional evaluations of standard component supports as a post restart item.

Additional criteria were developed to establish priorities for implementation of pipe support modifications identified by this review program. These restart criteria are presented in criteria document CEB-CI-21.89 (see TVA letters of August 31 and November 17, 1987(a)). The staff approved the criteria with certain restrictions in a letter to TVA dated February 23, 1988. All supports must satisfy the restart criteria before restart of Sequoyah; the present schedule for compliance to the long-term criteria is the end of cycle 4 for Unit 2 (see October 6, 1987 submittal).

Some problems were found in other civil engineering areas as well. These are noted in the inspection reports on the calculation program and will be addressed by the staff as post-restart items. In addition, the NRC staff's IDI identified a number of issues with TVA's civil calculations. These issues have been resolved by the staff for Sequoyah restart. The details of the resolution of remaining items in the civil calculation area are discussed in IRs 50-327, 328/88-12 and 88-13. The staff will provide an additional detailed evaluation of the civil engineering calculation program in a staff post-restart inspection report. All pre-restart items in the civil calculation area have been resolved.

2.3.3 Electrical Calculations

2.3.3.1 Introduction

As a result of deficiencies first identified to TVA by INPO after its audit on the Bellefonte and Watts Bar nuclear plants, and later confirmed by TVA during the Bellefonte electrical evaluation and quality assurance audit, and as a result of a number of employee allegations, the staff was concerned about the adequacy of the electrical system design at Sequoyah. Because of this concern, TVA reviewed the design calculations at Sequoyah and found the deficiencies listed below:

(1) the minimum set of electrical calculations required to support the Sequoyah plant design was not available;

(2) procedures controlling design changes were not fully adhered to;

(3) existing calculations were not considered when design changes were made; and,

(4) existing calculations that did not require change were not formally documented.

TVA believes that the majority of calculations required for the design were prepared informally during the design period. As a result, calculations were not officially documented or controlled, and those that were documented were not kept up to date.

Because of these deficiencies, TVA reviewed all the existing electrical calculations. TVA then established an electrical calculations program to ensure that the Sequoyah electrical system design meets all requirements for safe startup and operation and to document the adequacy of that design. This program requires necessary electrical calculations to be performed and design control procedures and a design change review program to be established. Moreover, TVA contracted with the Sargent & Lundy Company (S&L) to perform an independent assessment of its electrical calculations program. This assessment was to provide additional assurance that all the electrical calculations necessary to support plant restart have been identified and are existing, current, retrievable, and technically correct. S&L would also identify any additional electrical calculations necessary to fully document the design basis of the plant.

In late 1985, TVA identified a minimum set of electrical calculations that need to be in place and up to date to support Sequoyah restart. During January 14-16, 1986, the staff visited the Sequoyah site to review a draft scope of the minimum set of electrical system calculations and evaluate whether the scope included all pertinent onsite power system calculations necessary to support restart. The staff also assessed the adequacy of calculations with regard to approach, level of detail, and documentation. Each TVA system reviewer responsible for a particular analysis was present during the visit to explain the assumptions, methodology, and sources of data. The staff was provided with samples of the calculations and the documentation so that it could evaluate the calculations.

Subsequently, on February 27, 1986, TVA submitted a report entitled "Electrical Calculations Program for Sequoyah Nuclear Plant." This report provided a brief discussion of the Sequoyah electrical calculations program and presented the analyses for the systems listed above. Moreover, the report addressed the problems TVA found with these systems. These findings are documented in a series of significant condition reports (SCRs) that had been initiated to complete the required corrective actions.

TVA stated that additional information would be forthcoming to discuss the corrective actions taken for each SCR. This information was submitted on August 1, 1986, when TVA provided its review of all the SCRs and a description of corrective actions to be taken. An assessment by S&L of the Sequoyah electrical calculations program also was included. On the basis of its review, TVA acknowledged that revisions to the electrical calculations and related formal documentation for the APS, I&CS, and raceway systems would be necessary before restart.

On the basis of comments made in the S&L Report made in response to TVA's submittal of February 27, 1986 describing its Sequoyah Electrical Calculations Program and NRC inspection findings during the DBVP inspection (IR 50-327, 328/86-55), the scope and detail of the minimum set calculations were markedly increased. The scope and results of this program were described in a TVA letter, Gridley to Youngblood dated December 29, 1986. This letter also provided status information on disposition of discrepancies already identified in the calculations program.

The NRC reviewed this revised program both by field inspections during February 2-13, 1987 and by review of the program and of specific calculations in Bethesda. These reviews are documented in IR 50-327, 328/87-06 dated April 8, 1987 and in a NRC letter, Youngblood to White dated February 10, 1987. Also TVA's internal Engineering Assurance group produced two audit reports, EA 86-23 and 97-09. Based on comments in these documents, TVA again revised the scope and methods of the Electrical Calculations Program. The scope and results of this program were documented in a TVA letter dated June 12, 1987 which also provided updated status on correction of deficiencies identified already.

The NRC continued its review of the Electrical Calculations program during field inspections in June and October 1987 which are documented in IR 50-327, 328/87-27 dated August 24, 1987 and IR 50-327, 328/87-64 dated February 23, 1988. Certain electrical calculation areas were identified by the NRC staff as particularly critical and were reviewed in detail by the Reactor Operations Branch of the TVA Projects staff and its consultants. The NRC staff's evaluation of these areas is documented in subsequent subsections of this report and include the following specific areas:

(1) Auxiliary Power System (APS)

- load analysis
- voltage calculations
- Class 1E motor control center (MCC) control circuit and cable length calculation
- diesel generator load analysis

(2) Control Power System

- 125-volt dc vital instrument power system voltage calculations
- 120-volt ac vital instrument power system voltage calculations

(3) Instrumentation and Control Systems (I&CS)

- instrumentation accuracy calculations including seismic effects

(4) Raceway Systems

- justification for use of TVA's ampacity tables and justification for TVA's ampacity tables as specifically applied to control level cable trays, grouped conduits, conduits with more than three cables and duct banks

As noted in its June 12, 1987 submittal, TVA was verifying previously unverified assumptions to delete non-conservative design cable lengths, and correct deficiencies identified by DBVP and the as-constructed drawings review. By letter dated February 18, 1988, TVA provided a status report which noted that corrective actions required for Unit 2 restart are complete.

In its February 18, 1988 letter, TVA reported that its minimum set electrical calculations program was complete, provided updated status information on the correction of deficiencies identified in the calculation program and identified additional calculations added to the minimum set that had been completed to resolve open calculation issues. The letter stated that, with the exception of the calibration of level indicators of the refueling water storage tank, which is required for post-accident monitoring, all deficiencies. The exception is acceptable to the staff because TVA committed to, and the staff accepted it previously, as a post-restart item that will be completed by the Cycle-4 refueling outage.

Based on its review of TVA's minimum set calculation program, the staff concludes that the program has resulted in a group of electrical calculations sufficiently complete, technically correct, current and retrievable to to support restart of Unit 2. With the exception of those deficiencies identified in the electrical calculations program which are not required to be corrected until after Unit 2 restart, the staff concludes that the program has accomplished its purpose. The staff notes that TVA has committed to expand and formalize its calculation control program over the long-term to cover all calculations, not just those identified as the essential minimum set. The staff relies on this commitment as the most effective means to assure that TVA's electrical calculations required to assure safety are maintained in the acceptable condition that the present program has established. Further, this conclusion of general adequacy of the electrical calculation program does not extend to Unit 1 restart. This restriction arises for the following reasons:

(1) A number of calculations do not assume two unit operation and require upgrading to support Unit 1 operation.

(2) A number of deficiencies identified as required for restart have been completed for Unit 2 but not for Unit 1.

Lastly, there are a number of deficiencies designated to be corrected after restart and there are a number of long-term programs TVA has committed to undertake after restart. These are listed in the various documents cited above. Expeditious completion of these long term commitments was assumed in the staff's evaluation of the adequacy of Sequoyah's electrical calculations program.

2.3.3.2 Evaluation

Each system calculation is complex and requires in-depth knowledge of Sequoyah system operation. Therefore, the staff reviewed the analysis of each system to determine if it was complete relative to the stated purpose, if the assumptions were appropriate, if the applied methodology was correct, and if the results were reasonable to ensure the adequacy of electrical calculations and of documentation. The staff's individual evaluations are discussed below. The staff also audited other calculations including lighting systems and grounding.

2.3.3.2.1 Auxiliary Power System

(1) APS Load Analysis

Before determining the adequacy of APS voltages through calculation, TVA conducted an APS loading analysis for the 6.9-kV unit boards and the 6.9-kV and 480-volt ac Class 1E boards to account for and to document the power distribution equipment loading profiles for normal operation, full-load rejection, emergency shutdown, and cold shutdown. This load analysis is to be maintained and updated as a Q/A controlled document. For each mode of operation, TVA reviewed the latest as-built drawings and system functional diagrams to determine the loads on each board. The loads were further identified as being either off, running, starting, delayed starting, or delayed tripping according to each operating mode. For the minimum load condition at cold shutdown, an actual measurement of the load was taken 92 hours after a normal shutdown. The load analysis listed all the equipment, its operating status according to its operating mode, and the load represented by the equipment. The sources of information included the single-line diagrams, schematics, and design drawings. The staff reviewed the APS loading analysis and found the sources and the documentation to be complete and the analysis format to be appropriate for use in the voltage calculations.

Therefore, the staff concludes that the load analysis is comprehensive, sufficiently detailed, and acceptable to be used as the basis for board loadings for the steady-state and transient voltage calculations. The NRC staff notes that this load list was verified by walkdown in late 1986 and early 1987. Changes identified by this effort were incorporated in the various calculations which depend on the load analysis as input. The NRC staff reviewed this effort as part of its DBVP inspection and IDI. Deficiencies identified have been corrected and the load analysis is acceptable.

(2) APS Voltage Calculations

TVA performed APS voltage calculations to determine and document the following:

° steady-state voltages at 6.9-kV switchgear buses for unit startup, full-load operation, normal shutdown, and emergency shutdown with maximum and minimum unit generator/offsite power supply voltages

° transient voltage profiles at all Class 1E APS buses and safety-related motor terminals for design-basis conditions and minimum offsite power system voltages

- transient and steady state voltage profiles for all Class 1E buses and motors for operation on emergency diesel-generators with no offsite power available.

- optimum power transformer voltage tap settings

- adequacy of present degraded voltage relay set point selection

TVA used basic software packages that were developed in house and that are run on personal computers to calculate the APS voltages described above. (The validity of the computer software was evaluated by the staff as discussed in Section 2.3.4 of this report and found acceptable for use in the APS voltage calculations.) These computer packages are listed below:

- RADIAL was used to calculate transient/steady-state voltage at all 6.9-kV unit and shutdown boards interfacing with the plant from the grid.

- VOLT was used to calculate transient voltage at each 480-volt ac Class 1E board and to sum the 480-volt ac system board loadings for use in the 6.9-kV system calculations.

- VOLT 2 was used to calculate 480-volt ac level steady-state voltage. It determined the starting and running voltage of every load for the condition of minimum source voltage and maximum bus loading.

TVA developed cable and load data files based on the APS configuration, cable parameters, and the loads determined by the loading analysis to perform the APS voltage calculations. These data files were used in the computer programs listed above to calculate the APS voltages. The results are shown in Tables 2.1 and 2.2.

TVA performed the load analysis to ensure that the voltages on the 6.9-kV shutdown boards (Class 1E) would remain within the degraded voltage set points (6560 volt to 7260 volt) and all 6.9-kV Class 1E motors would have adequate starting and running voltage. The results of the analysis indicated that during operation of either or both units (1) the acceptable range for the 161-kV grid voltage would need to be from a minimum of 159 kV to a maximum of 166 kV for each common station service transformer (CSST) with taps set at 0.975 (-2.5 percent) and (2) the main generator voltage should be limited to 24.8 kV to limit the 6.9-kV shutdown board voltage to 7260 volts during normal operation. This, in turn, sets the unit station service transformer (USST) tap at 1.025 (+2.5 percent). The results of the analysis also showed that the worst-case scenario of maximum load would result from a full load rejection (FLR) for Unit 1 with a safety injection actuation signal (SIAS) and a Phase B containment isolation for Unit 2 because the containment spray pumps (700 horsepower) would start.

TVA stated that the results of the 6.9-kV plant/grid interface voltage calculations showed that there was no need to change the degraded voltage set point for 6.9-kV Class 1E shutdown boards and that all 6.9-kV Class 1E motors will have adequate starting and running voltages.

Table 2.1 Calculation Results for the 6.9-kV Class 1E Shutdown Boards

Case	Tap (%)	Grid Voltage	Range of Shutdown Board*	
			Time	Voltage
Maximum Load - Unit 1 - full load rejection, Unit 2 - safety injection with Phase B isolation	CSST** at -2.5	159 kV, min.	T=0 sec T=10 sec T=2 min	6118 to 6574 6631 to 6692 6718 to 6915
Minimum Load - cold shutdown	CSST at -2.5	166 kV, max.		7245 to 7262
During normal operation	USST*** at +2.5	Main generator at 2.48 kV		7212 to 7245

* Time delay trip set point for degraded grid voltage for the 6.9-kV shutdown boards is set at 10 seconds at 6560 volts.

** CSST - common station service transformer.

*** USST - unit station service transformer.

Table 2.2 Calculation Results for 6.9-kV Class 1E Motors for Maximum Load Case

Motor	Starting terminal voltage per unit	Required starting voltage per unit*	Steady-state running voltage per unit
Auxiliary Feedwater Pump 1A	0.951	0.765	0.969
Essential Raw Cooling Water Pump K-A	0.858	0.765	0.954
Auxiliary Feedwater Pump 2A	0.884	0.765	0.960
Containment Spray 2A	0.883	0.765	0.960
Residual Head Removal Pump 2A	0.884	0.765	0.961
Safety Injection Pump 2A	0.884	0.765	0.961
Centrifugal Charging Pump 2A	0.883	0.765	0.960
Essential Raw Cooling Water Pump QQ-A	0.857	0.765	0.963
Centrifugal Charging Pump 1A	-	0.765	0.969
Press HTR Group 1D	-	0.765	0.970
Essential Raw Cooling Water Pump R-A	0.857	0.765	-

* Rated at 80 percent.

TVA further acknowledged that the deficiencies (SCR SQNEEB 8607) found with regard to individual component voltages in the Class 1E 480-volt ac boards would occur during a degraded voltage condition. TVA presented the following corrective actions to resolve this problem: (1) delay two component cooling system pumps for a period of 20 seconds after receipt of an SIAS and (2) modify the 480-volt ac supply to the main feedwater isolation valves so that the electrically operated brakes are wired independently. TVA stated that the time delay was analyzed and found consistent with the plant design basis and that the necessary modification (ECN L6648) has been authorized. The resolution for the main feedwater isolation valves involves the installation of eight new cables and eight new solenoid valves that will operate at 80 percent of voltage. TVA has stated that these corrective actions have been completed for Unit 2 and have been scheduled as a post-Unit 2 restart item for Unit 1.

The staff has reviewed the corrective actions proposed by TVA and agrees that the deficiencies are resolved with these system changes made. Where TVA has included specific time delay devices to ensure adequate voltage, these devices should be included in the Technical Specifications for operability and surveillance. The staff finds this resolution acceptable.

The NRC staff also reviewed APS voltage performance when operating on the emergency diesel generators. First, the staff agrees with TVA that steady state performance on the EDGs is bounded by the offsite degraded voltage analysis and is therefore acceptable based on the above. However, based on test data discussed in subsection (4) below, the staff could not agree that the APS load analysis bounded the APS bus and motor voltage performance during the loading sequence of the diesel generators. The staff therefore required TVA to conduct margin analyses to demonstrate the adequacy of APS voltage during loading. These analyses included the following:

° Minimum and maximum bus voltage

° Margin to motor stall at minimum voltage

° Ability to accelerate all motors in allowable times at minimum voltage

° Ability to operate MOVs in requisite time at minimum voltage

° Misoperation of control and overcurrent protective devices in over and under voltage conditions

Although APS voltage did not remain in all cases within the Regulatory Guide (RG) 1.9 limits to which TVA committed in its FSAR, the staff concluded that the APS would perform its safety function and was acceptable. This finding was based on the staff's review of TVA's margin analyses identified above. Therefore, the staff finds the APS voltage performance while supplied from the EDGs to be acceptable for restart. The staff believes that these voltage fluctuations arise from use in the EDGs of exciter/regulator systems with slower dynamic response than those of a more modern type. For the permanent corrective action, the staff relies on TVA's February 29, 1988 commitment to undertake, after restart, an engineering evaluation and modification of the EDG exciter/regulator system so as to improve EDG transient voltage response in the long-term.

On the basis of its review of the APS voltage calculations, the staff concludes that the calculations are complete and acceptable and that adequate (steady-state and transient) voltage will be available at all Class 1E APS buses and motor terminals for all design-basis conditions with maximum and minimum unit generator/offsite power supply voltages. This conclusion of acceptability is limited to Unit 2 restart. Acceptability for Unit 1 operation will require completion of the 480-volt ac actions described above and revision of TVA's EDG load analysis to remove the assumption that Unit 1 is in cold shutdown.

(3) Class 1E MCC Control Circuit and Cable Length Calculations

To determine the ability of the Class 1E MCC control circuits to pick up the control devices (e.g., valves, starters, relays, solenoids) under the worst degraded voltage conditions, the licensee calculated the voltage profiles to these control devices from a supply bus (480-volt ac shutdown board) powered from the worst-case 6.9-kV board (at 6118 volts) upon initiation of an SIAS. To perform these calculations, the licensee identified all Class 1E circuits that are fed from Class 1E MCCs and reviewed control power transformer size, starter size and load parameters, cable lengths, and wire sizes. The cable lengths were increased by 15 percent over the design length as a conservative measure for the voltage calculations. As part of DBVP, these values were compared with installed lengths and the more conservative values were used for the calculation.

The minimum control voltage value used as acceptable criteria for the majority of the starters was 93.5 volts (85 percent of 110 volts). For Allis-Chalmers starters, the minimum control voltage value of 102 volts (85 percent of 120 volts) was used. These calculations showed 38 circuits to have a control voltage value of less than 93.5 volts, and analysis showed that no adverse effect would result if the energizing of these circuits is delayed for 15 to 30 seconds. The effective components and the planned time delay for each affected circuit(s) are given below:

° hydraulic injection valves (HIVs) on the upper-head injection (UHI) system with a delayed closing of 15-20 seconds (one circuit)

° various cooling and exhaust fans in the auxiliary building with a delayed start of 30 seconds (36 circuits)

° diesel engine heat exchanger-inlet control valve in the ERCW system with a delayed opening of 30 seconds (one circuit)

In its review of the APS voltage calculations for the worst degraded voltage conditions (i.e., 6118 volts), the staff noted that the voltage recovers to 6631 volts at 10 seconds when the trip set point for degraded grid voltage has been set at 6560 volts with a time delay of 10 seconds. Since the planned delays are long relative to the anticipated duration of the degraded voltage condition, the staff finds that the planned time delays for the sustained degraded voltage condition do not represent a safety concern. The planned time delays are acceptable for the reasons stated below:

° The staff reviewed a recent Sandia study (TRAC-PF1/MOD 1 dated January 29, 1986) of the failure of the upper-head accumulator shutoff valve that results in an injection of nitrogen into the vessel during a design-basis accident. The calculations from this study demonstrated that "...because of the extra water injected into the vessel by the upper head accumulator, failure to close the upper head accumulator shutoff valve is slightly beneficial with respect to cooling the core." Thus, it was concluded that "...there is no significant displacement of vessel water by the incoming nitrogen and the nitrogen that does enter the core does not seriously hamper reflood." Thus, an increase in the delay from four seconds to 15-25 seconds on the upper-head accumulator shutoff valve is acceptable.

- A 30-second delay in starting the cooling and exhaust fans in the auxiliary building will not adversely affect the safety-related equipment in the rooms and is acceptable.

- The diesel generator engine will not overheat by starting and running from a standby condition for 30 seconds without ERCW flowing to the diesel engine heat exchanger and is acceptable.

During its review of EDG test data documented in subsection (4) below, the NRC concluded that, although the degraded off site voltage condition did conservatively envelope voltage conditions during steady state operation of the EDGs, it did not envelope transient voltage conditions that occurred during sequenced loading of the EDGs. As a consequence, the NRC required TVA to analyze the performance of those specific MCC control devices that must operate during the loading sequence.

TVA's analyses of transient MCC control voltage concluded that, in general, almost all contactors and associated MCC control devices that were required to operate during the loading sequences would not be exposed to voltages below their design minimum for pick up and drop out. However there were two models of contactors manufactured by Allis-Chalmers and by Arrow-Hart that would be exposed to voltages below their design minimums. With more detailed review, TVA determined that the Allis-Chalmers contactors were used in manually-controlled circuits in the ERCW system that would not be in use during the sequence.

To justify the acceptability of function of the Arrow-Hart contactors TVA first determined the actual minimum contactor pick up and drop out voltage by laboratory test. TVA then calculated minimum voltage at the contactor coil terminals to determine margin for drop out and pick up. For drop out, TVA provided a bounding analysis that assumed maximum cable length, worst-case device loading and minimum MCC control transformer size. This analysis showed a bounding margin of 26% between minimum voltage seen during the loading sequence and drop out voltage.

In analyzing pick up, TVA first reviewed the detailed loading sequence to identify those Arrow-Hart contactors that were slaved to sequenced loads and therefore would be required to pick up at the minimum voltage caused by starting the associated large motor. TVA then calculated the actual minimum voltage at the contactor coil terminals using the actual cable lengths, additional control devices and minimum bus voltage that would occur during the specific step. The worst case was determined to be an MOV slaved to the auxiliary feedwater pump which starts at 20 seconds in the sequence. The margin between actual minimum voltage at the contactor coil terminals and required pick up voltage was 9.7%.

During pick up, a contactor coil draws a large amount of current and there is an increased possibility of blowing control fuses. Therefore, the NRC staff required a re-examination of MCC control fusing. TVA's re-examination identified the most critical circuit as a room cooler fan starting at time zero. The analysis showed 60% margin to fuse actuation during the one second delay associated with the low voltage pick up.

Based on the TVA analyses of performance during EDG sequencing, the staff concludes that adequate margin exists to assure proper operation of the MCC control circuits during EDG operation. The staff also notes that TVA has committed after restart to evaluate and upgrade the EDG exciter-regulator system which will improve EDG transient performance and therefore increase the stated margins.

On the basis of its review of the Class 1E MCC control circuit and cable length calculations and TVA supplemental analyses of performance with EDG transient loadings, the staff concludes that the Class 1E MCC control circuits can pick up the control devices under degraded voltage conditions.

This evaluation is limited to Unit 2 operation since the EDG load analysis, on which the analysis of MCC control performance during sequencing depends, assumes Unit 1 is in cold shutdown. The load analysis and the voltage from it on which this evaluation depend will require recalculation for two unit operation.

(4) Diesel Generator (DG) Load Analysis

In response to a number of employee concerns alleging generally that the Sequoyah diesel generators were overloaded, that load additions were not properly controlled and that frequency and voltage did not meet FSAR commitments, TVA performed a DG load analysis to determine the sequential loading and capability of each DG to start each load at the time required within acceptable voltage and frequency limits. TVA prepared a computer data base to show all loads connected to the power distribution boards that would be powered by the DG following a total loss of offsite power. The data base was developed by using as-designed logic and schematic drawings of the circuit operations for the various design events. All the loads on each power train were sorted and coded according to the time of start and/or stop. This load list/sequence is now being maintained and updated as a TVA QA controlled record. The accuracy of this list was verified by walkdown and the validity of walkdown data was inspected by the NRC during the DBVP inspections. TVA considered the following three possible accident conditions:

° a total loss of offsite power (LOOP)

° LOOP with concurrent SIAS-Phase A containment isolation

° LOOP with concurrent SIAS-Phase B containment isolation

For each of the accident conditions, TVA determined the sum of the loads, in horsepower kilowatts and kilovars, from 0 to 120 minutes for each of the four power trains.

TVA's independent DG contractor, Morrison-Knudsen Company, Inc. (M-K), further analyzed and evaluated the worst-case loading scenario to determine the capability of the DG to accept and carry sequenced and random loads within allowed voltage and frequency limits. Based on this analysis, in its August 1, 1986 submittal, TVA stated that a problem existed should random loads be running or started concurrent with the sequenced DG load (i.e., 700-horsepower containment spray pump) at the 30 second step. The random loads are automatic process loads that could be initiated at any time by temperature, level, or pressure. As a conservative approach, the random loads were considered as a block load applied with each sequence step; this resulted in a worst-case condition when the containment spray pump started at 30 seconds. The worst-case loading occurs for a LOOP with an SIAS Phase B containment isolation. Train 2B was the worst for all three cases. For all three cases the contractor concluded that generator 2B would be able to load at the required time and do so within an acceptable voltage and frequency limit for all times except at the instant the containment spray pump starts at 30 seconds.

To correct this problem, in its August 1, 1986 submittal, TVA proposed an intentional time delay of eight 480-volt ac loads to maintain the maximum load within the value of 4482 kW at the 30-second time. These loads include four supplies to the 480-volt ac board room air conditioning system (a part of the random loads which would be delayed for 2 minutes and 30 seconds) and four supplies to 125-volt dc vital battery chargers that charge the four 125-volt dc Class 1E batteries. (Delaying the loading of the 125-volt dc vital battery charger for 5 minutes poses no problem since the 125-volt dc vital batteries are designed to carry plant emergency loads for 2 hours during a LOOP.) The staff found that this time delay of the eight 480-volt ac loads would maintain DG 2B within the acceptable limits of loading.

However, TVA advised the staff, at that time, that the accident conditions for a LOOP with a delayed SIAS were being analyzed and that this would result in a revised DG load analysis. TVA submitted a revised DG load analysis, Revision 3, that included the three delayed SIAS sequences on December 29, 1986. However, TVA informed the staff by letter dated March 12, 1987, that "TVA is not evaluating these sequences because the delayed safety injection and loss of offsite power cases do not significantly contribute to the probability of core melt." The NRC Power Systems Branch Review Reminder No. 11 dated April 20, 1983, concluded that the frequency of core melt occurring as a result of a delayed SIAS following a LOOP is sufficiently low to exclude this series of events from consideration as a credible core melt initiator. Therefore, the staff agrees with TVA that these conditions need not be considered as a design event.

Also, in its March 12, 1987 letter, TVA stated that its previous resolution did not sufficiently reduce the transient load; thus, additional actions would be necessary. These additional actions were made necessary by an error in TVA's assumption of DG load limits in its analysis. These errors were identified by M-K. M-K pointed out that during the first three minutes of operation, the superchargers on Sequoyah's diesels are not operating at full capability. Therefore, the engine behaves like a naturally aspirated (non-supercharged) unit. A naturally aspirated engine is rated for operation at or below 90°F. Sequoyah's maximum ambient temperature is 97°F. This difference resulted in a derating of the engine in the first three minutes for which TVA had not accounted.

During a March 26, 1987 meeting, TVA provided a revised DG load analysis (Rev. 5) and proposed the following new actions to correct the problem:

- Change the load sequence time delay for the containment spray pump (CSP), CSP circulation fan, and containment spray header isolation valve from 30 seconds to 3 minutes.

- Delay starting of the electric board room air handling unit (AHU) for 220 seconds.

When these changes were implemented, the maximum load profiles were recalculated by TVA. The NRC staff reviewed the accident analysis consequences of these changes and approved them in License Amendments 59 and 51 issued September 18, 1987.

TVA consulted with M-K and, with the information provided in the contractor's report (No. 6957R, Rev. 1), transmitted by letter dated February 27, 1987, TVA proposed the DG ratings discussed below. During the first three minutes of operation, due to ambient temperature derating the diesel engines are limited to 4320 kW. After three minutes, the engine can be operated for two hours at its short time rating of 4840 kW and 4400 kW for periods of operation in excess of two hours. Further, the generators are limited to a total electrical load of 5500 kVA for two hours and 5000 kVA continuously. These limits, stated as separate engine and generator limits, are equivalent to the combined limits of 4400 kW for two hours and 4000 kW continuously at the power factor of 0.8 that were stated originally in the FSAR. The restatement takes advantage of the fact that TVA's loads run at a power factor larger than 0.8 and, therefore, the total load is generally controlled by the engine limit rather than by the generator. It should be noted that the FSAR did not recognize the derating during the first three minutes of warm up.

The NRC staff reviewed and approved this restatement of DG load limits. This review was documented in License Amendments 64 and 56 issued January 7, 1988. This change to the Technical Specifications increased the DG limits and correspondingly increased the loads at which the engines were to be tested during periodic surveillance.

On the basis of its review of Revision 5, the staff found that the load at each sequence step was below the DG ratings except for the steady-state rating case. TVA has stated that operator action will reduce the steady-state loading for this case and has provided a list of nonessential loads that can be shed by the operators and also that there will be a procedure (AOI-35) to reduce DG load to within the steady-state rating. The staff found this action acceptable. The staff has determined that the DG can start all the equipment within acceptable voltage and frequency limits.

In August 1987, TVA completed Revision 6 to its DG load analysis. This revision incorporated walkdown data on motor loads and cable lengths as well as a number of minor corrections. It also added several appendices containing confirmatory calculations requested by the staff. The impact of the various changes on loading results was trivial. The revised calculation was reviewed by the staff during a Calculations Program inspection in October 1987; no problems were noted.

As noted above, the NRC, as part of its review of the DG revised load limits, had approved an increase in the loads that the diesel were required to be tested with in the plant technical specification surveillance requirements. As part of the revised requirements, TVA was to load test the diesel generators at the new higher loads prior to restart. These tests were also used to validate the continued acceptability of the plant's preoperational test program. These surveillance tests were conducted during the period from July to November 1987.

In January 1988, TVA identified to the NRC data from these surveillance tests which raised significant questions about the operability of the EDGs at Sequoyah. These results were interpreted by TVA as indicating both a possible defect in one generator's (2A) exciter system and a more general problem in all generators in conforming with voltage limits during loading as stated in RG 1.9. A failed component was replaced in the exciter system of the 1A generator which corrected the first problem and left only the more general voltage problem. A detailed review of the test data by the NRC identified the following significant issues relevant to the second problem:

° the test results were worse than would be predicted by the calculational methods used to model diesel generator performance

° the test data when extrapolated to post-accident conditions showed that the diesel generators had less margin in terms of voltage behavior than calculations had predicted

Because of these issues, the NRC required TVA to undertake a major analytic effort with the following objectives:

° to identify the reasons why calculations did not predict the severity of the experimental results

° to improve the calculational methods to provide greater assurance that the calculational methods would conservatively predict post-accident behavior

° to quantify the margins available between diesel capability and specific electrical power system requirements in post accident operation. The specific margins in question were the following:

(1) minimum voltage during the loading sequence;

(2) diesel-generator power rating (kVA and kW);

(3) motor performance (starting and stalling);

(4) contactor performance (pick up and drop out);

(5) motor operated valve performance (torque and timing);

(6) overcurrent protection misoperation (circuit breakers and control fuses); and,

(7) sequence timing error (overlap and maximum loading time).

These required analyses and calculations were submitted to the NRC on March 1, 2, and 11, 1988. Based on its review of the results of these calculations the NRC reached the following conclusions:

- The most likely reason that the original calculations did not bound the test data was that generator voltage did not stabilize between successive steps in the loading sequence.

- The generator's inability to stabilize voltage was probably caused by the use of a voltage regulator lacking the speed and performance characteristics typical of those used in modern nuclear applications.

- The most recent analysis methods used by TVA bound the experimental data and predict the DGs behavior in post accident loading with adequate margin. However, this margin is less than was expected when the plant was licensed.

- The margin that remains is sufficient to assure safe operation of Sequoyah for restart and for the limited period of time until corrective action is take to re-establish the margin that was believed to exist at the time of licensing.

In a March 3, 1988 submittal, TVA committed to evaluate the performance of the EDGs and implement corrective action prior to restart after the next Unit 1 refueling outage. This schedule is acceptable to the NRC staff.

TVA also addressed the concern (SCR SQNEEB 8646) that voltage would fall below the 75 percent minimum stated in RG 1.9 and not recover within the specified time interval if the DG breaker closes at 80 percent of nominal voltage. TVA stated that this occurs because the 6.9 kV shutdown board DG supply breaker is designed to close at 80 percent of nominal voltage with the diesel running at 850 rpm. Since the present voltage relay used to control the supply breaker could not be reset higher with precision to correct the situation, TVA deliberately lowered the rate at which engine speed builds up. This provided greater time for voltage to increase and correct the voltage problem. However, this led to the incidental result that the frequency at breaker closure is below the ± 5% frequency stated in RG 1.9. The NRC has examined the frequency trace during test and determined that frequency continues to increase after breaker closure with its time zero loading at the same constant rate as before closure and reaches the allowable range in less than one second. The NRC concludes this deviation from the regulatory guide to be incidental, of no safety significance.

In examining the voltage test results discussed above, TVA and the NRC noted that voltage in at least one case exceeded the ± 10% voltage recovery limit stated in RG 1.9. This limit requires that, during loading, voltage recovers to ± 10% within 60% of the step interval. It should be noted that in RG 1.9, Revision 0, which TVA committed to in its FSAR, the requirement is 40% of the interval. However, the NRC found this to be unnecessarily restrictive and relaxed the requirement in subsequent revisions of the RG. Therefore, the NRC staff will not hold TVA to the unnecessarily restrictive limit. The NRC staff concludes that the very short (less than 1/2 second) overvoltage has no safety significance, has been adequately addressed by TVA, and is acceptable for restart. The staff notes that improvements to the excitation system discussed above would improve the DG performance.

As part of its March 1, 1988 submittal, TVA provided Revision 7 to its DG loading calculation for staff review. This revision slightly rearranges the 20 and 30 second steps to assure two major motors could not start at essentially the same time because of timer error and drift and therefore overload the engines. The change has no significance in the design basis accident analysis.

It should be noted that the major TVA calculations on which the staff's findings are based assume that Sequoyah Unit 1 is in cold shutdown and must be revised to support Unit 1 restart. Further, the staff notes its reliance on TVA's commitment to undertake, after restart, a major review and modification effort to improve performance of the DG regulator/exciter system.

2.3.3.2.2 Control Power System

(1) <u>125-Volt DC Vital Instrument Power System Voltage Calculations</u>

TVA performed the 125-volt dc vital control power system study to determine if there is adequate voltage available at the terminals of the selected components to continue proper operation during a loss of ac power. TVA performed voltage calculations for a representative sample of typical circuit types and categories because there are 600 safety-related circuits. TVA selected 35 circuits and classified them into the categories listed below:

° 6.9-kV shutdown board control circuits

° 480-volt ac shutdown board control circuits

° fuse column circuits (primarily solenoid valve circuits)

° auxiliary relay rack circuits

° reactor trip switchgear breaker control circuits

° 120-volt ac vital inverter feeder circuits

TVA analyzed the sample circuits by calculating the voltage available at the terminals of the loads and comparing this voltage with the manufacturer's minimum voltage rating. If a problem was identified in any of the categories, all the circuits in that category were evaluated. The staff finds this acceptable since the representative sample chosen was based on a worst-case approach.

To calculate the maximum voltage drop, a cable length of either the construction pull length or design length plus 30 percent was used with the cable temperature at 90°C. For the latter four categories above, the vital battery 2-hour discharge minimum terminal voltage of 105 volts dc was used. However, for the former two categories, the calculations were performed with a battery voltage of 120 volts dc. TVA based this assumption on Sequoyah's design criteria which state that the voltage shall be 120 volts dc. Because of the automatic undervoltage load shedding feature, the critical operational period for the 6.9-kV and 480-volt ac shutdown boards is immediately upon loss of ac power, i.e., battery voltage of 120 volts dc. The staff concurs with TVA's asumption since these mandatory loads will occur during the initial discharge phase of the battery duty cycle and each operation lasts only a fraction of a second. In addition, the battery is not expected to be discharged to a level of 105 volts dc since the diesel generators are designed to supply power to the chargers within a few minutes of loss of offsite power.

In its February 10, 1986 calculation (SCR SQNEEB 8605), TVA identified inadequate minimum dc input voltage to the 120-volt ac vital inverters on Unit 1 per the manufacturer's specification. The original vendor minimum input voltage specified for these inverters was 105 volts dc. Subsequently, the inverter vendor has performed a recertification test for the same type of inverter at TVA's Watts Bar and confirmed that the Sequoyah Unit 1 inverter will also operate properly at a 100-volt dc minimum, thus eliminating the concern. Two other problems surfaced as well: (1) inadequate dc input voltage for 24 solenoid valves associated with the steam dump system during a minimum vital dc system voltage condition (105 volts dc), and (2) excessive voltage drop (based on the manufacturer's data) for two flow-modulated solenoid valves between the modulator (valve controller) and the valve during any dc system voltage. As noted in a letter dated August 1, 1986(a), TVA stated that (1) the operation of these 24 valves is not required for safe shutdown, and (2) a further review by the manufacturer has found that adequate voltage is available for the flow-modulated solenoid valves.

On the basis of its review of the 125-volt dc voltage calculation along with the additional clarification, the staff finds that adequate voltage is available for proper operation during a loss of ac power and no further corrective action by TVA is required.

(2) <u>120-Volt AC Vital Instrument Power System Voltage Calculations</u>

The purpose of the 120-volt ac vital control power system study was to determine if the safety-related 120-volt ac loads powered from the 120-volt ac vital instrument power boards have adequate voltage for proper operation. TVA reviewed all safety-related loads for Units 1 and 2 and identified a total of 166 such safety-related circuits. These circuits were classified into four groups (i.e., relay, valve, monitoring, and instrumentation and control circuits) according to the type of load served. The voltage calculations were performed on a representative sample of each group (at least 10 percent). If the evaluation identified no failures in a group, a high degree of confidence was achieved and no further evaluation was performed. If a failure was identified, then the voltage calculation for every circuit in the group was performed.

The inverter (power source) was assumed (worst case) to be operating at full load with a maximum output (125 amp) and minimum output voltage of 117.6 volts (120 volts minus 2 percent) with a phase angle of 41 degrees. The voltage available at the terminals of each component supplied by the inverter was calculated and its adequacy determined by comparing with the manufacturer's minimum voltage rating. The cable lengths of either the construction pull length or the design length plus 30 percent were used with the cable temperature at 90°C. In those cases where a component could be energized by an alternate path, the path that produced the largest voltage drop was used in the calculation.

A preliminary TVA study, Revision 0, dated December 27, 1985, showed that eight circuits from three groups (i.e., valves, monitors, and instrumentation and control) have excessive voltage drop. These circuits were identified for corrective action, and further voltage drop analyses were performed on all the circuits in those groups. A new analysis dated January 30, 1986, identified a total of 12 circuits with excessive voltage drops that were documented for corrective action under SCR SQNEBB 8532.

The staff concurs that the use of such a sampling technique can be justified in determining the adequacy where a large number of circuits are involved. Further, this type of categorization sampling technique can be a useful tool to identify and localize problem areas in circuit design; therefore, the staff finds this technique acceptable.

TVA found that the above 12 circuits were divided into three groups: (1) radiation rate meters within the monitoring group, (2) post-accident sampling in the valve group, and (3) reactor vessel level instrumentation in the instrumentation and control group. TVA stated that corrections for these deficiencies would involve pulling larger size cable to reduce cable impedance and paralleling supply cables to reduce the current through various portions of the affected circuits. Those corrective actions required for restart of Sequoyah Unit 2 have been completed.

On the basis of its review of the 120-volt ac calculations and TVA's proposed corrective actions for resolving the identified deficiencies, the staff concludes that the safety-related 120-volt ac loads powered from the 120-volt ac vital instrument power boards will have adequate voltage for safe operation.

2.3.3.2.3 Instrumentation and Control Systems Instrumentation Accuracy Calculations

The NRC staff and its consultant, Science Applications International, reviewed a sample of 15 TVA instrumentation and control calculations for Sequoyah for technical accuracy. Guidance to prepare instrument set point calculations and to maintain set point accuracy that is needed to fulfill the design basis requirements of IEEE Standard 279-1971 is provided by IEEE Standard 603-1980, RG 1.105, and Instrument Society of America (ISA) Standard S67.04-1982.

The scope of the review calculations was generally limited to determining the expected accuracy of a safety-related set point as a result of the effect of harsh environment conditions imposed on individual instrument loop components. The reviewed sample did not include each type of calculation ordinarily prepared by an instrumentation and control design group. Specifically, the reviewed calculations did not establish an actual set point value for the instrument channel, nor did they generally address the set point accuracy of safety-related instrument loops subject only to a mild environment condition. Instrument set points are established by the mechanical/nuclear calculations. The NRC staff accepts TVA's assertion that accuracy for instruments that are not exposed to a harsh environment has been demonstrated by the operational experience at Sequoyah.

The calculations reviewed generally addressed the worst-case predicted accuracy or variability of an established safety-related process set point. The objective of a set point accuracy calculation was to determine the statistical allowance of an instrument channel. The expected performance of an instrument channel could then be assessed for conformance with process set point limits.

The methodology employed in the determination of the instrumemt channel statistical allowance was the square root of the sum of squares (SRSS) of individual effects such as those listed below.

- environmental allowance
- process measurement accuracy
- primary sensor element accuracy
- sensor calibration accuracy
- sensor measurement and test equipment accuracy
- sensor drift
- sensor temperature effect
- sensor pressure effect
- rack calibration accuracy
- rack measurement and test equipment accuracy
- rack comparator setting accuracy
- rack drift
- rack temperature effect

Several special cases of calculations involving analog control loop stability, instrument process line response time, and effects of radiation exposure were provided in the reviewed sample. The following calculations were reviewed:

(1) pre-operational tests in lieu of calculations for control loops (auxiliary feedwater stability calculation) (RIMS B43 860915 925 R0)

(2) instrument accuracy calculation for 1-PT-68-69 (RIMS B43 860809 901 R2)

(3) instrument accuracy calculation for 1-TE-68-1, -18, -24, -41, -60 and -83 (RIMS B43 860805 913 R3)

(4) response time of sensing lines (RIMS B43 861106 904 R1)

(5) set point scaling calculation for PDT-65-80, -82, -90, and -97 (RIMS B43 850830 903 R0)

(6) solenoid valve arc suppression networks located in harsh environment (RIMS B43 860619 901 R1)

(7) demonstrated loop accuracy for high-range radiation monitor
(RIMS B43 860624 914 R2)

(8) HVAC instrument accuracy evaluation
(RIMS B43 860829 917 R0)

(9) demonstrated accuracy calculation for O-LDT-67-470, -477, -482, and -487
(RIMS B43 860915 910 R0)

(10) demonstrated accuracy calculations for 1-PS-3-139A, B, and D and for 1-PS-3-144A, B, and D
(RIMS B43 860915 912 R0)

(11) verification of retrievability for isokinetic equipment calculations
(RIMS B43 860826 902 R0)

(12) control valve sizing retrievability review
(RIMS B43 860917 912 R0)

(13) safety-related flow elements locations
(RIMS B43 860915 917 R0)

(14) demonstrated accuracy calculation for 1-PS-3-148, -156, -164, and -171
(RIMS B43 860915 916 R0)

(15) filter design for PT-30-310 and -311
(RIMS B43 861022 901 R0)

The staff reviewed these calculations and requested additional information for calculations (1), (5), (6), and (8). Other calculations were either fully acceptable or were acceptable with minor comments. The staff met with TVA on August 19, and November 30 through December 2, 1987, to resolve staff concerns.

During these meetings, TVA presented revised information for calculations (1), (6), and (8). Calculation (5) was replaced with (RIMS B43 860917 919). The revised and new information for calculations (1) and (5) were acceptable to the staff. Revised information for calculations (6) and (8) is discussed below.

Calculation (6) - ARC Suppression Network

This calculation did not properly address the seismic integrity of the majority of arc suppression networks. Therefore, the staff concluded that the arc suppression network could fail during a seismic event. The TVA assumption that these devices are needed for only one cycle and therefore need not be seismically qualified is indefensible. TVA acknowledged the seismic integrity issue in the meeting and stated that the seismic qualification of these arc suppression diodes will be resolved and the arc suppression networks will be seismically qualified. TVA has submitted, by letter dated February 29, 1988, confirmation that the arc suppression diodes are seismically qualified.

Calculation (8) - HVAC Instrumentation Accuracy Calculation

TVA does not have any documentation to confirm the seismic qualification of the HVAC instrumentation. TVA has taken the approach that, after a seismic event, the plant staff will perform a physical walkdown to ensure that instrumentation is operable. TVA did not provide any procedures for ensuring instrument operability after a seismic event and did not establish acceptance criteria for determining what constitutes instrument degradation.

TVA also indicated that some instruments are required to have 5 percent accuracy, but it was unable to provide a calculation for the instrument set point and process safety limit values. The staff pointed out that HVAC set points (RIMS B44 871015 006) had recently been established at 90 percent of full range and that this may be inconsistent with the 15 percent accuracy limits.

TVA has acknowledged the NRC concerns and stated that it will revise the calculation and address the seismic threshold limits, specify the HVAC equipment to be inspected after a seismic event, provide an inspection procedure, and clarify the calculation accordingly. NRC staff considers this solution to be acceptable based on TVA's confirmatory response dated February 29, 1988 but does not believe the solution needs to be implemented prior to restart.

On the basis of its review, the staff finds the TVA instrument accuracy calculations to be satisfactory. TVA documented the proposed resolution to staff concerns in calculations (6) and (8) in the confirmatory letter dated February 29, 1988.

2.3.3.2.4 Raceway Systems

The staff evaluated TVA's justification for using its ampacity tables and the justification of these tables as applied to control level cable trays, grouped conduits, and conduits with more than three cables and duct banks.

INPO performed an audit in 1986 on the Bellefonte plant that revealed inadequacies in TVA's electrical design standards DS-E12.1.1 through DS-E12.1.4. These standards have been used to size all the insulated power cable ampacities (auxiliary and control) throughout TVA's nuclear plants. This finding, later confirmed by TVA's Bellefonte electrical evaluation team, was identified as a generic problem. By a report dated February 27, 1986, TVA described an analysis it has performed to demonstrate the adequacy of design standards DS-E12.1.1 through DS-E12.1.4. After reviewing both the standards and the supporting calculations, TVA concluded that the standards were incomplete and lacked the definition and information required for proper application. These deficiencies in design standards were identified in TVA Problem Identification Report (PIR) GENEEB8605.

By letter dated December 23, 1986, TVA informed the staff that design standards DS-12.1.1 through DS-E12.1.4 were superseded and that the new electrical design standard, DS-E12.6.3, "Ampacity Tables for Auxiliary and Control Power Cables (0-15,000 volts)," corrected all the inadequacies. The new standard also addresses ampacities for cable in conduit, cable tray, and duct bank as well as derating factors for cable coatings; 10 CFR 50 Appendix R, fire wraps; cable tray covers; and cable tray bottoms. TVA's submittal also presented the following information regarding the standard.

- Electrical Design Standard DS-E12.6.3 for sizing cables with regard to ampacity was developed in accordance with recognized industry standards on ampacity, i.e., Insulated Power Cable Engineers Association (IPCEA) P-46-426, National Electrical Code (NEC) Article 310 (1987), IPCEA P-54-440, and Institute of Electrical and Electronics Engineers (IEEE) 70 Tp 557 PWR.

- The cable ampacity derating factors for fire protective cable coatings, tray covers and/or bottoms, and Appendix R fire wraps are based on test reports from the manufacturers of the coating and wrapping material.

- The standard was developed utilizing TVA and Stone & Webster Engineering Corporation expertise.

- The standard was reviewed and found acceptable by Bechtel Power Corp.

- The methodology has been reviewed against and found to be consistent with the standards of Sargent & Lundy and Gilbert/Commonwealth.

Rather than examine each electrical cable to determine its adequacy with respect to ampacity ratings established under DS-E12.6.3, TVA developed a sampling program. All the cables were categorized into nine inspection lots according to their operating voltages, cable routings, covers, and wrappings. Each cable, counted only once, was included in the inspection lot reflecting the most limiting raceway configuration for ampacity in which it was routed. The nine inspection lots are listed below:

(1) V3-level cables routed in tray

(2) V3-level cables routed in conduit without Appendix R fire wrap

(3) V3-level cables routed in conduit with Appendix R fire wrap

(4) V4-level cables routed in tray without tray covers, bottoms, or Appendix R fire wrap

(5) V4-level cables routed in tray with tray covers, and/or bottoms, and/or Appendix R fire wrap

(6) V5-level cables routed in tray without tray covers, bottoms, or Appendix R fire wrap

(7) V5-level cables routed in tray with tray covers, and/or bottoms, and/or Appendix R fire wrap

(8) V4- and V5-level cables routed in conduit without Appendix R fire wrap

(9) V4- and V5-level cables routed in conduit with Appendix R fire wrap

The definitions of the three voltage levels are given below:

V3 - auxiliary and control ac and dc power cables operating at a voltage of up to 277 volts and a current of less than 30 amperes

V4 - auxiliary ac and dc power cables operating at a voltage up to 600 volts (This includes cables of 277 volts or less with a rated load current of 30 amperes or greater.)

V5 - medium voltage auxiliary power cables with a nominal rated voltage of 5, 8, or 15 kV

TVA established a separate engineering group to identify all the cables in each respective lot. This group reviewed all the cable trays and conduit drawings (as-built) to verify the existence and location of tray covers and/or bottoms, and Appendix R fire wraps. This survey was performed under "Walkdown Procedures for Ampacity (SMI-0-317-41)." Once all the cables in each lot were identified, the group determined a sample size for each lot by using the Military Standard 105D dated April 29, 1963, "Sampling Procedures and Tables for Inspection by Attributes." Among the chosen samples, the group determined the allowed ampacity of each cable by applying the derating and correction factors specified in DS-E12.6.3. The group evaluated the adequacy (pass/fail) of the cable ampacity by comparing the allowed ampacity and the actual ampacity, which is based on the full load current multiplied by appropriate factors according to load types (i.e., motor, transformers, heater). If the total number of defective cables found in each sample was less than the maximum (4 percent) specified by the military standard, the group considered the lot adequate. The failed cables were documented in a significant condition report (SCR) for corrective actions.

On February 27, 1987(c), TVA submitted the following results:

(1) V3 voltage level

Although this voltage level is restricted to control cables operating at a voltage up to 277 volts and a current of less than 30 amps, the great majority of cables in the V3 level carry low-level and/or intermittent signals for which the ampacity rating of the cable is of no concern. TVA provided justification and documentation (including supporting calculations) for excluding this group of cables (control function cables) from this program. Thus, TVA separated those V3 voltage level cables that require consideration as possibly being auxiliary "control power cables" (Inspection Lots (1), (2), and (3)) from those "control function cables" used for controlling the operating status of equipment. The sampling program was used to establish the extent of inclusion of control power cables in Lots (1), (2), and (3) and the adequacy of their ampacity rating. These results are given below:

Lot No.	Total No. of Cables	MS per 105D Sample Size	No. of Cables Sampled/ Analyzed	No. of Control Power Cables Found	No. Passed
1	5919	50	376	1	1
2	3331	52	693	4	4
3	3	3	3	0	0
Totals:	9253	105	1072	5	5

TVA sampled 1069 cables out of the 9250 cables for Lots (1) and (2). Analysis of the 1069 selected cables from these two lots showed only five cables that carried sufficient current to be considered as potentially having an ampacity problem. However, these five cables were found to be adequately sized in accordance with DS-E12.6.3. None of the three cables in Lot (3) carried sufficient current to be considered a problem. TVA found that the number of cables routed in V3-level raceways carrying other than very low and intermittent currents was substantially less than previously anticipated. Since all those control power cables analyzed presented no problem and since there were not enough sample cables carrying high currents in this voltage category, as required by he military standard, TVA performed no further evaluation.

(2) V4 and V5 voltage levels

The V4- and V5-level cables had a greater tendency to have a problem with ampacity because of the higher current levels and the practice of providing less conservatism in sizing high-power cables. TVA found that too many cables in Lots (4) through (9) did not pass the acceptance criteria (failed); therefore, additional power cables (100 percent) had to be inspected. Lots (4), (5), (6) and (9) received a 100% inspection of cables. For Lots (7) and (8), only 10 CFR 50.49 and associated cables were 100% inspected; the remaining cables in the lots were subject to a sampling approach. TVA identified 457 cable failures from these inspections; the results are provided below:

Lot No.	Total No. of Cables	No. Passed	No. Failed	No. to be Replaced
4	407	269	138	12
5	568	277	291	103
6	29	21	8	0
7	47	47	0	0
8	384	366	18	8
9	11	9	2	2
Totals:	1446	989	457	125

TVA used the criteria listed below to evaluate each failed cable:

- Tray covers and bottoms that were not required for personnel or cable protection or to meet licensing commitments were removed.
- The allowable cable ampacity was recalculated on the basis of existing tray fill.
- The actual load current was determined on the basis of existing connected loads.
- The load type multipliers were modified to reduce the ampacity margin by removing excessive conservatism.

With this approach, TVA found that 332 of the 457 failed cables were within allowable ampacity and therefore acceptable. The other 125 (a combined total from both units) will be replaced before restart of the applicable unit.

TVA's revised DS-E12.6.3 is based on industry standards and provides various derating factors that are applicable to the specific installed cable configurations. The staff finds DS-E12.6.3 acceptable for use in resolving the TVA ampacity problem at the Sequoyah units.

The staff finds that Military Standard 105D is not sufficiently well defined to obtain a 95/95 assurance level (i.e., giving 95 percent assurance that at least 95 percent of the population is acceptable). The staff believes that the proper sample size should have been determined by using the hypergeometric distribution function, which provides larger samples than the military standard. However, as discussed below, the actual sample size taken in the field exceeds the requirements of either the Military Standard or the hypergeometric distribution. Thus, this issue is moot.

However, for the V3 voltage level (Lots (1), (2), and (3)), TVA sampled a far greater number of cables than required by either approach. Since only five control power cables were found through an inspection of 12 percent of the V3 voltage cables and since these five cables were within the allowed ampacity, the staff finds that the sample size for the V3 level is acceptable and that these cables do not constitute a problem area.

A similar sampling process was conducted for the V4 and V5 voltage levels (Lots (4) through (9)). As a result of this inspection, 125 cables will be replaced before restart of Unit 2. Furthermore, TVA informed the staff that 108 new cables currently are being repulled while the others are being de-energized and/or removed because they are not being used to support operation of Unit 2. This will provide a 100/100 assurance level for the V4 and V5 cables.

Based on its review of the TVA submittal and the resolution of identified deficiencies in PIR GENEEB8605, the staff finds that the problem areas have been adequately identified and that the proposed corrective actions are acceptable.

However, the above acceptability was contingent upon resolution of two unverified assumptions. These are the accuracy of (1) the cable schedule data base and (2) the installed thickness of fire protective cable coating. The staff verified the accuracy of the cable schedule data base through inspections conducted during the DBVP inspection and IDI programs. The installed thickness question has been resolved because TVA presented calculations during the DBVP inspection that demonstrated, for the geometries at issue, the maximum temperature was bounded at an acceptable level for all reasonable thicknesses.

2.3.3.2.5 Short-Circuit Study - Medium Voltage System

2.3.3.2.5.1 Background and Analysis

In a letter dated December 29, 1986, TVA submitted electrical calculations for Sequoyah Nuclear Plant Unit 2, including short-circuit studies of the medium voltage system. The NRC, with its contractor, Science Applications International, conducted an independent technical evaluation of selected samples of the TVA Sequoyah electrical calculations. TVA issued Revision 1 to this calculation on June 1, 1987; the revision was reviewed by the NRC and its consultants during the calculation program and IDI from August through October 1987. Additional information on circuit breaker capability and analysis of calculational results was provided by TVA in a letter dated August 10, 1987. This section provides a description and NRC's evaluation of the adequacy of short circuit capability of Sequoyah's medium voltage system as described in these submittals.

The medium voltage system consists of non-Class 1E and Class 1E 6.9-kV switchgear, circuit breakers, and associated electrical equipment designated as startup boards, unit boards, and shutdown boards. The 6.9-kV shutdown boards in each power train derive power from either of two 6.9-kV unit boards or from their respective standby power source (diesel generator). The feeders connecting each shutdown board with these three sources are termed the normal, alternative, and standby feeders. The normal and alternate feeders can derive power from the nuclear unit, via separate unit station service transformers and separate 6.9-kV unit boards. The normal and alternate feeders for each bus can also derive power from separate preferred source circuits, routed through either of two separate common station service transformers and from either of two 6.9-kV unit boards. During conditions where neither the nuclear unit generator nor the preferred (offsite) power is available, each 6.9-kV shutdown board is energized from a separate standby diesel generator via the standby feeder. The standby ac power system is a safety-related Class 1E system that continuously supplies power for energizing all ac-powered electrical devices essential to safety. Power continuity to the 6.9-kV shutdown boards is maintained by switching among the nuclear unit source (the normal source), the preferred (offsite) source, and the standby (onsite) source. Source selection is accomplished by automatically transferring from the nuclear unit source, to the preferred source, to the standby source, in that order. The reverse transfers are manual.

To analyze short circuit capability, TVA selected fault locations within the 6.9-kV system for analyzing the short circuit current values to assess the capability of the installed equipment from the standpoint of fault protection. The faults were calculated on each 6.9-kilovolt unit board bus, and each 6.9-kV shutdown board. A three-phase bolted fault for each fault location was selected by TVA for purposes of calculating the maximum available fault current.

Since the 6.9-kV system is grounded through low impedance resistors, ground fault current is limited; this eliminates the need for making single-phase to ground fault calculations. Therefore, the bolted three-phase fault should yield conservative fault current estimates to determine the adequacy of the interrupt and withstand capability of the installed 6.9-kV circuit breakers. The purpose of the three-phase short-circuit current calculations was to determine the maximum value of short-circuit currents to establish the adequacy of the latching (asymmetrical current) and interrupting capability of the installed 6.9-kV system switchgear and circuit breakers.

The staff evaluated the design of 6.9-kV supplies and equipment against the requirements and recommendations of the documents normally used in the design of electrical power systems for nuclear power plants. Specifically, the requirements of General Design Criterion (GDC) 1, GDC 17, NUREG-0800 (Sections 8.2 and 8.3), RG 1.32, and IEEE-308 were compared to the Sequoyah electrical design. In addition, industry standards, such as ANSI 37.06-1964 and ANSI 37.010-1979, which are normally used for sizing electrical switchgear and equipment, were compared against the installed switchgear at Sequoyah to verify their ratings and capacities. This safety evaluation is based on the licensee's submittals and discussion with the licensee regarding the 6.9-kV switchgear.

The methods and assumptions used by TVA for calculating three-phase short-circuit currents at the 6.9-kV switchgear locations are reasonable and consistent with industry practice. Specifically TVA used the bases and recommendations of ANSI 37.010-1979. Both the modeling and assumptions used in making the three-phase fault calculations are appropriate and in conformance with good engineering practice. The fault current values obtained from these calculations provide the basis for sizing electrical switchgear and determining the withstand and interrupt capability of the circuit breakers. These calculated current values indicate the worst-case for the bolted three-phase electrical fault at each fault location. Good engineering practice in conformance with industry standards dictates that the electrical equipment specified for these locations (i.e., the unit boards and shutdown boards) must have a rating equal to or higher than the calculated values. This philosophy and practice is used industry wide to provide added conservatism to accommodate the normal aging and service of the equipment and any increase in load after installation.

The selection and application of power circuit breakers for ac power systems, such as the Sequoyah 6.9-kV system, had been standardized by ANSI-37.06-1964. This standard is intended as a guide for the selection and application of the circuit breakers by the user. In its submittal of December 29, 1986, TVA indicated that the Sequoyah design was based on ANSI-37.06-1964. According to this standard, the 500-MVA class circuit breakers and the shutdown boards would be rated for 8.25 kilovolts maximum voltage with a short-circuit current rating of 33,000 amperes (i.e., 471 MVA) and the switchgear would have a momentary rating of 60,000 amperes. In its August 10, 1987 letter, TVA provided the manufacturer's guaranteed performance data, which showed that the circuit breakers and associated buswork have a 500-MVA interrupting capacity rating and an 80,000-ampere momentary rating. These ratings are substantially above the ratings required for 500-MVA class switchgears. However, even using these actual ratings, Sequoyah's 6.9-kV circuit breakers are undersized, relative to TVA's most recent short-circuit calculation, by as much as 35 percent on the unit boards (through which unit generator and offsite power is routed to the safety boards) and by 8 percent at the shutdown (safety) boards. Similarly, the unit boards are undersized by 3 percent for momentary withstand capability.

In its August 10, 1987 letter, TVA noted that the calculation methods of ANSI C37.10, used for determining required capacity, assume a three-phase bolted fault on the bus and calculate the total available fault current on the bus from all sources. According to TVA, for the bus arrangements at Sequoyah, these assumptions are conservative for the main feeder breakers on both the unit and shutdown boards because the feeder breakers are not required to interrupt the fault contribution from the downstream loads. Appendix B to Revision 1 of the short-circuit calculation shows that the actual faults these breakers would be required to interrupt are 543.6 MVA for the unit board feeder and 494.6 MVA for the shutdown board feeder. It should be noted that in this particular calculation, TVA included the impedance of the buswork from the unit station transformer to the unit bus. The NRC staff and consultants have reviewed this calculation and conclude that it is technically correct.

ANSI 37.010-1979 states that it is necessary for the circuit breakers that are installed for a given voltage service to have a minimum of at least 100 percent capacity as compared to the maximum calculated fault values. Therefore, on the basis of the ANSI 37.010-1979 and ANSI 30.06-1964 criteria, the 6.9-kV circuit breakers installed at Sequoyah are undersized for the available fault currents. In its submittal of December 29, 1986, and in FSAR Section 8.1, TVA committed to meet the requirements of the appropriate regulations and industry standards and practices relating to the design of the 6.9-kV electrical power system. In particular, the requirements of GDC 1, GDC 17, RG 1.32, IEEE Standard 308, and ANSI Standards 37.010 and 37.06, address this issue. The relative criteria are stated below:

> GDC 1 - Quality Standards and Records: "Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function."
>
> GDC 17 - Electric Power Systems: "Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies."
>
> IEEE 308-1974 (Endorsed by RG 1.32) - Criteria for Class 1E Power Systems for Nuclear Generating Stations, Section 5.2.1(6), Protective Devices: "Protective devices shall be provided to limit the degradation of the Class 1E power systems."
>
> ANSI 37.010-1979 - Application Guide for AC High-Voltage Circuit Breakers Rated on Symmetrical Current Basis, Section 4.5, Short-Circuit Rating: "In the application of circuit breakers, it is necessary that none of the short-circuit current capability of a circuit breaker be exceeded."

In discussions with the staff and in its letter of August 10, 1987, TVA has indicated that although the interrupting capability of the unit board circuit breakers is well below the available fault currents, it is willing to assume a commercial risk in operating the plant for a limited time. TVA stated that it has instituted action to lower the available fault current at the unit boards in the long term. Where the shutdown board breakers (Class 1E) are concerned, TVA has indicated that the one-time end-of-life test is good enough to permit the use of these breakers. The one-time test results indicated that the contacts were heavily damaged, that the chutes were at the ultimate limit of their capacity, and that the breaker had reached its end-of-life condition. Although the vendor furnished the breaker test data to TVA, the vendor has refused to certify 550 MVA as the qualified rating for these breakers.

2.3.3.2.5.2 Evaluation and Conclusions

On the basis of TVA's most recent data, the NRC staff calculated the maximum fault levels to be expected on the most heavily loaded buses of the Sequoyah 6.9-kV system and the system's capability to handle such faults. This analysis assumes a maximum pre-fault voltage of 7.26kV. This voltage is consistent with a 24.8kV maximum grid voltage, a 1.025 tap setting on the USST and, when either unit is operating, a .975 CSST tap setting. These values are controlled by TVA procedure and their correctness was verified by NRC staff review. The staff analyzed the unit boards, which are the 6.9-kV boards that are not safety grade and that are fed from the station service transformers and that, in turn, feed the vital shutdown boards. The incoming breakers to the unit boards, from either the unit or reserve station service transformers, could be required to interrupt a fault as high as 545 MVA. This exceeds the breaker's design rating of 500 MVA and approaches, but does not exceed, the tested interrupting value of 550 MVA. The individual load breakers on the unit boards could be required to interrupt as much as 600 MVA when the diesel is operating, which is well above either the rated or tested capability of the circuit breaker. Even when the diesel is not operating, the unit board load breakers would be required to interrupt more than 560 MVA. As part of its review, the NRC staff also recalculated the momentary fault duty at the unit board, this time taking into account the impedance of the USST bus work. With this impedance added, the staff calculated the momentary fault current to be 80,200 amperes.

The staff analysis of the safety-related shutdown boards showed the feeder breakers would be required to interrupt about 490 MVA (with or without diesel generators), which is slightly below rated capacity. The individual shutdown board feeders would be required to interrupt about 530 MVA when the diesel is operating; this is above the design rating but below the tested capability. Without the diesel running, the shutdown board load breakers will be required to interrupt about 490 MVA.

The momentary asymmetric current on the buswork and unit board circuit breakers, with the diesel generator operating, is at or very slightly above 80 kiloamperes, which is the momentary rating of the switchgear; without the diesel generator operating, the momentary current is about 76 kiloamperes. The momentary current on the safety-grade shutdown boards, for all conditions, is less than 67 kiloamperes.

In summary:

- In normal operation, the non-safety-grade unit board feeder circuit breakers may be required to interrupt a fault greater than designed but less than tested and the unit board load circuit breakers may be required to interrupt a fault significantly above the designed and tested value.

- When the emergency diesel generator is operated in parallel with the unit station service transformer, the non-safety-grade unit board buswork and switchgear may be subjected to physical forces from momentary fault currents slightly above design values.

- During parallel operation, the individual load breakers on the shutdown boards may be subjected to interrupting duty above the design rating but less than the tested capability. The shutdown board feeder breaker will be minimally within design rating.

- During normal operation, the vital shutdown board incoming breakers will be within design rating, but only with 1 to 2 percent of margin.

The staff noted that a less conservative approach than that typically used for design was used for these calculations in that bus and cable impedance was considered and line current rather than bus current was used. However, a three-phase bolted fault in itself is a conservative assumption because real faults tend to exhibit some impedance and some degree of phase imbalance and arcing, all of which tend to lower the fault current below that of the bolted fault. Actual faults also tend to occur most often at loads or in cables rather than at the circuit breaker terminals as was assumed in the calculation. Even a relatively short length of cable between the breaker and the fault would lower all the calculated fault values to less than the breaker's designed capacities.

In the calculations it was conservatively assumed that the diesel generator will be operating in parallel with the system when the fault occurs and that all motors on the involved buses will be operating at that time. The diesel generators only rarely are run in parallel with the system, generally about an hour per month for testing. Because of redundancy, all motors on all the involved buses are seldom run simultaneously. The staff considered all these factors in reaching its conclusions about the adequacy of the system.

The staff reviewed the protection schemes and bus arrangements associated with the switchyard and 6.9-kV distribution system and concludes that no credible single fault on a unit board, even if the incoming breaker failed to clear the fault, will cause cascading failure of the 161- or 500-kV switchyards. Neither will any credible fault on a unit board, even if cascaded to the alternate feed, prevent energizing all shutdown boards from at least one of the station's common service transformers. The staff further concludes that a fault will not be transferred to the alternative source and cannot cause loss of the alternate supply even if the initial fault breaker fails to open. This is because a fault trip signal from an incoming circuit breaker on either the unit or shutdown boards locks-out not only the affected breaker but also the incoming breaker for the alternate supply.

On the basis of its review of the certified performance data and test data submitted by TVA, the staff concludes that even though the circuit breakers were specified and are rated at 500 MVA, the certified performance data would support an interrupting rating of 526 MVA. This results from the breakers being certified to interrupt 44 kiloamperes at 6.9 kV under ANSI 37.04 duty cycle rather than at the normal 6.6 kV. The test data would support a rating of 531 MVA using the ANSI 37.04 duty cycle. Further, the test data provides a reasonable basis for believing the circuit breaker would interrupt a fault up to 550 MVA although ANSI 37.10-1979, Section 4.10.3, warns against exceeding the service capability of the circuit breaker "even if only one interrupting operation may be imposed."

On the basis of the above evaluation of the Sequoyah 6.9-kV electrical system, the TVA submittal, and the Science Applications International Technical Evaluation Report, the staff provides the following conclusions.

The methods and assumptions used by TVA for calculating three-phase short-circuit currents are reasonable and consistent with industry standards and practice. TVA used good engineering basis in modeling the postulated fault locations for evaluation of the 6.9-kV electrical switchgear and circuit breakers.

The staff concludes that the vital 6.9-kV system (the shutdown boards and associated circuit breakers) fault calculations are appropriately conservative and the vital system is in substantial conformance with the applicable regulations, FSAR commitments, and industry standards. The staff bases this conclusion on two major factors. First, the shutdown board load breakers, in the rare circumstance when the diesel is operating in parallel with the system, may have to interrupt a fault larger than the breakers' warranted capacity (500 MVA); however, they will be functioning within their service capability (531 MVA) as defined by appropriate industry standards and demonstrated by appropriate testing. Further, even if one of the load breakers were to fail, the shutdown board incoming breaker would operate within its warranted rating as a backup, thereby preventing fire and severe physical damage to the shutdown board as a whole, or to equipment in its vicinity. Second, the vital 6.9-kV breakers could only be required to operate beyond their warranted rating when, for a few hours a month, the diesel generator is operating in parallel with the unit generator and system. In normal operation, when the diesel is not paralleled with the preferred source, no vital breaker will be required to operate above its warranted design rating. The staff further concludes, from its review of backup breaker operation and lockout capabilities, that the requirement for independence between redundant trains and between alternate offsite supplies is maintained. The staff notes that the NRC calculated fault value for the load breakers to 530 MVA is at the tested service capability of 531 MVA and slightly above the guaranteed values of 526 MVA. The staff notes this lack of margin but believes that the corrective actions discussed below for the non-safety grade breakers will lower the fault level on the vital buses and introduce additional margins.

The staff concludes that the fault calculation for the non-safety grade unit and associated circuit breakers is appropriately and conservatively done and accurately reflects the condition of the non-vital 6.9-kV system. The staff concludes that the 6.9-kV system does not meet the Sequoyah FSAR commitment (Section 8.2.1.8, pg. 8.2.2) that "a fault on a non-safety load supplied from a 6900-volt unit board will be isolated so that the continuity of power to that unit board and to the shutdown board fed from that unit board will not be jeopardized by that fault." The calculations show that a fault on a non-vital unit board load may substantially exceed the rated or tested capacity of the load breaker and will require the unit board incoming breaker to trip, thereby violating the above commitment. In this situation, the unit board feeder breaker operating as a backup breaker will be required to clear a fault greater than either the design rating or tested service capability, thereby violating the intent of ANSI 37.10-79, Sections 4.1.0.1 and 4.1.0.3, another FSAR commitment.

As mentioned above, a reasonable basis does exist for believing that the incoming breaker will clear the fault successfully. Even if it did not, the staff concludes that the switchyard circuit breakers for the unit main transformer would clear the fault by differential protection, thereby protecting the 161-kV alternate offsite source. When the unit board incoming breaker is actuated on backup overcurrent, it locks out the unit board transfer to the alternate source, thereby protecting the alternate offsite source. Also, once the fault is cleared by the switchyard breakers, the unit board can transfer to the protected alternate source and, in turn, power the vital shutdown boards. The combination of these features provide a sufficient basis for the staff to conclude that, until the breaker capacity problem is resolved, a fault on a unit board, coupled with a load breaker failure and an incoming breaker failure, will not result in an inability to supply the vital shutdown boards from a reliable source of offsite power. Therefore the staff concludes that no single fault will disable any more than one of the redundant auxiliary power trains nor will any single fault interrupt offsite power from the preferred and alternate sources to any other bus. This conclusion is independent of whether or not any 6.9kV circuit breaker exposed to the fault fails to clear.

In its letter of August 10, 1987, TVA committed to resolve the problem of unit board breaker capability. This will be done after Unit 2 restart. To ensure that this issue is resolved, the staff requires that a detailed description, analysis, and installation schedule for implementation of corrective actions be submitted for staff review before June 30, 1989. TVA has committed to provide this information. The analysis is to include revised fault calculations for both unit and shutdown boards. These calculations shall demonstrate that after corrective action, all circuit breakers will always operate within their service capability as defined by appropriate standards and verified by test or manufacturer's guarantee. On this basis, the staff concludes that the fault calculation for the 6.9-kV system provides reasonable assurance that the 6.9-kV system will provide sufficient capacity and capability to meet its safety function as defined in 10 CFR 50, Appendix A, GDC 17.

The staff notes that Revision 1 to calculation APS 008, dated June 1, 1987, and submitted to the staff for review includes analysis of Unit 1 and the effect of two unit operation on fault levels. Also the revised calculation reflects verification of technical data on motors and cable lengths based on walkdown data. Therefore, there are no unverified assumptions remaining in the 6.9-kV fault analysis and the analysis as reviewed is applicable and acceptable, subject to the limitations discussed above, for Unit 1 as well as Unit 2 operation.

2.3.3.3 General Conclusions on the Sequoyah Electrical Calculations Program

On the basis of its review of the electrical calculations, the staff finds that

- TVA's analysis includes the essential auxiliary power systems required for safe plant operation.

- The input data is sufficiently comprehensive and detailed for consideration of all modes of plant operation. The calculations assumed worst-case system and plant conditions. The methodology used in these analyses was appropriate for assessing problems in the systems. TVA has stated that it will correct the problems identified in the specific areas before restart.

- TVA's proposed resolutions for each deficiency identified in the electrical calculations are acceptable. TVA has provided a commitment to implement the proposed resolutions before restart.

- The content and format of each system calculation is adequate for documentation purposes.

- All documentation of the electrical calculations necessary for restart is in-place and up-to-date by computer program for easy manipulation (i.e., data is retrievable for maintenance and update).

Thus, the staff believes there is reasonable assurance that the systems addressed will provide safe restart and operation of Sequoyah Unit 2.

2.3.4 Branch Technical Position PSB-1

2.3.4.1 Introduction

The staff reviewed an October 2, 1980 verification test done at Sequoyah in response to PSB-1 requirements and found inconsistencies between the load values used in voltage distribution studies and those determined by the test. In addition, changes had been made in the configuration of the auxiliary power system and in the computer programs used for voltage drop calculations. Therefore, by letter dated March 26, 1986, the staff recommended that TVA perform a new verification test, as prescribed by BTP PSB-1.

During a meeting with the staff on April 16, 1986, TVA presented additional information and clarification to its test report (submitted to the NRC on October 3, 1980) to support its position that additional verification testing of the auxiliary power system was not necessary. Subsequently, TVA agreed to provide those items listed below.

- A confirmatory analysis to demonstrate that the new computer program is comparable to the computer program that was used in the original test report. TVA will use the same load values with the new computer program and compare the calculated voltages.

° Analyses to demonstrate that there is no significant configuration change between the 1980 and 1986 systems. TVA will use the data from the tests of July 12 and 16, 1980, with the 1980 and 1986 system models and compare the calculated voltages.

° More detail on how the two tests (July 12 and 16, 1980) were conducted, and a description of how the circuit breakers were aligned for each.

On June 2, 1986, TVA submitted its response to the staff's concerns and a report entitled "NRC Branch Technical Position PSB-1 Reanalysis." Although the staff reviewed this information, the staff could not conclude that sufficient data were provided to demonstrate that the computer program could predict the transient response of the system. The staff informed TVA of its conclusion by letter dated August 1, 1986, and transmitted additional questions on August 7, 1986. TVA responded by letters dated September 11 and December 3, 1986.

The staff's evaluation of TVA's information on the need for verification testing is presented below.

2.3.4.2 Evaluation

Computer Hardware and Program Changes

The mainframe computer and its VNEW program that were used for the previous verification tests have been replaced by the personal computer and a new program called RADIAL. The staff was concerned whether the new program is equivalent to the old program in analytical techniques and assumptions, and at the April 16, 1986 meeting, the staff asked TVA to provide a confirmatory analysis using the July 12, 1980 test configuration to demonstrate that there is no appreciable difference in the calculated voltages of the two programs.

The TVA comparison analysis was submitted on June 2, 1986, and included a SWEC computer program. TVA ran all three programs using identical loads for each board. The results are given below.

Board		Computer Program (Predicted Voltage) VNEW	RADIAL	SWEC
6.9-kV	Start Bus A	7152	7151	7148
6.9-kV	Start Bus B	7011	7008	7005
6.9-kV	Unit 1B	7011	7008	7005
6.9-kV	Shutdown 1A-A	7004	7002	6998
480-volt	Shutdown 1A1-A	495	495	495
480-volt	Reactor Vent 1A-A	483	483	Not conducted

The staff found no appreciable differences in the voltage values that were obtained from the three computer programs. These results indicate that the analytical techniques and assumptions of both TVA's programs are equivalent for steady-state. However, the test results did not demonstrate the transient response and steady-state at the 120/208-volt level. Thus, the new computer program only has been verified for the steady-state case down to the 480-volt level.

In its August 1, 1986, letter, the staff asked TVA to provide additional justification for not performing the PSB-1 test down to the 120/208-volt level. In its September 11, 1986, response, TVA described the two 120-volt ac control power systems as (1) the 120-volt ac vital instrumentation and control power system (VCPS) fed from the vital inverters and (2) the Class 1E 120-volt ac MCCs supplied from the 480/120-volt control power transformers. For the 120-volt ac VCPS, the vital inverters are designed to maintain the output voltage regulation within +2 percent of 120-volt ac, with an input voltage of 480 volts ac, +7.5 percent. In addition, when the 480-volt ac input is lost (or acceptably degraded) the battery will supply the loads with no interruption of regulated power.

For the Class 1E 120-volt ac vital control power for MCCs, TVA referred to its recent transient voltage calculations, which were performed under worst-case conditions (i.e., the worst expected transient voltage at each MCC) to demonstrate that adequate voltage exists to pick up the control devices (e.g., motor starter, solenoids, and relays) for expected transient conditions.

The staff found that (1) TVA's new computer program can adequately predict the response of the Sequoyah power system down to the 480-volt level, (2) the VCPS through its inverter and battery backup design eliminates the effects from 480-volt ac degraded voltage input or transients, and (3) the worst-case transient calculations indicate that the 480/120-volt ac MCC control power transformers can adequately perform their safety functions.

The staff agreed that the 120-volt ac VCPS design features and the voltage calculations performed by TVA for the worst-case 120-volt ac MCC voltages ensure that adequate voltage will be available to components supplied by the 120-volt ac control power system. Thus, no additional tests to demonstrate system response at the 120/208-volt level are necessary.

Change of 100 Valve Motors

The staff also was concerned that the replacement of 100 valve motors with motors of different electrical characteristics might affect the plant's steady-state load, necessitating a re-analysis of the new system loadings.

However, TVA indicated that this change will affect only the transient loading and voltage; the steady-state load remains the same. Therefore, the staff finds that the change of 100 valve motors represents no overall load increase for the steady-state condition.

Addition of Two Start Buses and One Common Station Service Transformer

The staff expressed concern that TVA had added two new start buses, which could result in new loads or impedance. In response, TVA explained that the buses had not actually been added, but that two start buses had been split into four; thus no new loads or impedances would be added. Although a third common station service transformer has been added, the circuit breakers are normally open, making the transformer available as a backup for either of the other station service transformers. TVA demonstrated that this change has little effect on the overall configuration of the auxiliary power system by comparing the voltage analyses of the 1980 (two start buses) and 1986 (four start buses) configurations. The comparison was performed using the test data of July 12 and 16, 1980, and the new computer program. The results were shown in the Summary Tables I and II of the TVA re-analysis report transmitted by TVA on June 2, 1986. They are summarized below:

Board	Configuration (Voltage)			
	Test I*		Test II**	
	1980	1986	1980	1986
6.9-kV Start Bus A	7154	7155	7045	7041
6.9-kV Start Bus B	7051	7045	7067	7062
6.9-kV Unit 1B	7051	7045	7067	7062
6.9-kV Shutdown 1A-A	7044	7038	7060	7055
480-volt Shutdown 1A1-A	501	500	501	501
480-volt Reactor Vent 1A-A	493	500	494	501
Start of the ERCW pump (Term. V)	Not conducted		6705	6695
Start of auxiliary building exhaust fan 1A	Not conducted		495	458

* Based on data of July 12, 1980.
** Based on data of July 16, 1980.

On the basis of these results, the staff finds that there is no appreciable voltage difference (a maximum difference of 1.5 percent) between the 1980 and 1986 configurations which indicates that the new configuration has not significantly changed the old electrical system configuration.

Re-analysis of the 1980 Verification Test Results

In its response of June 2, 1986, TVA explained how the circuit breakers were aligned for the 1980 verification tests.

TVA had compared the calculated board voltages (based on load values derived from breaker alignment and the supply voltages) with the board voltages obtained from the tests. This procedure deviated from BTP PSB-1 (PART B.4), which requires loads and voltages for a given test configuration to be measured, with these measured load values then used on each board as input to the computer model to calculate the voltages; subsequently, the analytically derived voltage values and the test results are compared. During the meeting on April 16, 1986, the staff asked TVA to perform new analyses using the load values obtained during the tests as input to the new computer program, to be consistent with PSB-1. These results, as given in the submittal of September 11, 1986, are given in the following:

Board		Test I* Voltage			Test II** Voltage		
		Measured	Analysis	Percent	Measured	Analysis	Percent
6.9-kV	Start Bus A	7200	7154	0.6	7000	7045	0.6
6.9-kV	Start Bus B	7000	7051	0.7	7000	7067	1.0
6.9-kV	Unit 1B	7100	7051	0.7	7090	7067	0.3
6.9-kV	Shutdown 1A-A	7000	7044	0.6	7100	7060	0.6
480-volt	Shutdown 1A1-A	495	501	1.2	500	501	0.2
480-volt	Reactor Vent 1A-A	484	493	1.9	489	494	1.0
Start of the ERCW pump (Term. V)		Not conducted			6787	6705	1.2
Start of auxiliary building exhaust fan 1A		Not conducted			466	459	1.5

* Based on data of July 12, 1980.
** Based on data of July 16, 1980.

TVA used board meters, test meters, and Brush recorders for taking test measurements. However, because of calibration problems, the Brush recorder did not yield consistent results and TVA's response of June 2, 1986, did not include measured values from the Brush recorder. TVA also indicated that there were current transformer and power transformer inaccuracies. TVA stated that all measurements were taken by board meters whose accuracy was limited to 5 percent. (Because the staff had not specified the allowed accuracy limit in Position 4 of FSAR Question 8.33, TVA established a 5 percent tolerance as the maximum acceptable difference between the measured voltages and calculated voltages.)

BTP PSB-1 (Part B.4)

TVA had performed the 1980 verification tests at Sequoyah in response to FSAR Question 8.33. Subsequently, BTP PSB-1 (Part B.4) was issued as part of the NRC Standard Review Plan in July 1981. Part B.4 of PSB-1 provides detailed guidance on the performance of verification tests.

Although Question 8.33 does not explicitly include all of the guidance of Part B.4 of PSB-1, it does so by implication. Therefore, the staff evaluation of the 1980 tests was based on establishing a correlation between these tests and the testing and expected test results specified in Part B.4 of PSB-1. In a meeting on April 16, 1986, TVA concurred that the intent of Position 4 of Question 8.33 is the same as that of Part B.4 of PSB-1, even though the PSB-1 requirements are more specific.

The specifics of Part B.4 of PSB-1 are given below:

° loading the station distribution buses, including all Class 1E buses down to the 120/208-volt level, to at least 30 percent

- recording the existing grid and Class 1E bus voltages and bus loading down to the 120/208-volt level at steady conditions and during the starting of both a large Class 1E motor and a non-Class 1E motor (not concurrently) (Note: to minimize the number of instrumented locations (recorders) during the motor starting transient tests, the bus voltages and loading need only be recorded on that string of buses that previously showed the lowest analyzed voltages.)

- using the analytical techniques and assumptions of the previous voltage analyses and the measured existing grid voltage and bus loading conditions recorded during conduct of the test, calculating a new set of voltages for all the Class 1E buses down to the 120/208-volt level

- comparing the analytically derived voltage values against the test results

With good correlation (within 3 percent) between the analytical results and the test results, the validity of the mathematical model used in the voltage analysis is established. However, the above procedure involves testing both the steady-state and transient response characteristics. The transient testing requires starting both a large Class 1E and non-Class 1E motor.

The intent of such a transient test requirement is to detect potential spurious load shedding or separation of a Class 1E system from offsite power when a large motor is started. The ability of the computer model to predict the effects of the motor transient in the system is verified by comparing the data measured during the transient test with the computer-predicted transient values. When both the steady-state and transient analyses are complete, the validity of the mathematical model is verified.

On the basis of its review of TVA's submittal of June 2, 1986, the staff has concluded that there is reasonable assurance that TVA's new computer program can adequately predict the steady-state response characteristics of the Sequoyah auxiliary power system. The staff's findings regarding the transient aspects of the PSB-1 test are given below.

- The test report showed instrument recording problems indicating that starting motor dip values were not reliably established. (There were no transient data for the motor and the Class 1E buses.)

- The selected motor sizes (700 and 150 horsepower) were not large enough to show any significant transient effect (the dip was only for one cycle). BTP PSB-1 (Part B.4) requires starting both a large Class 1E and a large non-Class 1E motor (not concurrently).

- No transient voltage analysis had been performed by comparing the results of calculations performed by the new computer program with the data obtained during the starting of large motors.

In addition to its evaluation of August 1, 1986, the staff also transmitted to TVA on August 7, 1986, a request for additional information on the transient aspects of the PSB-1 test. TVA responded in letters dated September 11 and December 3, 1986. In the absence of an explanation regarding the transient measurements taken during the starting of large motors and how these values were used to determine that the computer model could accurately predict transient effects, TVA provided the Brush recorder traces (voltage and current) taken at the motor terminals for the 460-volt Auxiliary Building General Supply (ABGS) fan and the largest 6.6-kV ERCW pump on the 6.9-kV shutdown board. The measured voltage values for the equipment were compared with the old (VNEW) and new (RADIAL/1980 and 1986 configuration) voltage values calculated from the computer programs. The results are given below:

		Calculated Voltage			
Equipment (hp)	Measured Voltage	VNEW	RADIAL, 1980	RADIAL, 1986	Difference (%)
ERCW pump 700	6787	6763	6703	6695	1.4
ABGS fan 150	466	449	459	458	3.8

TVA found a maximum deviation of 3.8 percent between the measured voltage and the voltage calculated using VNEW and loadings derived from the closed circuit breaker configuration and individual load ratings. The deviation is more than the 3 percent guideline in PBS-1; however, the measured voltages are within 2 percent of the new computer program voltages derived using measured bus load values. Therefore, the staff concludes that TVA's new computer model can accurately predict the transient response of the system.

With respect to the request that it "provide the brush recorder traces of load currents obtained during the motor starting transient tests which were used in the transient calculations performed after the test to predict system bus voltages," TVA provided the measured starting and running currents for Phases A and C of the 6.6-kV ERCW pump and 460-volt ABGS fan. Although the Brush recorder traces included both the voltage and current measurements, the main focus of PBS-1 deals only with the voltages available in the Class 1E buses. Therefore, the measured current values were not used to calculate bus voltages, but were provided to show the actual length of the motor starting transients as opposed to the voltage traces, which changed very little because of the stiffness of the power source. However, the measured Phase A starting currents were used to calculate the first-cycle voltage dips, which were compared with the measured voltage values. The results were found to be the same. In its review of these recordings, the staff found a difference in the Phase A and Phase C running current values, which could be indicative of a phase unbalance condition or a motor abnormality. In addition, if these unbalanced current values were used, they could affect the system bus voltage calculations.

By letter dated December 3, 1986, TVA explained that the differences in the current readings for Phases A and C are not indicative of a phase current unbalance, but result from instrument calibration problems. The fact that no real unbalance existed between Phases A and C was substantiated by a comparison of the board instrumentation meters measuring the same currents. (The board meter readings indicated no substantial difference in currents for Phases A and C.)

The staff has reviewed the recordings of the voltage traces and finds them consistent with TVA's discussion of the motor transients. Thus, the staff finds that no actual unbalance of motor phase currents existed and that the voltage traces are adequate for the PBS-1 analysis.

To address the one-cycle voltage dip experienced during the motor-starting transient test, TVA provided the Brush recorder traces of the terminal voltage and current for the 6.6-kV ERCW and 460-volt ABGS fan, which were obtained during the motor transient tests. The current traces clearly indicated that the acceleration times were about 1 second for the ERCW pump and about 7 seconds for the ABGS fan. From the voltage traces, TVA determined that the 6.6-kV ERCW pump motor start did depress the terminal voltage for approximately the acceleration time (i.e., 60 cycles). However, the measured voltage dip for the 460-volt ABGS fan was for only approximately 6 cycles. For both cases, the worst part of the voltage dip occurred during the first cycle. TVA further found that this corresponds to the instant that the motor rotor is locked and the motor starts to accelerate. TVA also stated that there was no measurable voltage sag at either the 6.9-kV or the 480-volt switchgear buses during motor start.

On the basis of its review of TVA's Brush recorder traces, the staff finds the TVA assessment of the motor-starting voltage transient acceptable.

The staff also expressed concern about whether conservatism was used in calculating the effects of starting large motors. In response, TVA stated: "Our analyses are not a true transient calculation which would show the exponential voltage recovery due to the change in motor impedance while accelerating. Our calculations assume that the voltage dip is at its lowest point for the entire acceleration time of the motor." Further, TVA stated: "Our transient analyses model the 6.9 kilovolt shutdown board voltage depressed at the 1 cycle voltage for the entire acceleration time of the 6.6-kilovolt required starting loads."

The staff finds that the TVA transient analysis model represents a more conservative condition with respect to the motor-starting voltage and its duration for the voltage recovery time. Therefore, the staff concludes that the TVA method for calculating the effects of starting large motors results in a more conservative transient voltage calculation than the exponential voltage recovery that actually occurs during motor acceleration.

The staff asked TVA to provide the worst-case voltage calculation on Class 1E boards during the starting of a reactor coolant pump following an accident. TVA determined the worst case for the 6.9-kV Class 1E shutdown boards was approximately 2 minutes after a safety injection and Phase B containment isolation with the 161-kV grid at 159 kV. TVA stated that although the voltage at the 6.9-kV Class 1E boards dipped to 6761 volts when the 6000-horsepower reactor coolant pump was started, it recovered to 6902 volts after approximately 14 seconds. TVA stated that this voltage transient would not actuate the 6.9-kV Class 1E shutdown boards degraded voltage relays, and that adequate voltage would be available for Class 1E loads.

The staff has reviewed this assessment and concludes that the Sequoyah auxiliary power system is capable of successfully starting a reactor coolant pump following an accident under minimum grid voltage without adversely affecting Class 1E loads.

2.3.4.3 Conclusions

The staff finds that:

- The new configuration has not affected the overall voltage profile of the 6.9-kV boards.
- The change of 100 valve motors represents no overall load increase.
- The replacement of the VNEW computer program with the RADIAL program is acceptable.
- Although Position 4 of FSAR Question 8.33 (which applies to Sequoyah) contained no specific accuracy requirement and the measurement accuracy of Sequoyah was 5 percent, the consistency of the results between the analyses and test values (within 2 percent shows that the model consistently predicts steady-state system performance.
- Although no test and analyses were performed down to 120/208-volt level (where the ability of the Class 1E control circuit to pick up the control devices such as the starter, relay, and solenoid is determined), TVA has demonstrated that adequate voltage is available to components supplied by the 120-volt ac control processing. Therefore, no additional tests are necessary.

In regard to the transient aspect of the test, the staff finds that:

- The TVA justification regarding the 120-volt ac control power system design features and calculations is acceptable, and no additional PSB-1-related steady-state and transient tests for the 120/208-volt level are necessary.
- Review of the Brush recorder voltage and current measurements taken at the terminals of the ERCW and ABGS motors and the supporting information provided by TVA showed that (a) the differences between the calculated transient voltages from the new computer program and the measured transient voltages are within the PSB-1 guideline; (b) the one-cycle voltage dip is an accurate measure of the actual minimum transient voltage; and (c) the difference in recorded currents (between Phases A and C) is the result of a recorder calibration problem and is not indicative of a current-unbalance problem.
- In comparison with the exponential voltage recovery model normally used in calculating the effects of starting large motors, TVA's transient analysis model, which assumes the voltage dip at its lowest point for the entire accelerating time of the motor, is conservative.
- The replacement of the VNEW computer program with the RADIAL program is acceptable.
- TVA's worst-case calculation for voltages on Class 1E buses shows that the auxiliary power system is capable of successfully starting a reactor coolant pump following an accident.

On the basis of its review of the steady-state aspect and transient calculations provided by TVA, the staff concludes that TVA's new computer program can adequately predict the transient and the steady-state responses of the Sequoyah auxiliary power system. Thus, a new verification test for the auxiliary powered system voltage study is not required.

TVA originally asserted that the voltage dips associated with a degraded grid condition bounded the dip associated with operation of the vital buses supplied from the EDG. Based on its review of TVA's calculation and test data associated with DG operation, the staff concluded that steady-state operation of the DGs was in fact bounded by the above results. However, the staff concluded that, during the automatically sequenced loading of the EDGs, voltage transients could occur which are more severe than anticipated in the PSB-1 transient analysis. Therefore the staff required TVA to do separate calculations to analyze operability of safety related electrical equipment during DG loading. These calculations are evaluated in 2.3.3.2, above, and the staff concluded that margins were adequate.

2.4 Alternately Analyzed Piping and Supports

2.4.1 Introduction

SNPP Section III.5 describes a TVA program to verify the adequacy of piping and pipe supports that had been installed and qualified by alternate analysis (AA) criteria. TVA's AA criteria use general criteria and guidelines to locate supports in lieu of rigorous piping analysis. The AA criteria were generally used for nuclear safety class piping systems that are 4 inches in diameter and smaller, with some exceptions as discussed in the SNPP. Nuclear safety class piping is defined in Section 3.2 of the Sequoyah FSAR. AA criteria also were used for the design of some piping that is not nuclear safety class, such as piping Category I(L) systems, which are designed for seismic loads to prevent unacceptable interactions with safety class structures and components. The 2-inch and smaller AA piping was generally qualified and supported by the field organization using a series of typical support drawings. The larger AA piping sizes had uniquely engineered pipe support designs.

TVA initiated the AA program to address several deficiencies identified with the AA piping designs and the AA design documentation. As a result of these deficiencies, TVA issued nonconformance reports and significant condition reports related to the implementation of the AA criteria. In addition, the TVA Employee Concerns Program had raised a concern with TVA's resolution of all AA discrepancies in the nonconformance reports. The Employee Concerns on AA piping will be addressed in a separate staff evaluation.

TVA contracted with Earthquake Engineering Inc. (EQE) to evaluate Category I(L) AA piping systems. EQE conducted walkdowns of Category I(L) piping systems and reviewed a sample of the interfaces between Category I(L) piping and deadweight supported piping. EQE compared the Sequoyah piping configurations with the EQE earthquake data base; piping and supports not covered by their data base were evaluated.

TVA is conducting a two-phase program to resolve the concerns on the Category I (safety class) AA piping systems. Each phase of the program is discussed in the following.

2.4.2 Evaluation

Phase I Scope

TVA provided a description of the Phase I program activities in Section III.5.2.1 of the SNPP. The restart program implementation was controlled by nine program procedures, SQN-AA-001 through SQN-AA-009. The staff audited the Phase I program during the week of October 6, 1986. The audit team consisted of staff members and consultants from Brookhaven National Laboratory. The audit focused on the restart program scope, interim acceptance criteria, and program implementation.

The scope of the Phase I program includes those systems required to mitigate events addressed in FSAR Chapter 15 and safely shut down the plant. These systems include the majority of the safety-related systems in the plant. This scope is consistent with the scope of Phase I of the Design Baseline Verification Program. The Phase I review effort involved screening of AA piping systems for specific deficiencies that had been identified in TVA's AA program as discussed earlier.

The Phase I scope included the areas of concern listed below:

° consideration of the effects of anchor movements at the interface of large, rigorously analyzed piping systems - The effects of large, rigorously analyzed piping system deflections at the attachment point to AA piping systems had not been adequately evaluated in all areas. These deflections could result in excessive stress in the AA piping and excessive loads on the supports.

° consideration of the torsional effects of large, motor-operated and pneumatically operated valves in small diameter piping - The torsional loads that would result during a seismic response of the valve operators, had not been adequately evaluated in all cases. These torsional loads could result in excessive stresses in the piping and excessive loads on the supports. In addition, large displacements of the valves could result in damage to the valves and their attachments, or damage to adjacent equipment.

° consideration of the effects of non-seismically designed (deadweight supported) piping on seismically designed AA piping systems at the interface boundary - The effects of the deadweight supported piping on the seismic supported AA piping systems had not been adequately evaluated in all cases. Large seismic deflections in the deadweight supported piping could result in excessive pipe stresses or loads in the seismically analyzed AA piping systems. The restart program evaluated pipe sizes greater than a 2-inch nominal diameter. This issue is a greater concern for larger diameter piping systems because of the larger piping loads that could be generated.

° consideration of thermal flexibility analyses for piping systems with operating temperatures greater than 200°F - Thermal expansion flexibility analyses may not have been adequately performed in all cases. Excessive thermal expansion stresses in the piping system could result in fatigue or strain ratchet type failures in piping after repeated heatup and cooldown cycles. This issue is a greater concern for high temperature piping systems where thermal expansion deflections that must be accommodated by piping flexibility are greater.

The staff evaluation of restart program implementation was based on an audit of the Unit 2 program. During the audit, the staff and its consultants reviewed the program procedures and sample calculations, and conducted a field inspection of sample piping/support system runs. Piping documentation packages were reviewed to identify Phase I areas of concern. Identified areas were then screened against simple criteria. For example, if anchor movements did not exceed 1/16 inch at branch connections, no further analysis was required. If the screening criteria were not met, the analyst performed simplified hand calculations or computer analysis to qualify the piping. Pipe support loads were then compared against design loads. If support loads exceeded design loads, a detailed pipe support evaluation would be performed. Piping/support systems that did not qualify were modified. TVA's proposed support criteria were used to design the modifications.

During the audit, the staff and its consultants reviewed a number of piping and pipe support design packages. The packages covered piping systems in different buildings with different potential short-term safety concerns. The package review covered all levels of analysis from simple screening to detailed computer analysis. In addition, a field inspection was conducted for two sample piping systems in the reactor building and two sample systems in the auxiliary building.

On the basis of this audit, the staff concluded that TVA had adequately defined and was adequately implementing a program to ensure that short-term safety concerns would be identified, evaluated, and resolved before plant restart. However, two items were not fully resolved during the audit:

° TVA was unable to provide the basis for the deflection criteria that ensure that pipe supports are rigid. In a letter dated January 28, 1987, TVA stated it will perform an evaluation during the long-term program to justify the adequacy of the criteria. This was acceptable to the staff.

° The staff field inspection identified loose washers in unistrut clamp supports. TVA provided information on a current bolt-tightening program that will correct the problem. This issue was addressed in a separate staff evaluation dated March 11, 1988(b) on unistrut support design.

Following a July 18, 1986 meeting with the NRC, TVA, in a letter dated August 18, 1986, defined a set of interim acceptance criteria for evaluating piping and pipe supports in the restart program. The criteria were developed so that the restart program could be performed in a timely manner, with minimum support modifications. The criteria are not in accordance with FSAR commitments or with current code requirements; they are, however, intended to provide increased confidence that the piping/support systems, required for Chapter 15 accident mitigation and safe shutdown are adequate for short-term operation. TVA provided additional information and subsequently eliminated some of the originally proposed interim criteria in submittals dated September 4 and November 10, 1986, and August 17, 1987. TVA stated that piping and supports that meet the interim criteria, but not the long-term criteria, will not be modified before restart but will be re-evaluated and, if needed, modified during the long-term program.

TVA originally defined the proposed interim criteria in terms of exceptions to FSAR commitments. These exceptions and the staff's evaluations of them are listed below:

- Piping Criteria Exception: Secondary stresses resulting from seismic anchor movements (SAM) and thermal plus thermal anchor movements (TAM) will be evaluated for piping systems greater than 200°F. For piping systems 200°F or less, secondary stresses resulting from SAM plus TAM will be evaluated.

  Evaluation: Consistent with the Phase I scope, thermal expansion stresses were generated for piping systems with maximum temperatures exceeding 200°F. For piping systems 200°F or less, thermal expansion stresses were not calculated. The small thermal deflections for piping systems 200°F or less are a concern when a large number of thermal stress cycles are anticipated. The staff concludes that the exception does not represent a significant risk to plant safety based on the limited number of thermal cycles anticipated for interim operation; therefore, this is acceptable.

- Pipe Support Criteria Exceptions:

  Exception 1: Only safe-shutdown earthquake (SSE) seismic loads will be evaluated; operating-basis earthquake (OBE) loads will not.

  Evaluation: The staff concludes that this exception is acceptable for interim operation because OBE loads are, by definition, smaller than SSE loads. Therefore, a demonstration that the plant can be safely shut down for an SSE ensures that it can be safely shut down for an OBE.

  Exception 2: The effects of friction loads resulting from thermal growth need not be considered in the re-evaluation of existing supports.

  Evaluation: The staff concludes that this exception is acceptable for interim operation because friction loads are not expected to be significant. TVA had performed a study for the Watts Bar plant pipe supports that demonstrated that friction loads do not generally govern the design of supports. In a letter dated January 28, 1987, TVA committed to perform a similar study for Sequoyah as part of the long-term program.

  Exception 3: The allowable loads for expansion anchor bolts will be based on a minimum safety factor of 2.5 for wedge bolts and 2.8 for self-drilling anchors.

  Evaluation: These allowables are consistent with the plant's original design basis. In the long-term program, TVA will ensure that IE Bulletin 79-02 safety factors (that is, 4 and 5 for wedge bolts and self-drilling anchors, respectively) are met. This is acceptable to the staff.

In addition to the proposed interim acceptance criteria, TVA has also proposed criteria for support evaluations taken from Section 3.8.4 of the current NRC Standard Review Plan and from Subsection NF of Section III of the ASME Code. These criteria are not in accordance with the Sequoyah FSAR; nonetheless, the use of these criteria on an interim basis is acceptable to the staff.

However, the long-term program should use the criteria that meet the commitments in the FSAR.

Phase II Scope

TVA discussed the scope and activities of the Phase II effort in Section 5.2.2 of the SNPP. Phase II will evaluate the remaining Category I AA safety class piping systems not required for restart for the areas of concern identified in the Phase I program. Phase II also will address instrument lines and their supports. The acceptance criteria for Phase II will be TVA's established design criteria for piping and supports. TVA presented the scope and the schedule for Phase II in a letter dated April 8, 1987(a). In addition to the deficiencies evaluated in the Phase I program, TVA also will address the areas of concern listed below in the Phase II program.

- consideration of thermal flexibility analyses for piping systems with operating temperatures between 120°F and 200°F

- consideration of the interface between AA piping and deadweight supported piping for pipe sizes less than or equal to 2 inches in nominal diameter

- consideration of the effects of long piping runs and large concentrated weights

The bases for resolving the additional deficiencies in the Phase II scope are discussed below.

The deflections resulting from thermal expansion are relatively small and would not produce gross distortion or failure of piping systems with operating temperatures less than 200°F. Although the thermal deflections for these systems would not be large, it is possible some of these systems could exceed Code allowable stress limits. If the Code allowable stresses were exceeded, the main concern would be the potential for developing fatigue cracks after a number of thermal stress cycles. The staff agrees with TVA's conclusion that for low temperature systems, the small possibility of such fatigue cracking does not represent a significant risk to plant safety for short-term operation.

The staff concludes that evaluation of the interface between AA piping and deadweight-supported piping for pipe sizes less than or equal to 2 inches in diameter need not be considered in the restart program. The weight of small diameter piping is relatively small; consequently, any seismic loadings on this piping would be relatively small. Seismically designed valves and equipment and supports at the interface of seismic and deadweight-supported piping are normally relatively stronger for small piping than for larger piping. It is, therefore, unlikely that movement of the deadweight-supported piping would result in their propagation of a pipe break into the seismic piping.

The staff concludes that evaluation of potentially inadequate supports for long piping runs (in the axial direction) and large concentrated weights need not be considered in the restart program. TVA addressed the most significant concentrated weights, and motor-operated and pneumatically operated valves in the restart program. Frictional effects from vertical and lateral supports would reduce any theoretically calculated responses for long runs of piping. Therefore, the staff agrees with TVA's evaluation that potential deficiencies with long piping runs and other concentrated weights do not represent a short-term safety concern.

2.4.3 Conclusions

The staff concludes that TVA has defined an adequate program for resolution of short-term safety concerns required for plant restart. On the basis of its audit of sample design packages and a field inspection of sample Unit 2 piping systems, the staff found that the program was adequately implemented. The staff concludes that completion of the Phase I program for Units 1 and 2 will provide confidence that sufficient safety margins exist--in the design of AA piping/support systems required to mitigate FSAR Chapter 15 events and safely shut down the plant--to allow the plant to restart.

2.5 Cable Tray Supports

TVA's original design criteria for cable tray supports were developed between 1972 and 1974. Although these design criteria included the effects of earthquakes, they did not consider the effects of design-basis accidents (DBA). In 1975, TVA revised the original design criteria to include the DBA loads, but the original designs were never reviewed to ensure that they complied with the revised criteria. This deficiency affected only the cable tray supports attached to the steel containment vessel (SCV); however, other deficiencies found in 1984 and 1986 dictated a thorough review of the adequacy of all the cable tray supports. During that review, TVA discovered that the existing cable tray supports could not satisfy the basic commitments made in the FSAR. At a meeting on July 17 and 18, 1986, TVA proposed a set of interim acceptance criteria for cable tray supports that were less stringent than those in the FSAR. As a part of its request, TVA also committed to restore the original FSAR criteria for the affected cable tray supports in an orderly manner after restart.

The staff evaluation consisted of (1) ensuring that the proposed interim acceptance criteria were justifiable from the standpoint of safe operation of the plant and (2) confirming that the design calculations for cable tray supports were, as a minimum, in conformance with the interim criteria. The staff and its consultants (Brookhaven National Laboratory) visited the plant twice and met with TVA once July 21 through 24, 1986, and a more extensive audit during September 29 through October 3, 1986. Specific requests for additional information were developed as a result of these meetings.

TVA responded to the questions resulting from the July 21 through 24, 1986 meetings in a letter dated August 18, 1986. This report discusses the justification for the interim acceptance criteria and how the criteria were to be implemented.

During the audit of September 29 to October 3, 1986, the staff (1) evaluated the cable tray support walkdowns performed by TVA by physical inspection of the plant, (2) reviewed the calculations performed by TVA to evaluate the adequacy of cable tray support systems with respect to the interim acceptance criteria, (3) reviewed additional data supporting the interim acceptance criteria, and (4) evaluated a portion of the concrete strength test data.

2.5.1 Interim Acceptance Criteria

2.5.1.1 Evaluation

(1) Damping

TVA proposed to use 7 percent of critical damping for the cable tray for the safe-shutdown earthquake and design-basis accident (SSE/DBA) loading, as compared with the 5 percent allowed in the FSAR. To support these criteria, TVA contends that:

° Substantial cable tray test data demonstrate that the damping for cable tray supports is considerably larger than 7 percent. The cable trays at Sequoyah have the natural frequencies and general characteristics of those tested.

° Another plant was allowed to use 15 percent damping for its cable trays, which are very similar to those at Sequoyah.

° NRC RG 1.61 allows 7 percent damping for bolted structures. While some of the cable tray supports are welded, most of the mass is on the trays, which are bolted to the supports.

A considerable amount of data indicates that damping in cable tray systems is greater than 5 percent for SSE-type loadings. This occurs because of the considerable damping in the cables themselves and in the cable connection to the tray. During the walkdowns performed in the week of September 29, 1986, the staff verified that the Sequoyah cable trays and cable tray supports are generally similar to those tested and found acceptable in other nuclear power plants. The staff believes that those cable tray tests (which indicate damping values in the range of 10-20 percent) are applicable to Sequoyah. In addition, TVA has performed calculations to determine the effect of this increase in damping. The typical stress ratios (defined as actual stress/normal stress allowable) are given below for cable tray supports in the auxiliary building.

		Stress ratio	
Support	Member	7% damping	5% damping
Section-P	Main member	1.397	1.397
	Bracket	0.532	0.554
	Joint	0.516	0.521
	Anchorage	1.49	1.51
1G	Main member	1.038	1.045
	Bracket	0.863	0.875
	Joint	1.154	1.277
	Anchorage	1.403	1.55
5	Main member	1.04	1.005
	Bracket	0.555	0.558
	Joint	0.55	0.584
	Anchorage	1.13	1.17

These stress ratios are less than the allowable ratio for the SSE loading condition, which is 1.6. These figures indicate that the change in damping from 5 percent to 7 percent has little effect on the stress ratios. Thus, for restart purposes, the 7 percent damping proposed by TVA for DBA/SSE loading is acceptable to the staff.

(2) DBA/SSE Load Combination

In the FSAR, TVA committed to use the absolute sum combination of SSE and DBA loading effects. TVA now proposes to use the square root of the sum of the squares (SRSS) combination for the interim acceptance criteria. TVA contends that the SSE and DBA loads are both low probability events and are unlikely to occur together; therefore, use of the SSRS combination of their load effects is appropriate.

TVA's proposed approach is reasonable because of the uncoupled nature of the SSE and DBA loadings. Both loads are dynamic, and the absolute sum of their effects would only occur if the SSE and DBA events occurred at the same time and the peak response of the tray supports to both the SSE and the DBA events coincided. The probability of such a coincidence is rather low. Thus, the staff finds the SRSS method a reasonable load combination approach for plant restart and it is acceptable.

(3) Elimination of 1/2 SSE Load Case

In the FSAR, TVA commits to considering the SSE and 1/2 SSE loads. TVA now proposes to use the SSE loading only for the interim acceptance criteria. TVA argues that the SSE case is usually more severe and that the safe shutdown of the plant is ensured if the SSE criterion is met.

The 1/2 SSE load is, by definition, less than the SSE load (ignoring the effect of the damping ratio). Generally, when the computed SSE stresses are compared with the SSE stress allowables, the computed stresses tend to be more critical than they are in corresponding stress comparison for the 1/2 SSE case. However, several of the proposed interim acceptance criteria relax the allowable stresses for the SSE loading case. This could, in some instances, make the 1/2 SSE loading case more critical than that of the SSE from the standpoint of design. However, a demonstration that the plant can be safely shut down for the SSE automatically shows that it could be safely shut down for the 1/2 SSE. Additionally, the plant Technical Specifications require plant shutdown after a seismic event that equals or exceeds the 1/2 SSE acceleration levels. The proposed elimination of 1/2 SSE case is acceptable to the staff on an interim basis.

(4) Allowable Stresses

In the FSAR, TVA makes a commitment that the cable tray support stresses be less than 0.9 times the yield strength for SSE/DBA loading. TVA now proposes to change this requirement to 1.7 times the American Institute of Steel Construction (AISC) allowables for SSE plus DBA loading, and 1.6 times the AISC allowables for the SSE alone. The justifications provided by TVA note that these allowables are stated in the NRC Standard Review Plan and have been used in the review and approval of many plants.

Considering the high ductility of the steel used in nuclear power plant structures (steel must meet the American Society of Test Methods (ASTM) Standards for A3 , A441, A527, or A572 steels), the Standard Review Plan allows up to 1.7 times the AISC allowable stresses under such low probability loadings as the SSE and DBA. During the audit at Sequoyah, the staff verified that the actual AISC allowable stresses were reduced if the structural member section was not compact and that the 1.6 or 1.7 factor was applied to these reduced AISC allowable stresses. Therefore, the criterion of using up to 1.7 times the working stress allowable for cable tray support calculations is acceptable.

2.5.1.2 Implementation of Interim Criteria

(1) Cable Tray Supports Attached to Steel Containment Vessel

The re-evaluation of supports attached to the steel containment vessel was required to resolve Nonconformance Report (NCR) SQNCEB 8414. The NCR addressed the fact that the cable tray supports on the steel containment vessel were not designed for DBA loadings.

A total of 560 cable tray supports are attached to the steel containment vessel. All supports are attached to the outside of the vessel by welding to the horizontal or vertical vessel stiffeners. Support members are generally 4-inch by 4-inch or 2-inch by 2-inch tubular steel members. Cable trays are generally attached to the supports by clip angles that are welded to the support member and bolted to the cable tray. Most supports are simple 2-inch by 2-inch cantilever brackets welded to vertical stiffeners. The next largest category of supports are 2-inch by 2-inch cantilever brackets welded to a 4-inch by 4-inch member spanning between vertical stiffeners. Most supports were analyzed by grouping all similar configuration and selecting the worst-case envelope of the supports within each group. The majority of the supports (551) were enveloped by five typical designs. The remaining nine unique supports were individually analyzed. A walkdown of the cable tray systems was performed to establish actual tray loading. Measurements of the cross section of cable trays were taken, and actual tray loadings were calculated from the profile measurements.

The GTSTRUDL computer code was used to analyze the supports. The cable tray and its supports were modeled using elastic beam elements. A typical model included two supports and one cable tray span. The flexibility of the model support points was modeled using spring constants determined by a finite element analysis of the containment vessel and stiffeners. Cable mass and tributary mass of the adjacent spans were included as lumped masses. Response spectrum analysis was used to analyze the SSE and DBA events. The events were analyzed separately using 10 percent peak frequency broadened, as required in the FSAR, and 7 percent damped spectra. Modal response combination was performed by the SRSS method. The directional response combination for the DBA event was implemented by absolute summation of the three directional responses. For the SSE, the directional response combination was performed by taking the absolute sum of the highest horizontal response and the vertical response. The DBA response was combined with the SSE response by the SRSS method. Finally, the response resulting from dead weight was combined absolutely with the combined response of the SSE and DBA. Resulting stresses were evaluated against the criterion of 1.7 times the AISC allowables.

The effects of containment vessel expansion resulting from DBA thermal and pressure loading on the cable tray supports were also evaluated using the thermal loading capabilities of GTSTRUDL. The containment expansion effects resulting from pressure were converted to an equivalent temperature gradient and then added to the actual thermal gradient. The total temperature gradient effects were applied to the cable trays supports to determine their stresses.

The largest reaction load from the cable tray support analysis was applied to a containment vessel model to determine stresses in the vessel wall and stiffeners. Maximum stresses were evaluated against the applicable ASME Code allowables.

Supports that failed to meet the interim acceptance criteria were analyzed using the actual tray loading determined by the field walkdown. If the criteria were met with the reduced weight, the load rating of the tray was reduced and controls were established to prevent additional weight beyond the reduced capacity.

TVA has completed the calculations for all the supports attached to the containment. The results indicated a need to modify 3 existing supports and to add 12 new supports. All modified and new supports were designed to meet original design criteria requirements. Two of the modifications were required to prevent overstressing the supports, and one modification was required to prevent overstressing the containment stiffeners. Twelve additional supports were required in areas where span length exceeded the allowables.

The staff and consultants reviewed sample qualification calculations and performed a walkdown of the affected supports. The staff audit team also reviewed selected calculations covering the DBA response spectra generation, thermal and pressure-induced displacements, stiffness of the steel containment vessel (SCV) stiffeners at support attachment points, and effects of support loads on the SCV wall and stiffeners. Based on the audit results, the staff concluded that methods used in re-evaluating the SCV cable tray supports were adequate and that the interim acceptance criteria were appropriately implemented to qualify the supports for the plant restart.

(2) Cable Tray supports on the Reactor Building Shield Wall

Many cable trays located in the annulus between the SCV and the shield wall are supported from the shield wall. In these cases, the base plate of the cable tray support is bolted directly to the shield wall using wedge-type expansion bolts. These supports consist of either cantilevered tube steel configurations or tube steel members mounted parallel and bolted directly, with little clearance to the shield wall. Because the total annulus clearance is only 5 feet, the maximum span length of the main member in the cantilevered configurations is less than 5 feet. TVA determined that because the surface mounted tube supports were mounted adjacent to the concrete their response amplifications to seismic inputs would be negligible. Therefore, these surface-mounted supports are qualified for the seismic response of the reactor shield building at their points of attachment. On the other hand, all cantilevered supports were qualified either by individual analysis or by comparison to cable tray and supports enveloping configurations for which analyses were performed.

Although there are approximately 400 supports attached to the shield wall, they are segregated into three generic and a number of special support configurations representing the cantilevered and the surface-mounted types. For the three generic configurations, TVA selected a bounding or enveloping case to evaluate their acceptability based on considerations of support location, loading and member span. Supports identified as MK 9e, MK 11c, and MK 18b were the bounding cases because each was installed at a high elevation, carried maximum loads (four trays), and exhibited maximum member spans. The special configuration supports were each evaluated, because they exhibited unique configurations. The staff found the TVA selection and categorization of the supports acceptable.

TVA performed a walkdown of all shield wall-mounted supports. In the walkdown for the generic and special supports, the configurations were confirmed; the dimensions of the base plate including any eccentricities of the tube attachments and bolt holes and the proximity to other bolted structures were noted; the span lengths and full profiles were recorded; and the presence of thermal insulation and multiple attachments were noted. For all other supports, a visual check of all these attributes was made and any deviation was measured, if appropriate, and recorded. The as-built information obtained in the walkdown was used in the evaluations. Furthermore, all instances of tray overfill, base plate bolt hole oversize or attachment eccentricities and bolt hole shear cone interference were evaluated.

The staff performed a walkdown in the annulus area. Tube attachment eccentricities and ground wire attachments were observed for supports Mk 9b and Mk 15, respectively, but no real deficiencies were noted. The supports and trays appeared adequately constructed and firmly anchored.

An audit of the calculations for the shield wall-mounted supports was conducted. The calculations were retained in a single file identified by calculation no. CSG-86-009. In the file were copies of all the analyses performed for these supports from April 1986 to the present. These included the latest GTSTRUDL and BASEPLT II computer analyses for each generic support and selected special supports, the numerical development of bounding load cases, the assessment of all anchor bolt shear cone interferences, and the evaluations performed to bound the conditions of base plate eccentricity noted in the walkdowns. In general, the calculations were complete and understandable. However, in those instances where revisions were made to earlier calculations, the earlier calculations were not labeled "superseded," making the audit difficult. The audited calculations have demonstrated that each cable tray support attached to the shield wall had sufficient capacity to meet the interim criteria for the SSE load condition.

(3) <u>All Other Cable Tray Supports</u>

There are 2900 cable tray supports in Category I structures (excluding the steel containment building and the reactor building shield wall). Most of these are in the auxiliary building (1700) and the control building (850).

The staff reviewed the selection of the worst-case supports in the auxiliary building, documented in TVA calculation RIMS B25 860913 825. The selection process started with a review of the drawings that contained support details. After considering factors including the number of cable trays for each support, span length, and floor elevation, 10 worst-case support configurations were identified. Each configuration may represent a group of specific supports with different geometries or it may represent a unique situation. For those configurations that represent a group of supports, the following three criteria were used to select the specific worst cases: (1) supports having the largest span lengths and largest weights, (2) maximum weight with the length selected for the first mode period at peak response of the spectrum, and (3) maximum length with weight selected for first mode period at peak response of the spectrum. The TVA central technical group reviewed these cases and added five more cases.

The same selection process was applied to cable tray supports in the other buildings. Thus, altogether, TVA considered 30 original worst-case supports and 5 additional ones.

The staff finds that TVA has used good engineering judgment in its selection of the worst cases and finds the approach used acceptable for restart.

TVA performed walkdowns for each of the worst-case supports to evaluate the following:

- the weight in the trays (Profiles were measured for trays that were more than 75 percent full and weights calculated.)
- any additional attachment to the support (Sketches were made detailing the attachment.)
- the cases where the tray support is not mounted concentrically on the base plate
- whether the support is fire protected
- any violation of TVA's Construction Specification G-32 (e.g., close spacing of adjacent anchorages resulting in overlapping of shear cones or anchor plates placed near an edge of a concrete member)
- any other unusual details

Reports on the results of the walkdown were prepared and signed by the preparer, checkers, and a quality control staff member. The staff reviewed the results during the September 29 to October 3, 1986 audit and found them accurate with one omission. An interference was noted for support Mark 31: a 6-inch conduit was close to a bracket of this support, and seismic-induced motion could be expected to cause the bracket to impact the conduit.

All accessible supports in the reactor building (inside containment) also were inspected. The inspection verified the TVA walkdown findings, which included cases of supports not installed concentrically on base plates and cracked concrete under base plates. These discrepancies are discussed in Section 2.5.1.6. No additional deviations were observed.

TVA prepared a GTSTRUDL model of each of the worst case supports based on the drawings and the results of the walkdown. The supports were modeled as beam elements. The mass of the cable trays was lumped on the appropriate brackets with the tray masses distributed equally to the adjacent supports. A response spectrum analysis was performed using the 7 percent damped spectrum. The model used for support marked "Section P-P" was reviewed during the staff audit and found acceptable.

TVA's responses to several issues raised during the July 21 through 24, 1986 meeting were evaluated by the staff during the September 29 through October 3, 1986 audit. These issues and their resolutions are addressed below:

- A few locations were identified where the span of the trays was more than 8 feet. These conditions occurred where the trays are inclined at a 45-degree angle. The horizontal projection of the span is less than 8 feet, but the inclined span is greater. TVA has performed load tests (TVA calculation RIMS B46 860311 003) to evaluate a cable tray in this configuration. The tray was found to have a capacity of 140 pounds per foot, which indicates a safety factor of more than 3 over the full tray design loading of 45 pounds per foot. This is acceptable.

- Several groups of cables cascade vertically from a conduit or from one tray to another in the control building. TVA has performed tests at Wyle Laboratory to demonstrate that the cascading cables can withstand SSE seismic-induced loading. The tests have been evaluated by an independent TVA consultant. The TVA consultant has concluded that the cables are not overstressed because they are not stressed beyond their tension capacities. TVA has provided the staff witn a copy of its evaluation of the Wyle Laboratory test results that confirms the fact that the cables are not overstressed. The staff reviewed this report and found it acceptable.

With resolution of the confirmatory items (Section 2.5.1.6), the staff concludes that the program conducted by TVA for qualification of these cable tray brackets and supports was adequate and acceptable for restart.

2.5.1.3 Anchoring in Concrete

This discussion applies to supports that are anchored in concrete by means of base plates, anchor bolts, and embedded plates.

Several concerns relating to safety factors and methods of analysis were identified at the July 21 through 24, 1986 meeting. These have been addressed by TVA and were discussed during the September 29 through October 3, 1986 audit. They are discussed below.

TVA proposed that self-drilling (SSD)- and wedge (WB)-type expansion bolts used for base plate anchorages be designed for a safety factor of 2.0 under the load combination of SSE plus DBA. The TVA staff indicated that this would be an interim criterion. In the Phase II design qualification work, the minimum safety factors for SSD and WB would be upgraded to 2.8 and 2.5, respectively. In defense of this proposal, the TVA staff indicated that during the implementation of IE Bulletin 79-02, the NRC staff had accepted a safety factor of 2.0 for both types of expansion bolts on an interim basis. The same logic can apply in case of the interim evaluation of the expansion anchor bolts at Sequoyah for restart.

After reviewing TVA's proposal, the staff concluded that TVA should use, as a minimum, the original FSAR design criterion requiring 2.5 for WB and 2.8 for SSD as safety factors for the interim period and for the long-term effort, TVA should determine the actual safety factors and evaluate them against the requirements of IE Bulletin 79-02.

Some of the conservative assumptions used in TVA's standard design practice tend to support a view that the actual safety factors against the pull-out of expansion bolts will, in general, be higher than those calculated. For example, TVA uses the expansion bolt capacities based on 3000 psi concrete, whereas the concrete strength data at 90 days indicate that the actual strength of the concrete could be much higher than 3000 psi. This could increase the expansion bolt capacities significantly. Another example of the conservatism is that in normal installation, TVA procedures require preload of bolts to twice the design load. A minimum of 25 percent of the bolts are tested for slippage at that preload level. Any slippage (as indicated by a drop in load of the load indicator) was regarded as a failure. This requirement is more stringent than the accepted industry practice of allowing some slippage. These conservative design and installation practices form the basis for the staff's acceptance of the safety factors noted above for restart purposes.

TVA, in its submittal of January 14, 1987, committed to the interim criteria proposed by the staff; therefore, this is acceptable.

2.5.1.4 Base Plate Analysis

As discussed above, TVA performed frame analyses to evaluate the distribution of forces throughout the cable tray supports. The cable tray mass is distributed evenly between adjacent supports. Overloaded trays were evaluated in walkdowns. Trays that were less than full were considered to be full with the exception that some of the supports located on the steel containment vessel were evaluated for actual tray loads. The SSE loading was used as an input, and two alternate types of analysis were performed. The first type of analyses performed were response spectrum analyses. If there were no modes with natural frequencies less than 33 cycles per second (cps), a seismic load equivalent to the tray and support mass times the zero period acceleration (ZPA) was applied to the support. The second type of analysis performed was static analysis with a load equal to the tray and support mass multiplied 1.5 times the peak spectral acceleration. The deviation between the center of cable tray's mass points and brace connection joints had not been considered by TVA for all supports at the time of the staff audit. The supports on the steel containment were evaluated for the effects of the eccentricity. TVA will consider this in calculations to be developed. The staff does not expect that this will lead to significant changes in response forces; however, this will be treated as a confirmatory open item. In other respects, the staff considers the current analyses used by TVA are acceptable.

The loads from the frame analysis are used to evaluate the adequacy of the support members and base plates. Standard engineering methods are used to evaluate stresses in members and are considered acceptable by the staff. The BASEPLATE II computer program is used to evaluate stresses in the base plate and bolts and bearing stresses in the concrete. BASEPLATE II is a preprocessor code that generates input data for an ANSYS computer code solution. This also is acceptable to the staff.

Plate finite elements are used to model the base plate and elastic springs are used to model the anchor bolts. The concrete is modeled with an elastic spring in series with a gap element so that the concrete acts in compression but not in tension. TVA has performed sensitivity studies to develop criteria for the finite element modeling of the base plate. The modeling and analysis of the base plate are acceptable.

2.5.1.5 Concrete

TVA provided its responses to the questions related to concrete quality raised by the staff. The resolution of this issue is discussed in Section 2.6 of this report.

2.5.1.6 Confirmatory Items

The staff identified the confirmatory items listed below during the audit of September 29 through October 3, 1986, to be resolved by TVA before restart.

(1) An unused bolt hole was observed in the main tube member of support MK 11d in the annulus. It should be verified that this support is adequate.

(2) The 1/8-inch fillet welds used throughout the supports to the shield wall do not satisfy American Welding Society (AWS) Standard D1.1-85 Section 10.5.31. The adequacy of these welds is to be investigated based on data to be obtained in a scheduled TVA test program.

(3) The spring constant for self-drilling bolts was used for BASEPLATE II analyses. Most of the bolts are wedge bolts. The BASEPLATE II analyses must be revised to reflect the proper bolt type.

(4) An error was found in one of the element components for support MK 11d in the annulus. The evaluation of this support should be revised.

(5) An interference between a conduit and support MK 31 in the auxiliary building was observed during the audit. TVA must evaluate the significance of this condition.

(6) The evaluation of all worst-case supports in the auxiliary building must be completed and documented.

(7) The interim acceptance criteria for anchor bolts should be based on safety factors of 2.5 and 2.8 for the wedge bolts and self-drilling bolts, respectively. TVA should fully document its implementation of these criteria.

(8) TVA is to develop and submit for staff acceptance calculations that demonstrate that the eccentricity of the cable tray mass will not adversely affect the qualification of supports (e.g., for supports not installed concentrically on base plates).

(9) TVA is to provide its final evaluation report addressing the design adequacy of cascading cables tested at the Wyle Laboratories for staff review.

(10) TVA will complete all required cable tray supports modifications, as determined by the TVA evaluations, against the staff-approved interim acceptance criteria, before restart.

From reviewing the information provided in TVA submittals dated January 14, and February 4, 1987, staff concludes that TVA has taken proper corrective action for the above ten confirmatory items and that this is acceptable for plant restart. TVA conducted a test for the wedge bolt anchor in the area of the cracked concrete in accordance with TVA Construction Specifications and found that no degradation of the base plate anchor was observed. Based on an engineering judgment, this is considered to be acceptable for restart. However, an audit of the above items, including the cracked concrete, will be conducted following restart of the plant.

2.5.1.7 Conclusion

On the basis of its review of the material provided by TVA, two audits of TVA design documents, and a plant walkdown, the staff concludes that the interim acceptance criteria proposed by TVA for Sequoyah restart as modified in accordance with this report are acceptable.

2.5.2 Diesel Generator Building Supports Analysis

2.5.2.1 Summary of Issue

An NRC inspection (see IR 50-327, 328/85-29) revealed that cable tray support systems for the two diesel generator building at Sequoyah had not been designed to appropriate quality standards. The design for cable tray support systems failed to consider the effects of rigid body motion from the response spectrum ZPA in the determination of seismic loads for the design analysis. In this case, the ZPA of the response spectrum is 0.37g for the operating basis earthquake (OBE) and is 0.74g for the SSE.

The staff reviewed five cable tray support design calculations in the diesel generator building and two cable tray support design calculations in the additional diesel generator building. The staff found that these calculations had been performed using a modal superposition dynamic computer analysis. The computer programs consider only the dynamic modal response in the frequency range of interest. No consideration was given to the effects of rigid body motion from the response spectrum ZPA. As a result, the accelerations generated from the dynamic analysis were generally small when compared to the response spectrum peak accelerations. The use of these small accelerations alone in the design of the rigid supports for the cable tray support system was not conservative and was not adequate in terms of satisfying regulatory requirements.

TVA mistakenly used the computer-generated dynamic analyses so that much smaller responses (e.g., accelerations and forces) could be used in the design of cable tray supports. The dynamic earthquake analyses for the diesel generator building and the additional diesel generator building show that the peak accelerations from the response spectra are significantly larger than values used by TVA for design.

Use of these larger accelerations in designing the cable tray supports would have resulted in much larger structural sizes in the support systems.

2.5.2.2 Evaluation

In a letter dated November 25, 1985 and in Section III.3 of the SNPP, TVA describes the corrective actions it has taken. These actions include a re-evaluation of the cable tray supports in the diesel generator building and the additional diesel generator building to include the effects of the ZPAs. Other calculations--such as those for conduit supports and duct supports--were reviewed, and TVA determined that the dynamic computer analysis was not used.

The dynamic analysis method has not been identified in any other building at Sequoyah, and TVA no longer uses this analysis method. The calculations of the specific designer also were reviewed for cable tray supports in the control building and the auxiliary building to ensure that these supports were adequately designed to serve their intended function.

TVA has issued a design input memorandum for the cable tray support design criterion SQN-DC-V-1.3.4. The memorandum provides more stringent management control and technical review of dynamic analysis in the design of cable tray supports. It requires that the modal superposition dynamic analysis shall be performed and checked only by certain qualified engineer, as designated by TVA's civil project engineer. Further, TVA's Civil Engineering Branch central staff has provided direction and training for the re-analysis effort and will do so for any future designs/evaluations.

2.5.2.3 Conclusion

TVA has evaluated all cable tray support calculations in the diesel generator building and the additional diesel generator building for a failure to take the effect of ZPA into account. In those instances where the originally calculated acceleration was less than the ZPA, the ZPA was applied in the re-analysis. Results of the re-analysis indicate that the existing cable tray supports are still able to serve their intended function during a seismic event. Therefore, on the basis of its inspection and its review of the information presented by TVA, the staff finds that no structural modifications are required.

2.5.3 Cable Tray Support Base Plate Installations

2.5.3.1 Summary of Issue

Sixteen base plates (eight per unit) for the cable tray supports in the auxiliary building were improperly installed in that every hole in the base plates was drilled per the engineering drawing with a diameter 3/8 inch larger than specified by TVA procedures.

The staff reviewed cable tray support design drawings for conformance to design analysis and TVA's commitments. The staff found that the base plates with oversize holes had been used in the installation. Design Drawing 48N1369, Revision 2, specified 1-3/16-inch-diameter holes in the base plates for 3/4-inch-diameter wedge bolts. In accordance with TVA procedure, the correct hole diameter in a base plate is 1/16 inch larger than the nominal bolt diameter. In the above case, the correct hole diameter in the base plate should have been 13/16 inch. The incorrect dimension on the design drawing resulted from a misinterpretation of the designer's sketch by the drafter. The error was not found in the checking and review process because the original design calculations were not compared to the final design drawing, nor was the error identified in the inspection by TVA's construction QC inspectors.

2.5.3.2 Evaluation

TVA corrected the error by making special washer plates to cover the oversize holes and provide the bearing surface for the bolts. TVA checked the auxiliary building and control building drawings done by the same drafter. TVA also checked a number of calculations that had checked by the same checker to ensure there was no recurrence of this problem.

2.5.3.3 Conclusion

TVA has completed all the necessary corrective actions regarding the above deficiencies. As a result, the modified connections are judged to be able to serve their intended function as required by the design. On the basis of the above information and its review of Section III.3 of the SNPP, the staff finds the issue of oversize holes in the base plate has been acceptably resolved.

2.6 Concrete Quality

The TVA evaluation of Employee Concern IN-85-995-002, related to the adequacy of the concrete quality at the Watts Bar Nuclear Plant site, prompted the NRC staff to request further evaluations of the in-place strength of the concrete at the Sequoyah site.

The NRC staff and its consultants visited TVA headquarters during the week of January 5, 1987, to audit the procedures and the data base on which the TVA evaluation was based and to review the TVA findings. The potential deficiencies investigated include: (1) violation of sampling frequency, (2) low strength concrete and its effects on the Category I structures, and (3) lack of procedural control for bedding mortar.

TVA has completed its evaluation and has documented the final findings in Enclosure 1 to its letter of February 6, 1987.

TVA has determined that more than 90 percent of the relevant 90-day strengths are available and that only 5 percent of the 28-day strengths were deficient. Therefore, less than one-half percent of the concrete is unaccounted for by this procedure (5 percent deficient results with 10 percent missing data). For the concrete mix with the design strength specified at 90 days, an equivalent strength was calculated for each time period. The equivalent strength is that strength level, calculated from the mean strength and standard deviation, which may be expected to be exceeded by 90 percent of all strength tests. The lowest equivalent strength so determined was used to analyze each affected structural member. All were found satisfactory.

During the audit, the staff and its consultant checked the transfer of data from original test reports to the computer printout on which the calculations were based. A few isolated errors were found, but in each case when the error was corrected, the conclusions based on the calculations were not changed.

Both the methodology and the data base confirmed the validity of the TVA evaluation approach and conclusions.

A spot check of the structural calculations indicated that they were based on the correct concrete strength values, as applicable. TVA has redone some calculations to evaluate for newly determined concrete equivalent strengths. There were no written standards with which bedding mortar was required to comply. However, its use was well documented and regular strength tests were made and reported. A large part of the mortar was used for lubricating pump lines. TVA analyzed walls containing bedding mortar by very conservative assumptions. The staff concluded that TVA utilized adequate controls and standards in their evaluation of the bedding mortar used at the Sequoyah site.

The staff requested TVA to examine all concrete sampling records for demonstrating compliance with sampling frequency requirements during the exit meeting following the staff audit. TVA provided additional information by letter dated April 8, 1987(b), to supplement that in Enclosure 1 of its February 6, 1987 letter. The staff reviewed this information and found it to be acceptable.

On the basis of its review, the staff concludes that all previous concerns related to adequacy of the structural criteria for concrete strength and frequency of sampling and controls and standards for the bedding mortar have been resolved for restart.

2.7 Miscellaneous Civil Engineering Issues

Based on several significant condition reports (SCRs), TVA has identified a need to address the seismic qualification of components in meeting code and regulatory requirements. This effort includes the review of components, piping, pipe supports, cable tray supports, conduit supports and heating/ventilating duct supports as well as structures. TVA has committed to resolve any identified problems by analysis, testing or design changes with the corrective actions being integrated into the restart schedule. The specific restart requirements are to be determined by TVA management review. These topics are addressed by separate TVA programs and are addressed specifically in Sections 2.3.2, 2.4, and 2.5 of this SER, as well as Part 2 (Employee Concerns).

Section 15 of Part III of the SNPP addresses miscellaneous civil engineering issues related to Sequoyah.

Another effort initiated by TVA in the civil engineering discipline involves the capability of embedded plates and concrete anchors for cable tray and pipe supports to meet the TVA commitments made regarding the code allowable conditions. This area of review also relates to an employee concern in the construction category (No. 11301). The employee concern report identified an issue regarding TVA's implementation of IE Bulletin 79-02 criteria for calculating base plate flexibility. TVA plans to resolve this issue by reviewing a sample of 60 base plates to verify that the design calculations meet the requirements of the applicable base plate design criteria. The DBVP is addressed in Part 2 and Section 2.2 of this SER. As a result of the DBVP, the issue has been found by TVA not to be a restart item. However, as part of the calculations review program, TVA has re-evaluated approximately 5600 pipe support calculations, which considered the effects of base plate flexibility.

An additional issue involved TVA's implementation of IE Bulletin 79-14. This issue was addressed by an employee concern report related to engineering (EN 21202). The employee concerns report found that TVA's IE Bulletin 79-14 program was adequate for Unit 2. However, TVA initiated a program to inspect 2500 pipe supports to verify the as-built or as-modified condition with the documented design for Unit 1. Discrepancies identified are to be evaluated against the design criteria and repairs or modifications made as necessary to bring the support into conformance with the as-designed condition. This effort is being performed under a TVA special maintenance instruction. The supports in the program that have been identified as being required for operation on safe shutdown have been inspected as a restart activity as part of the pipe support enhancement program. This review area is discussed in Sections 2.3.2 and 2.4.

On th basis of its review of the TVA plans to execute these special programs, the NRC staff finds that with proper implementation of the plans the special issues should be fully resolved.

2.8 Heat Code Traceability

2.8.1 Introduction

Section III.15.6 of the Sequoyah Nuclear Performance Plan (SNPP) describes a TVA commitment to investigate materials control concerns involving FSAR commitments, design requirements, and traceability relative to pressure boundary piping components in the Sequoyah safety-related piping systems. The multi-phased investigation is concerned with clearly determining the commitments made and compliance to those commitments relative to design, fabrication, installation and traceability of documentation.

The issue of heat code traceability has also been evaluated through the employee concern program (element report MC40703). In particular, the key issue that developed from this review was the use of TVA Class B small bore pipe and fittings in TVA Class A applications. The TVA resolution of this problem is discussed below.

2.8.2 Evaluation

TVA designated an Employee Concern Task Group (ECTG) on July 1, 1986 to investigate materials control concerns. The results of this investigation were documented in TVA Element Report No. MC-40703-SQN. This report identified more than 200 possible discrepancies between Sequoyah Units 1 and 2 on safety-related piping (99 at Unit 1 and 110 at Unit 2).

The following corrective actions have been implemented to correct the existing problems identified by the ECTG Report and to preclude their recurrence:

(1) PIRSQNNEB8638 will ensure the clear definition of the applicable code edition and addenda of ANSI B31.7 used in the fabrication, erection, installation, and use of Nuclear Class Piping components, in the upper-tier documents.
(Corrective Action Tracking Document (CATD) No. 40703-SQN-01-R2 and CATD No. 40703-SQN-03-R0).

(2) CAQR SQP870627 will ensure that all Nuclear Class I, II, and III (TVA Class A, B, and C/D) pressure-retaining piping components will be examined and their suitability for use verified and documented in accordance with the applicable requirements, or replaced. (CATD No. 40703-SQN-02-R0, CATD No. 40703-SQN-06-R0 and CATD No. 40703-SQN-07-R0.)

(3) CAR-86-064 will ensure that site procedures contain the necessary detailed instruction to provide for the receipt, storage, and installation of Nuclear Class Piping Components in compliance with the applicable code requirements. (CATD No. 40703-SQN-04-R0.)

(4) CAR-84-064 will ensure that inspectors will receive the required training to ensure that Nuclear Class Piping Component material identification verification is performed and documented, in accordance with the applicable code requirements, throughout their receipt, storage, and installation at SQN. (CATD No. 40703-SQN-05-R0.)

(5) SCRSQNMEB8614 R1 and ECN L6784 will ensure that TVA design drawings contain clear and consistent identification of where (location) and how (e.g., double automatic valve, specially bored fitting) the piping classification changes, as stated in the FSAR, are effected. (CATD No. 40703-SQN-08-R0.)

(6) PIRSQNMEB8793 will ensure that either the FSAR or the design drawing contain a clear definition of the boundary between the primary coolant loops and their branch lines. (CATD No. 40703-SQN-09-R0.)

TVA (Division of Nuclear Engineering) then assembled a new investigative unit, the Heat Code Traceability Task Group (HCTTG), to evaluate and resolve the issues raised by the ECTG. The results of this investigation were documented in TVA's report B25870225-036. This report (B25870225-036) reduced the 209 original discrepancies to a total of 7 items of noncompliance.

The investigations led to the issuance of three Corrective Action Reports (CARs)--SQ-CAR-86-052, SQ-CAR-86-055, and SQ-CAR-86-064--which document the proposed applicable corrective actions to the discrepancies and program deficiencies.

As a result of disagreements between members of the ECTG and the HCTTG regarding the proposed TVA corrective actions to resolve the employee concerns, independent experts were contracted to assess the issues. The report documenting the findings of consultants Kelly and Landers was issued as an attachment to the element report 40703, submitted to the NRC on May 13, 1987. This report partially stated:

> The current, as-analyzed stress values of TVA Class A small bore piping have been reviewed. The nodal points which exceeded 60 percent of either code allowable stress or actual allowable stress were tabulated. There were approximately 2600 nodal points used for the small bore piping analysis of TVA Class A piping. Two and

one-half percent of the nodal points had stress ratios which were not capable of meeting the 40-percent reduction on the code allowable stress. Similarly, 1.8 percent of the nodal points had stress ratios which were not capable of meeting the 40-percent reduction on the actual allowable stress.

The report also partially concluded:

> In summary, the material control problem is limited to small bore piping. This report demonstrates that there is no technical difference in Class A and Class B piping components. In conclusion, the engineering evaluations demonstrate that the installed small bore pipe and fittings comply with ANSI B31.7c Code requirements when the 40 percent allowable stress reduction factor is used in lieu of NDE. Thus, plant safety is assured.

This reduction in allowable stress refers to paragraph 1-724 in ANSI B31.7c-1971 which states in part:

> Unless otherwise required by the Design Specification, and provided all other applicable requirements of this division (1-274) are met, the non-destructive examination requirements of this division do not apply to:
>
> 1. Non-pressure-retaining material:
>
> 2. Seamless pipe and tube, seamless forged socket welding fittings, and seamless wrought butt welding fittings 2-inch nominal pipe size and smaller provided that:
>
>   a. The pipe, tube and fittings are made of P number 1 or P number 8 materials that meet all requirements of one or more of the standard materials specifications listed in Tables 1-724 and A-1.
>
>   b. The design stress intensity values (S_m) of Table A-1 used in the design analysis are multiplied by a factor of 0.60.

Note: The major difference between the small-bore pipe material requirements of Class A, B and C materials is the application of non-destructive testing to Class A materials.

The three previously mentioned Corrective Action Reports (SQ-CAR-86-052, 86-055, and 86-064) document the result and corrective actions associated with the various discrepancies noted in the three (ECTG, HCTTG, and consultants Kelly and Landers) reviews performed at Sequoyah.

TVA also performed additional reviews in this area in order to verify the accuracy of the employee concerns and to assess the possible effect on the safety of the Sequoyah plant. These reviews were performed by Bechtel Power Corporation, Structural Integrity Associates, and Aptech Engineering. The highlights of these reviews are summarized in the following.

Bechtel Power Corporation Audit

The purpose of this audit was:

> To verify, by examination of objective evidence, compliance with those aspects of the TVA Quality Assurance Program associated with materials. Audit to address program applied both during the construction phase and the operations phase.

This audit concluded that TVA had generally complied with the connected quality programs and applicable implementing procedures for material control for both construction and operations. The exceptions to this compliance were 5 Audit Findings (2 for construction, 3 for operations) and 6 Audit Observations (5 for construction, 1 for operations).

With regard to programmatic deficiencies, the Bechtel Power Corporation audit did state:

> The findings of this audit do not reveal a deficiency in programmatic controls. However, there were instances of implementation errors (i.e., incompletely recorded heat numbers, heat numbers recorded on items or documentation partially illegible, etc.) which can create traceability questions requiring laborious and costly research and investigation efforts.

Structural Integrity Associates (SIA) Evaluation

The three tasks assigned to SIA by TVA for its investigation were:

(1) Survey the available documentation and industry personnel involved in the construction of other light water reactors built during the same time frame as Sequoyah to determine the codes and standards invoked for design and construction of those plants and to present the methods used by other utilities for materials control and maintenance of traceability during plant construction.

(2) Obtain a knowledgeable, independent interpretation of the traceability requirements of the various construction codes along with an historical background of traceability and marking requirements.

(3) By survey of the available data bases, determine whether any component service failure has ever been attributed to improperly documented material or resulting from a traceability flaw.

This report summarized:

> ...that materials traceability, although not a code requirement, has been important to plant owners. Traceability of materials has generally been maintained to a high degree although not 100%.
>
> Even though a small fraction of material of questionable or incomplete pedigree is known to have been installed and placed in service, no failures attributable to such material have been reported. The methods used by TVA in the design, procurement, and

construction of piping systems for the Sequoyah units appear to have been typical of the day. The heat code traceability questions raised by the Nuclear Safety Review Staff report are not unique. Those questions relative to Sequoyah do not appear to present an unresolved issue.

Aptech Report

This report encompassed a review of nuclear material manufacturers programs, policies, and practices, as well as nondestructive examination versus nuclear classes. This report concluded:

> For absolute and unquestionable traceability, the procurement document, the heat code number, and the manufacturer must be known. Also, if any NDE was performed by someone other than the manufacturer, a separate document was generated showing the NDE method performed and the identity of the material.
>
> The rejection rate of NDE performed on small bore fittings manufactured by forging or machining was less than one percent.
>
> Even today, there are no markings put on small seamless piping products to indicate the class unless the purchasing document actually requires this to be done. All manufacturers that were contacted have marked the NDE performed on the material since 1980. Prior to that time, some did and some did not. We believe that NAVCO and the material manufacturers procedures and QA programs met the NAVCO requirements of both ANSI B31.7 and ASME III.

NRC Staff Review Summary

The NRC staff conducted a special team inspection at Sequoyah as discussed in Inspection Report 50-327, 328/87-44. The objective of the inspection was to determine the accuracy of the information contained in the element report and to determine the adequacy of TVA's conclusions and corrective actions. At the conclusion of the inspection effort the NRC staff concluded that TVA generally performed an extensive review of the heat code traceability issue. The information contained in the element report was found to accurately scope and review the identified issues. However, several inadequacies were identified during the NRC staff's review of supporting engineering calculations for small bore piping; these are listed below:

(1) TVA has not performed minimum wall calculations for pipe schedules other than schedule 160. TVA needs to perform those calculations to ascertain that a pressure problem is not present.

(2) The acceptance of 2-1/2 percent of nodal points for small-bore piping, based upon the use of actual material properties and thicknesses, is not acceptable. TVA needs to review those nodal points again and upgrade them, either by performing the additional nondestructible examination, or by adding more supports to reduce the loads, or by replacing the piping.

(3) TVA Design Criteria for Detailed Analysis of Category I Piping Systems, SQN-DC-V-13.3, Rev. 3 provides the loading conditions and stress limits for Category I piping systems in Table 3.1-1. Footnote 3 of this table states that the allowable stress levels are given in ANSI B31.1-1967. TVA's calculations of the allowable stresses for small-bore piping used ASME Section III, Appendix I allowables which do not meet the criteria in SQN-DC-V-13.3.

TVA responded to these items in letters dated December 4, 1987 and March 2, 1988; these responses have been found acceptable by the staff.

2.8.3 Conclusions

The NRC staff believes that TVA has properly characterized the problems with heat code traceability as a part of the SNPP and adequately addressed the employee concerns identified in TVA Employee Concern element report MC-40703, "Heat Code on Related to Material Control."

3 SPECIAL PROGRAMS

The Sequoyah Restart Task Force identified a number of technical issues of particular interest that are to be addressed before restart. These include major regulatory programs, such as environmental qualification of equipment and fire protection, as well as specific technical issues, such as adequacy of electrical cables. The resolution of these issues are discussed in the sections below. In some cases, there are related employee concerns; individual evaluations of the element reports are provided in Appendix A.

3.1 Fire Protection

3.1.1 Introduction

Following a staff inspection of July 16-20, 1984, at Watts Bar on compliance with Appendix R to 10 CFR 50, the staff issued a Confirmatory Action Letter to TVA on August 10, 1984. This letter identified the actions to be taken by TVA to implement a complete review of the Appendix R program at Sequoyah. On December 18 and 21, 1984, TVA submitted the results of the Sequoyah Appendix R re-evaluation, which were needed to complete the actions required by the letter of August 10, 1984.

Based on TVA's submittal of December 21, 1984, Sequoyah Units 1 and 2 were not in compliance with 10 CFR 50, Appendix R, Sections III.G, III.J, III.O, and III.L. TVA failed to meet Section 2.C.(13).a of the Sequoyah Unit 2 operating license, which requires TVA to maintain in effect and fully implement the fire protection requirements of 10 CFR 50, Appendix R, Sections III.G, III.J, III.L, and III.O.

The staff conducted a special Appendix R inspection January 14-18, 1985, to verify that TVA had completed the items required by the letter of August 10, 1984. This inspection evaluated structures, systems, and components important to safe shutdown to determine if the existing and/or proposed plant fire protection features would provide a level of protection equivalent to the requirements of 10 CFR 50, Appendix R, Sections III.G and III.L. In addition, after the staff received TVA's submittal of December 21, 1984, the scope of this inspection included the NRC staff's determination as to whether the proposed fire protection features were capable of limiting potential fire damage so that one train of systems essential to achieving and maintaining hot standby from either the control room or emergency control stations would be free of fire damage.

As a part of its re-evaluation effort, TVA developed operating procedures that addressed the required manual operation of valves for cold shutdown and casualty procedures that addressed the repairs associated with the residual heat removal (RHR) pumps, RHR room coolers, and various cold shutdown valves. In addition, to demonstrate that one train of systems necessary for hot standby is free from fire damage, TVA developed a fire shutdown logic (SDL) that defined the safety functions and sets of equipment required to achieve safe shutdown conditions under postulated fire conditions. The SDL is supplemented by key diagrams that identify the redundant paths/equipment required to achieve hot standby and subsequent cold shutdown.

From the SDL and the associated keys, TVA identified cables in block diagram form for required components/equipment. These cables were then color traced and plotted on physical cable separation drawings. From these color-coded drawings, TVA evaluated and identified specific cable interactions. TVA performed a field verification of actual equipment locations, where necessary, to ensure that separation was adequate. Specific cable interaction identification sheets were prepared for locations where redundant divisions were not separated in accordance with the requirements of Section III.G.2.

In addition to evaluating TVA's Appendix R separation analysis during its inspection of January 14-18, 1985, the staff evaluated TVA's associated circuit analysis. TVA's Type II (spurious operation) associated circuit analysis was performed by determining the components that must be prevented from spuriously operating. These components also are listed in the fire SDL diagram and associated keys. TVA then evaluated cable separation for these components in the same way it evaluated those cables that must remain operable for safe shutdown.

The analysis also identified several circuits, not required by Appendix R, that did not have proper fuse/breaker coordination. These circuits were identified as Type I (common power supply) and Type III (common enclosure) associated circuits, and corrective actions were necessary to comply with Section III.G.2 requirements and ensure that adequate electrical protection was provided.

TVA's Appendix R re-evaluation identified 1[illegible]1 plant areas where redundant cabling/equipment associated with those systems necessary to bring the plant to hot and cold shutdown interacted.

In addition, by letters dated October 1, 1981, December 18, 1984, and January 11, March 4, and August 5, 1985, TVA requested 22 additional deviations from the requirements of 10 CFR 50, Appendix R. By letter dated March 19, 1986, TVA withdrew the request for a deviation associated with separation of safe shutdown circuits and components inside the containment. By letters dated May 29 and October 6, 1986, the staff approved the 2 outstanding deviation requests associated with Section III.O, reactor coolant pump oil collection system; the 17 outstanding deviation requests associated with Section III.G., fire protection of safe shutdown capability; and the 2 outstanding deviation requests associated with Section III.L, alternative or dedicated shutdown capabilities, including the deviation request regarding T-cold instrumentation in the auxiliary control room.

In its submittal of December 21, 1984, TVA committed to complete the fire protection modification not associated with the pending deviation requests by June 30, 1986. On July 7-11, 1986, the staff conducted a site assessment to verify that TVA had implemented the required fire protection modifications. Five items that were to be inspected were not yet completed. For those five items, TVA committed to having them completed by June 30, 1987. On June 22-26, 1987, the staff conducted another site visit to inspect these items. As a result of this inspection visit only two items remained open. These open items were the completion of spray systems in the two 480-volt shutdown boardrooms in the auxiliary building and source range nuclear instrumentation (part of Regulatory Guide 1.97 work to be completed after restart).

3.1.2 Evaluation

3.1.2.1 Compliance with 10 CFR 50, Appendix R, Section III.G (SNPP Part 7.2.1)

Section III.G of Appendix R to 10 CFR 50 requires in part that one train of systems necessary to achieve and maintain hot shutdown condition be free of fire damage. For cables or equipment located within the same fire area outside containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

(1) separation by a fire barrier having a 3-hour rating

(2) separation by a horizontal distance of more than 20 feet with no intervening combustibles (In addition, fire detectors and an automatic suppression system shall be installed.)

(3) enclosure of one train in a fire barrier having a 1-hour rating (In addition, fire detection and an automatic fire suppression system shall be installed.)

Of the 121 plant-specific interactions identified, TVA's re-evaluation identified 39 significant cable interactions where a fire could jeopardize the plant's ability to achieve and maintain safe shutdown conditions and where additional fire protection and modifications were needed to comply with Appendix R, Section III.G. The staff evaluation of the significant cable interactions, with regard to maintaining one train of redundant safe shutdown systems free from fire damage and therefore satisfying the requirements of the rule, is given below.

<u>Auxiliary Building, Elevation 669'-0'</u>

In corridor 669.0-A1, cables 2PL3011B, 2PL3013B, and 2PL3014B for the Unit 2 centrifugal charging pump (CCP) B-B room cooler and cables 2PP562B and 2PP5646B for Unit 2 CCP B interact with cables 2PL3001A, 2PL3003A, and 2PL3004A for the Unit 2 CCP A-A room cooler and cables 2PP550A and 2PP552A for the Unit 2 CCP A. This cable interaction occurs within the corridor from columns A-5 to A-15 and between column lines S and T. Thus, a postulated fire in this area could cause a loss of both redundant trains of the Unit 2 charging pumps. On this basis, reactor coolant system (RCS) makeup capabilities and reactor coolant pump (RCP) seal injection would be jeopardized.

TVA has rerouted the Unit 2 CCP A-A room cooler cables (2PL3001A and 2PL3003A) and Unit 2 CCP A cables (2PP550A and 2PP552A) out of the interaction area to ensure adequate separation. TVA has indicated that these cables have been wrapped on elevations 714 and 690. Cables 2PL3004A and 2PL3014B have been disconnected.

In addition, in corridor 669.0-A1 cables 1PP562B and 1PP564B for the Unit 1 CCP-B and cables 1PL3011B and 1PL3013B for the Unit 1 CCP B-B room cooler interact with cables 1PP550A and 1PP552A for the Unit 1 CCP A and cables 1PL3001A and 1PL3003A for the Unit 1 CCP A-A room cooler. This cable interaction occurs within the corridor from columns A-3 to A-6 and between column lines S and T. Therefore, a postulated fire in this area could cause a loss of both redundant trains of Unit 1 charging pumps. On this basis, RCS makeup capabilities and RCP seal injection would be jeopardized.

TVA has rerouted to shorten the cable run and enclosed the Unit 1 CCP B-B room cooler cables 1PL3011B and 1PL3013B and Unit 1 CCP B cables 1PP562B and 1PP564B in a 1-hour fire barrier in the interaction area to ensure adequate separation.

A fire in corridor 669.0-A1 also could cause both redundant auxiliary lube oil pumps for the Units 1 and 2 CCPs to fail. Therefore, to ensure that the CCPs will start, TVA has installed auxiliary lube oil pump bypass start capabilities for the CCPs. This condition was identified by TVA's re-evaluation cable interaction study no. 93 and 68.

On the basis of the above modifications and the sprinkler protection in corridor 669.01-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of the CCP system would be maintained free from fire damage.

Auxiliary Building, Elevation 690'-0"

In auxiliary building common area 690.01-A1 near column line A-2 and T, the following cables associated with the Units 1 and 2 train A component cooling water (CCW) pumps are routed at the top edge of the partial height fire barrier wall separating the CCW system pump redundant divisions:

Unit 1 CCW Pump A Conduits	Unit 2 CCW Pump A Conduits
1PL4725A	2PL4725A
1PL4726A	2PL4726A
1PL4731A	2PL4731A
	2PL4721A

A postulated exposure fire associated with the train B CCW pump for either Unit 1 or 2 could cause fire damage to the cabling for the train A CCW pumps of either unit. In addition, the postulated fire condition could damage cables 1PL4735S and 1PL4736S associated with the CCW pump C-S. Thus, if an exposure-type fire were to occur on the B train side of the fire barrier separating the redundant pumps, both redundant trains of CCW pumps could be rendered inoperable.

TVA has enclosed the Unit 1 train A CCW pump conduits (1PL4725A, 1PL4726A, and 1PL4731A) in a fire barrier having a 1-hour rating. TVA has also indicated that they have rerouted Unit 2 CCW pump A conduits 2PL4725A, 2PL4726A, 2PL4731A and 2PL4721A. The fire barrier will extend protection to the subject conduits until there is 20 feet of horizontal separation from the Units 1 and 2 train B CCW pumps. This condition was identified by TVA's re-evaluation cable interaction study no. 4.

On the basis of its approval on May 29, 1986, of TVA's outstanding deviation requests, the modifications proposed above, and the sprinkler protection in common area 690.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area near column lines A-2 and T, one train of the CCW system would be maintained free from fire damage.

From columns A-11 to A-13 and between column lines Q and R, Channel I RCS temperature loop cables 2PM591I, 2PM778I, 2PM686I, and 2PM371I interact with Channel II RCS temperature loop cables 2PM595II, 2PM784II, 2PM691II, and 2PM876II. A postulated exposure fire in this plant area could cause a loss of all temperature indication for all four Unit 2 RCS loops.

TVA has enclosed cables 2PM591I, 2PM778I, 2PM686I, 2PM871I, 2PM595II, 2PM784II, 2PM691II, and 2PM876II on auxiliary building elevation 690'-0' in a 1-hour-fire-rated fire barrier. This condition was identified by TVA's re-evaluation cable interaction study no. 49.

A postulated fire condition in this plant area will also cause a loss of cabling associated with all three channels of pressure indication for all four Unit 2 steam generators. Therefore, TVA has enclosed conduit 2PM2084I containing cables 2PM1335I, 2PM1474I, 2PM1595I, and 2PM1715I on auxiliary building elevation 690'-0" in a 1-hour fire barrier. This condition was identified by TVA's re-evaluation cable interaction study no. 51.

As a result of the above modifications and the sprinkler protection in common area 690.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area from columns A-11 to A-13 and between column lines Q and R, the temperature indication for all four Unit 2 RCS loops and the pressure indication for all four Unit 2 steam generators would be maintained free from fire damage.

From columns A-5 to A-13 and between column lines R and T, the following cables associated with A and B train CCP room coolers, CCW pumps, CCP, and essential raw service water (ERCW) pumps interact:

Cable Identifier	Safe Shutdown Component
2PL3001A	Unit 2 CCP A-A room cooler
2PL3003A	
2PL3011B	Unit 2 CCP B-B room cooler
2PL3013B	
2PP550A	Unit 2 CCP A-A
2PP552A	
2PP562B	Unit 2 CCP B-B
2PP564B	
2PL4725A	Unit 2 CCW pump A-A
2PL4726A	
2PL4731A	
2PL4739A	Common CCW pump C-S
2PL4739B	
2PL4742B	Unit 2 CCW pump B-B
2PL4743B	
2PL4748B	
1PP700B	ERCW pump L-B
1PP712B	ERCW pump N-B

2PP700B	ERCW pump M-B
2PP712B	ERCW pump P-B

A postulated exposure fire in this plant area could jeopardize both redundant trains of Unit 2 charging pump room coolers, preclude all RCS makeup and RCP seal injection capabilities, and cause a loss of component cooling water to safe shutdown systems.

TVA has rerouted cables 2PP550A and 2PP552A for the Unit 2 train A CCP out of the interaction area (and wrapped within the interaction area) to ensure adequate separation. TVA also has installed auxiliary lube oil pump bypass start capabilities for CCPs (auxiliary lube oil pump cables not tabulated). This bypass switch allows the CCPs to be started without the auxiliary lube oil pumps running.

Cables 2PL4739B and 2PL4731A are necessary for local control of the CCW. TVA has rerouted these cables and enclosed them in a 1-hour fire barrier where necessary to ensure adequate separation. The train B ERCW cables have been enclosed in a 1-hour fire barrier to achieve adequate separation from the train A CCW pumps for Units 1 and 2. In addition, train A CCP room cooler fan cables for Unit 2 have been rerouted (and wrapped in the interaction area) to provide adequate separation from the train B CCP cabling located in this area. The remaining listed cables are contained in two raceways that are separated (or wrapped) as part of an Appendix R deviation request commitment. In this cable interaction area, TVA also has enclosed pressurizer pressure instrument cable 2PM1086III in a 1-hour fire barrier along its entire route through auxiliary building common area on elevation 690'-0". These interaction conditions and corrective actions were identified by TVA's interaction study no. 92.

On the basis of the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests, the above modifications, and the sprinkler protection in common area 690.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area between columns A-5 to A-13 and between column lines R and T, one train of CCP room coolers, RCS makeup, and RCP seal injection capabilities and the CCW system would be maintained free from fire damage.

Between columns A-4 and A-3 near column line T, cables 1PP785B and 2PP785B associated with Units 1 and 2 train B ERCW MCCs interact with CCW pumps 1A-A, C-S, 1B-B, 2B-B, and 2A-A. Thus, a postulated fire in this plant area could preclude train B ERCW water supply to CCW heat exchangers.

TVA has enclosed cables 1PP785B and 2PP785B in a 1-hour fire barrier where there is not 20 feet of separation between trains. This interaction condition and corrective action were identified by TVA's interaction study no. 102.

As a result of the staff's evaluation and approval (May 29, 1986) of TVA's outstanding deviation requests, the above modification, and the sprinkler protection in common area 690.0-A1, the staff finds there is a reasonable assurance that, if a fire occurred between columns A-4 and A-3 near column line T, the train B ERCW system would be maintained free from fire damage.

From Columns A-2 and A-5 and between column lines R and U, the following train B ERCW cables interact with train A CCP cables:

Cable Identifier	Safe Shutdown Component
1PP700B	ERWC pump L B
1PP712B	ERWC pump N-B
2PP700B	ERWC pump M-B
2PP712B	ERWC pump P-B
1PP550A	Unit 1 CCP A-A
1PP552A	
1PL6145A	Unit 1 CCP A-A auxiliary lube oil pump
1PL6149A	
1PL3001A	Unit 1 CCP A-A cooler fan and valve FCV-67-168
1PL3003A	
1PL4725A	Unit 1 CCW pump A-A
1PL4726A	
1PL4731A	

Cable Identifier	Safe Shutdown Component
2PP700B	ERCW pump M-B
2PP704B	
2PP706B	
1PP679A	
1PP712B	ERCW pump P-B
2PP716B	
2PP718B	
PP328A	ERCW to diesel generator
PP330A	Unit 1 heat exchanger A-A valve 1-FCV-67-660
PP448A	ERCW to diesel generator
PP450A	Unit 2 heat exchanger A-A valve 2-FCV-67-66
1PP693A	ERCW pump Q-A
1PP691A	
1PP681A	ERCW pump J-A
2PP679A	ERCW pump K-A
2PP681A	
2PP691A	ERCW pump R-A
2PP693A	
1PP475A	Diesel generator breaker 1912
2PP454A	Diesel generator breaker 1922
2PP475A	Unit 2 diesel generator train A breaker control
PP302A	Unit 1 diesel generator train A start/stop function
PP304A	
PP306A	
PP310A	
PP312A	
1PP460B	Diesel generator breaker 1914
1PP480B	
2PP480B	Diesel generator breaker 1924
PP662B	Unit 2 Diesel generator train B start/stop function
PP666B	
PP670B	
PP672B	

A postulated fire in this plant area could cause a loss of ERCW water supply to both redundant trains of the Units 1 and 2 diesel generator heat exchangers and preclude the ERCW water supply to both redundant trains of component cooling system heat exchangers.

In addition, this postulated fire condition could render both redundant trains of onsite power capabilities for both units inoperable.

TVA has installed a 1-hour fire-rated wall to separate A and B ERCW cables and breakers 1914 and 1912 cables associated with onsite power capabilities from Units 1 train B diesel generator to Unit 1 train B 6.9-kV shutdown board and Unit 1 train A diesel generator to Unit 1 train A shutdown board, respectively. The 1-hour fire barrier wall was installed down the A-8 column line on auxiliary building elevation 714'0" from Q line to a point 20 feet east of Q line. This barrier also will separate breakers 1922 and 1924 cables as well as the diesel generators 1A and 2B start/stop-function cables.

In addition, TVA has indicated that they have enclosed cable PP328A in conjunction with the firewall for ERCW valves 1-FCV-67-66, 1-FCV-67-67, 2-FCV-67-66, and 2-FCV-67-67 in a 1-hour fire barrier until there is 20 feet of separation from the redundant train. These interaction conditions and their corrective actions were identified by TVA's interaction studies nos. 16, 34, and 82.

Based on the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests, the above fire protection modifications, and the sprinkler protection in common area 714.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area from columns A-6 to A-10 and between column lines Q and S, one train of ERCW and onsite power distribution capabilities would be maintained free from fire damage.

From columns A-6 to A-14 and between column lines Q to U, a postulated fire could involve cables for both Units 1 and 2 motor-driven and turbine-driven auxiliary feedwater pumps, their associated automatic level control valves, and wide and narrow range level indications. This could cause a loss of both redundant trains of auxiliary feedwater to the steam generators.

TVA has indicated that they have rerouted and enclosed in a 1-hour fire barrier the conduits which contain a narrow range level transmitter power cable 2PV255III, and conduit which contain power cable 1PV255III to all four steam generator narrow range level transmitters. In addition, TVA has developed a procedure with regard to regaining manual control of the auxiliary feedwater system with a fire in this plant area. These interactions conditions and their corrective actions were identified by TVA's interaction studies nos. 21 and 41.

As a result of the above fire protection modifications and procedural corrective actions and the sprinkler protection in common area 714.0-A1, the staff finds there is a reasonable assurance that, if a fire occurred in this area from columns A-6 to A-14 and between column lines Q to U, one train of the AFW system and its associated instrumentation would be maintained free from fire damage.

From columns A-4 to A-8 and between column lines Q to R, common power cable (2PV320J) for Channel I RCS temperature loops interacts with the Channel II power cable (2PV330K). Therefore, a postulated fire in this area could cause

Unit 2 RCS temperature indication for all four RCS loops to be rendered inoperable.

TVA has indicated that they have rerouted and enclosed cables 2PV320J and 2PV330K in a 1-hour fire barrier. This modification will ensure that power for Channels I and II RCS temperature instrumentation is not affected by a fire in this plant area. This condition and TVA's corrective actions were identified by TVA's interaction study no. 42.

Based on the above fire protection modifications and the sprinkler protection in common area 714.0-A1, the staff finds there is a reasonable assurance that, if a fire occurred in this area from columns A-4 to A-8 and between column lines Q to R, the power cables for the Unit 2 RCS temperature instrumentation loops would be maintained free from fire damage.

Near column A-12 between column lines Q and R cables associated with Channels I and II, RCS pressure indication instrumentation interacts. Thus, a postulated fire in this area could jeopardize both redundant channels of RCS pressure indication inoperable.

TVA has rerouted Channel I common power cable 2PV320J to shorten its route through this plant area. In addition, this cable is enclosed in a 1-hour fire barrier in the area where it interacts with cables associated with RCS pressure instrumentations P-68-66 and P-68-342C. These interaction conditions and proposed modifications were identified by TVA's interaction study no. 43.

Thus, as a result of the bove fire protection modification and the sprinkler protection in column area 714.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area near column A-12 between column lines Q and R RCS, pressure indication would be maintained free from fire damage.

The area from columns A-11 to A-13 and between Q and U contains the following trains A and B cables for safe shutdown systems:

Cable Identifier	Safe Shutdown Component
2PL3001A	Unit 2 CCP A-A room cooler
2PL3003A	
2PL3011B	Unit 2 CCP B-B room cooler
2PL3013B	
2PP550A	Unit 2 CCP A-A
2PP552A	
2PP554A	
2PP556A	
2PP562B	Unit 2 CCP B-B
2PP564B	
2PP566B	
2PP568B	
2PL4725A	Unit 2 CCW pump A-A
2PL4726A	
2PL4727A	

2PL4731A	
2PL4732A	
2PL4738B	Common CCW pump C-S
2PL4638B	
2PL4742B	Unit 2 CCW pump B-B
2PL4743B	
2PL4744B	
2PL4748B	
2PL4749B	

A postulated fire in this plant area could jeopardize both redundant trains of Unit 2 component cooling and charging pumps.

To provide adequate separation between redundant centrifugal charging and component cooling pumps, TVA has rerouted the cables associated with Unit 2 train A CCP and CCW pumps out of the subject area of fire influence. In addition TVA has indicated that cables for the Unit 2 train A CCP room cooler and one train of pressurizer level instrumentation were rerouted and have been enclosed in a 1-hour fire barrier within the subject area of fire influence. These cable interaction conditions were identified by TVA's interaction study no. 86.

On the basis of the above fire protection modifications, the staff's evaluation and approval (May 29, 1986), of outstanding Appendix R deviation requests, and the sprinkler protection in common area 714.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area from columns A-11 to A-13 and between Q and U, one train of the CCW and CCP systems would be maintained free from fire damage.

Auxiliary Building, Elevation 734'-0"

In the Unit 1 480-volt shutdown board room 1B2-B, train A cable trays transverse the southwest corner of the room. The following cables are associated with these train A cable trays:

Cable Identifier	Safe Shutdown Component
1PP679A	ERCW pump J-A
PP681A	
PP691A	ERCW pump Q-A
PP693A	
PP679A	ERCW pump K-A
PP681A	
PP691A	
RCW pump R-A	
PP693A	
P373A	Diesel generator breaker 1912
P374A	
P458A	
P378A	
PP475A	
PP478A	

PP4545A	
PP475A	Unit 2 diesel generator train A breaker control
P469A	Diesel generator breaker 1922
PP478A	Diesel generator breaker 1922
PP498A	
PP454A	
B11I, 1B16I	Normal power feed to 480-volt shutdown board 1A1-A and 1A2-A
B12III, 1B17III	Alternate power feed to 480-volt shutdown board 1A1-A and 1A2-A
75	Unit 1 diesel generator train A emergency stop
PL4900A	Power feed to vital battery charger I

A postulated fire in this plant area could jeopardize the Unit 1 ERCW supply to the emergency diesel generators and CCW heat exchangers. In addition, a postulated fire in this area could render both redundant trains of Unit 1 480-volt power distribution to safe shutdown systems inoperable.

TVA has indicated that cables associated with ERCW pumps J-A, Q-A, K-A, R-A, ERCW valve 1-FCV-67-66 have been protected with open head water spray, and the normal control power feed to the Unit 1 480-volt shutdown boards 1A1-A and 1A2-A were rerouted out of the subject area of fire influence. An alternate supply is available to vital battery charger I. In addition, TVA has protected the train A cable trays transversing the southwest corner of the Unit 1 480-volt shutdown board room 1B2-B with an independent thermal-actuated open-head water spray system from the wall penetration to the floor penetration. These cable interaction conditions were identified by TVA's interaction studies nos. 22 and 81.

As a result of the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation request, the 1-hour barrier installation, and the water spray system installation, the staff finds there is reasonable assurance that, if a fire occurred in the Unit 1 480-volt shutdown board room 1B2-B, one train of the ERCW and the 480-volt power distribution system would be maintained free from fire damage.

In the Unit 2 480-volt shutdown board room 2A2-A, from columns A12 to A13 between column lines Q and R, B train cable trays transverse this area. The following cables are associated with these train B cable trays:

Cable Identifier	Safe Shutdown Component
2PP704B	ERCW pump M-B
2PP706B	
2PP716B	ERCW pump P-B
2PP718B	
1PP704B	ERCW pump L-B
1PP706B	
1PP716B	ERCW pump N-B
1PP718B	
2PP562B	Unit 2 CCP B-B
2PP564B	

2PP566B	
2PP568B	
2PL3013B	Unit 2 CCP B-B pump room cooler
2PP483B	Diesel generator breaker 1924
2PP480B	
PP377B	
PP477B	
2PP460B	Breaker 1924
2PP377B	
2B25IV	Normal and alternate control power feed to Unit 2 train B
2B30IV	480-volt shutdown boards
2B26II	
2B31II	
B78B	Unit 2 diesel generator B-B remote control

A postulated fire in this plant area could jeopardize Unit 2 ERCW supply to the emergency diesel generators and CCW heat exchangers. In addition, a postulated fire in this area could render both Unit 2 redundant trains of the 480-volt power distribution to safe shutdown systems inoperable.

TVA has protected the cables associated with ERCW pumps M-B, P-B, L-B, N-B, Unit 2 train B CCP and room cooler cables. Cables associated with control power for the train B 480-volt shutdown board routed in the cable trays transversing the northwest corner of the Unit 2 480-volt shutdown board room 2A2-A were routed out of the area with an independent thermal-actuated open-head water spray system from the wall penetration to the floor penetration. These cable interactions were identified by TVA's interaction studies nos. 23, 75 and 79.

On the basis of its evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests and completion of the water spray system installation in the Unit 2 480-volt shutdown board room 2A2-A, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of ERCW and the 480-volt power distribution system would be maintained free from fire damage.

In the Unit 2 480-volt shutdown board room 2A1-A, cables 2B25IV, 2B30IV, 2B26II, and 2B31II, associated with the 125-volt dc control power normal and alternate supply to the Unit 2 480-volt shutdown board 2B1-B, interact with the Unit 2 480-volt shutdown board 2A1-A. Therefore, a postulated fire in this area could render both redundant Unit 2 480-volt shutdown boards inoperable, causing a loss of all control power to safe shutdown systems.

TVA has indicated that conduit 2B29IV containing cables 2B25IV and 2B30IV (normal control power) has been rerouted out of the interaction area of the Unit 2 480-volt shutdown board room 2A1-A. This modification was identified by TVA's interaction study no. 83.

Based on the above fire protection modification and the sprinkler protection in the 480-volt shutdown board room 2A1-A, the staff finds there is reasonable assurance that if a fire occurred in this area one train of the 480-volt control power distribution capabilities would be maintained free from fire damage.

The Unit 2 train A 6.9-kilovolt shutdown board room contain cables 1PP765B, 1PP753B, and 1PP762B, which are the 6.9-kilovolt power feeds from the Unit 1 train B 6.9-kilovolt shutdown board to the 480-volt shutdown transformer. These cables are associated with the Unit 1 480-volt shutdown boards 1B1-B and 1B2-B and interact with Units 1 and 2 train A 6.9-kilovolt shutdown boards. Thus, a postulated fire condition in this room could render all Unit 1 power distribution capabilities inoperable.

TVA has enclosed cables 1PP765B, 1PP753B, and 1PP762B in a 1-hour fire barrier as they pass through Unit 2 train A 6.9-kilovolt shutdown board room 2A1-A. This modification was identified by TVA's interaction study no. 3.

Based on the above fire protection modification and the sprinkler protection in the Unit 2 train A 6.9-kilovolt shutdown board room 2A-A, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of Unit 1 power distribution capabilities would be maintained free from fire damage.

In auxiliary control room 734.0-A1, cables contained in cable trays PO-A, PN-A, and PM-A interact with cables in tray PA-B. These cables are for both redundant divisions of safe shutdown equipment having normal to auxiliary transfer switches in the auxiliary instrument rooms. In addition, cable B77A associated with 2-FCV-67-66 interacts with cable B76B associated with 1-FCV-67-67 in the same plant location. A postulated fire in this area could cause a loss of all normal to auxiliary control room Units 1 and 2 safe shutdown functions and ERCW supply to emergency diesel Unit 1 train B and Unit 2 train A heat exchangers.

TVA has enclosed cable trays PO-A, PN-A, and PM-A and cabling associated with 2-FCV-67-66 in a 1-hour fire barrier as they pass through the auxiliary control room. This fire protection modification was identified by TVA's interaction studies nos. 98 and 105.

As a result of the above fire protection modification and the sprinkler protection in auxiliary control room 734.0-A1, there is reasonable assurance that, if a fire occurred in this area, one train of the normal to auxiliary control room safe shutdown transfer function and the ERCW supply to the emergency diesel generators would be maintained free from fire damage.

In 125-volt vital battery board room I 734.0-A4, cables 1B26IV, 1B31IV, 1B25II, and 1B30II (which provide normal and alternative power feed to Unit 1 480-volt shutdown boards 1B1-B and 1B2-B) transverse this room along the east wall. A postulated fire in this area could render safe shutdown equipment and the 125-volt dc control power to train A safe shutdown systems inoperative.

In addition, routed along the east wall of 125-volt vital battery board room IV 734.0-A22 are cables 2B11III, 2B16III, 2B12I, and 2B17I (which provide normal and alternative power feed to the Unit 2 480-volt shutdown boards 2A1-A and 2A2-A). Thus, a postulated fire in this area could render Unit 2 train A safe shutdown equipment and the 125-volt dc control power to Unit 2 train B safe shutdown sytems inoperative. This condition was identified by TVA's Appendix R re-evaluation study no. 107.

TVA enclosed conduit 1B29II containing cables 1B25II and 1B30II and conduit 2B20II containing cables 2B11III and 2B16III in a 1-hour fire barrier as they pass through vital battery board rooms I and IV, respectively.

Thus, based on the above fire protection modifications and the sprinkler protection in the 125-volt vital battery board room I 734.0-A4 and 125-volt vital battery board room IV 734.0-A22, the staff finds there is reasonable assurance that, if a fire occurred in either of these areas, one train of the 480-volt electrical power distribution capabilities would be maintained free of fire damage.

In Unit 2 train B 6.9-kilovolt shutdown board room 734.0-A24, cables 2PP759A, 2PP750A, and 2PP756A (which are the 6.9-kilowatt power feeds from the Unit 2 train A 6.9-kilovolt shutdown board to the 480-volt shutdown transformers associated with Unit 2 480-volt shutdown boards 2A1-A and 2A2-A) are routed on the ceiling to the rear of the Unit 2 train B 6.9-kilovolt shutdown board 2B-B. A postulated fire in this area could jeopardize both redundant trains of Unit 2 power distribution capabilities to safe shutdown systems.

TVA has enclosed cables 2PP759A, 2PP750A, and 2PP756A in a 1-hour fire barrier as it passes through the Unit 2 train B 6.9-kilovolt shutdown board room. This condition was identified by TVA's Appendix R re-evaluation interaction study no. 2.

As a result of the above fire protection modification and the sprinkler protection in the Unit 2 train B 6.9-kilovolt shutdown board room 734.0-A24, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of Unit 2 power distribution capabilities would be maintained free from fire damage.

In the Unit 2 train A 6.9-kilovolt shutdown board 734.0-A2 from columns A3 and A4 and between column lines R and U, the following safe shutdown cables interact:

Cable Identifier	Safe Shutdown Component
1PP550A	Unit 1 CCP A-A
1PP552A	
1PP553A	
1PP554A	
1PP556A	
1PP557A	
1PP555A	
1PL6145A	Unit 1 CCP A-A auxiliary lube oil pump
1PL6146A	
1PL6147A	
1PL6148A	
1PL3002A	Unit 1 CCP A-A room cooler and FCV-67-168

1PL3003A	
1PL4729A	Unit 1 CCP pump A-A
1PP564B	Unit 1 CCP B-B
1PL6152B	Unit 1 CCP B-B auxiliary lube oil pump
1PL6155B	
1PL6156B	
1PL3013B	Unit 1 CCP B-B room cooler and FVC-67-170
2PL4733B	CCW pump C-S
2PL4734B	
2PL4737B	
1PL4735S	
1PL4736S	

Thus, a postulated fire in this area could render both redundant trains of Unit 1 charging pumps inoperable.

TVA has enclosed cables 2PL4733B and 2PL4734B in a 1-hour fire barrier where there is not 20 feet of separation from the train A functions associated with the Unit 2 train A 6.9-kilovolt shutdown board room. TVA also has disconnected cable 2PL4737B at the shutdown breaker. This will preclude spurious operation of the CCW pump CS interlock function in the event of a fire in this area.

In addition, TVA has rerouted the Unit 1 CCP-B cables out of the Unit 2 train A 6.9-kilovolt shutdown board room until there was 20 feet of separation from the train A function. TVA has indicated that the cable was also wrapped with a 1-hour barrier in the interaction area. TVA also rerouted cable 1PL3003A associated with the Unit 1 CCP cooler fan A-A to gain 20 feet of separation from CCW pump C-S. These conditions were identified by TVA's re-evaluation interaction study no. 66.

Thus, based on the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests, the above fire protection modifications, and the sprinkler protection in Unit 2 train A 6.9-kilovolt shutdown board room 734.0-A2, the staff finds there is reasonable assurance that, if a fire occurred in the area, one train of the CCP system will be maintained free from fire damage.

In the Unit 1 480-volt shutdown board 1B1-B room 734.0-A6, cables 1B11I and 1B16I (which are the 125-volt normal control power feeds to Unit 1 480-volt shutdown boards 1A1-A and 1A2-A) interact with 480-volt shutdown board 1B1-B and associated cables. Thus, a postulated fire condition in this area could jeopardize both redundant trains of the 480-volt power capabilities to safe shutdown equipment.

TVA has rerouted conduit 1B20I and junction box 1622 (which contains cables 1B19I, 1B11I, and 1B16I) as they pass through the Unit 1 480-volt shutdown board room 1B1-B. This condition was identified by TVA's Appendix R re-evaluation interaction study no. 80.

As a result of the above fire protection modification and the sprinkler protection in the 6.9 kV shutdown board room 734.0-A2, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of the 480-volt power distribution capabilities would be maintained free from fire damage.

Auxiliary Building, Elevation 749'-0"

In the Unit 2 train B 480-volt transformer room 749.0-A10, cables 2PL4975A and 2PL4978A from 480-volt shutdown boards 2A1-A and 2A2-A to diesel generator auxiliary boards 2A1-A and 2A2-A interact with the 480-volt shutdown and emergency transformers 1B1-B, 1B2-B, and 1B-B and associated cables to diesel generator auxiliary boards 2B1-B and 2B2-B. Therefore, a postulated fire in this area could cause a loss of all Unit 2 onsite power capabilities to safe shutdown systems.

TVA enclosed cables 2PL4975A and 2PL4978A in a 1-hour fire barrier as they pass through the Unit 2 train B 480-volt transformer room. This condition was identified by TVA's Appendix R re-evaluation interaction study no. 11.

Based on the above fire protection modification and the sprinkler protection in the 480-volt transformer room 749.0-A10, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of the Unit 2 onsite power distribution capabilities will be maintained free from fire damage.

Power cables PP710B, PP711B, PP590B, and PP591B to the Units 1 and 2 train B 6.9-kilovolt shutdown boards interact with the Unit 2 train A 480-volt reactor motor-operated valve (MOV) boards and associated cables at the conduit bank near column A-11 and column line I in the Unit 2 train A 480-volt reactor MOV board room 749.0-A16. Therefore, a postulated fire in this plant area could jeopardize the operation of all Unit 2 train A safe shutdown MOVs and Unit 2 train B safe shutdown equipment.

TVA has enclosed 6.9-kilovolt shutdown boards 1B-B and 2B-B power supply cables PP710B, PP711B, PP590B, and PP591B in a 1-hour fire barrier as they pass through the 480-volt reactor MOV board room 2A. This modification was identified by TVA's interaction study no. 14.

Thus, as a result of the above fire protection modification and the sprinkler protection in the Unit 2 train A 480-volt reactor MOV board room 740.0-A16, the staff finds there is reasonable assurance that, if a fire occurred in this area, Unit 2 train B 6.9-kilovolt power distribution capabilities will be maintained free from fire damage.

Cables 1PL4982B and 1PL4985B from the Unit 1 480-volt shutdown boards 1B1-B and 1B2-B to the diesel generator auxiliary boards 1B1-B and 1B2-B interact with the 480-volt shutdown and emergency transformers 1A1-A, 1A2-A, and 1A-A in the Unit 1 train A 480-volt shutdown transformer room 749.0-A7. Postulating a fire in this plant area could cause a loss of all Unit 1 onsite power capabilities to safe shutdown systems.

TVA has enclosed cables 1PL4982B and 1PL4985B in a 1-hour fire barrier as they pass through the Unit 1 train A 480-volt shutdown transformer room. This condition was identified by TVA's interaction study no. 10.

Based on the sprinkler protection in the Unit 1 train A 480-volt shutdown transformer room and the above modification, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of the onsite power capabilities for Unit 1 would be maintained free from fire damage.

Auxiliary Building, Elevation 759'-0"

In Unit 2 control rod-driven equipment room 759.0-A3, cables 2PL4975A and 2PL4978A from Unit 2 480-volt shutdown boards 2A1-A and 2A2-A to the diesel generator auxiliary board interact with cables 2PL4982B and 2PL4985B from the Unit 2 480-volt shutdown boards 2B1-B and 2B2-B to the diesel generator boards. In addition, cables PP590B, PP591B, PP710B, 1PP820B, and 2PP820B to diesel generators 1B and 2B are located in this area. Thus, a postulated fire in this area could cause a loss of heating, ventilation, and air conditioning (HVAC), diesel fuel transfer, and ERCW support systems to emergency diesel generators 2A and 2B.

TVA has enclosed cables 2PL4975A and 2PL4978A in a 1-hour fire barrier as they pass through the Unit 2 control rod-driven equipment room. This condition was identified by TVA's interaction study no. 13.

As a result of the above fire protection modification and the sprinkler protection in control rod drive equipment room 759.0-A3, the staff finds there is reasonable assurance that, if a fire occurred in this area, the A train of those systems necessary to support Unit 2 onsite power capabilities will be free from fire damage.

Cables 1PL4982B and 1PL4985B to Unit 1 diesel generator auxiliary boards 1B1-B and 1B2-B interact with train A 480-volt cables 1PL4975A and 1Pl4978A to Unit 1 diesel generator auxiliary boards 1A1-A and 1A2-A in Unit 1 control rod drive equipment room 759.0-A1. A postulated fire in this plant area could cause a loss of HVAC, diesel fuel transfer, and ERCW support systems to diesel generators 1A and 1B.

TVA has enclosed cables 1PL4982B and 1PL4985B in a 1-hour fire barrier as they pass through the Unit 1 control rod drive equipment room. This condition was identified by TVA's interaction study no. 12.

Thus, as a result of the above modification and sprinkler protection in the Unit 1 control rod drive equipment room 759.0-A1, the staff finds there is reasonable assurance that, if a fire occurred in this area, one train of those systems necessary to support Unit 1 onsite power capabilities would be maintained free from fire damage.

Auxiliary Building Between Elevations 669'-0", 690'0", and 714'-0"

Near the unprotected north stairway opening associated with the auxiliary building common area from columns A4 to A5 and between column lines S and T on elevation 669'-0", cable 1SG220A for dc control power to the turbine-driven auxiliary feedwater pump interact through this opening with cables 1PP650A, 1PP652A, 1PP662B, and 1PP664B for the 1A-A and 1B-B motor-driven auxiliary feedwater pumps and 1SG221B for alternate dc control power to the turbine-driven auxiliary feedwater pump on elevation 690'-0". In addition, cables 1PP700B, 1PP712B, 2PP700B, and 2PP712B for ERCW pumps L-B, N-B, M-B, and P-B on elevation 690'-0" interact through this opening with cables PP328A, PP330A, PP448A, and PP450A associated with diesel generator heat exchanger valves 1-FVC-67-66 and 2-FCV-67-66 on elevation 714'-0". Thus a postulated fire on elevation 669'-0" in the area of the unprotected stairway opening could jeopardize ERCW to Units 1 and 2 diesel generators and impact the operability of both Unit 1 redundant motor-driven and turbine-driven auxiliary feedwater pumps.

In regard to interaction studies nos. 104 and 6, TVA has installed additional closely spaced sprinklers around the perimeter of the north stairway at each elevation. When the sprinkler is actuated, this arrangement will form a water curtain, which should preclude fire propagation from one auxiliary building elevation to another.

Therefore, based on the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests and completion of the sprinkler water curtain around the north stairway opening, the staff finds that there will be reasonable assurance that, if a fire occurred in the area of the stairway, one train of the Units 1 and 2 ERCW and Unit 1 AFW systems would be maintained free of fire damage.

In the area of the unprotected south stairway opening associated with the auxiliary building common area from columns A11 and A12 and between column lines S and T on elevation 669'-0", cables 2SG220A for dc control power to the turbine driven auxiliary feedwater pump interact through this opening with cables 2PP662B, 2PP664B, 2PP650A, and 2PP652A for the Unit 2 train A and B motor-driven auxiliary feedwater pumps and 2SG221B for alternate dc control power to the turbine-driven auxiliary feedwater pump on elevation 690'-0". In addition, on elevation 669'-0", cables 2PP550A, 2PP552A, 2PP562B, and 2PP564B for charging pumps 2A-A and 2B-B interact through this opening with 2PL4731A, 2PL4734B, 2PL4742B, 2PL4743B, and 2PL4748A for Unit 2 train A, train B and common component cooling system pumps on elevation 690'-0" and cables 2PL4725A, 2PL4726A, and 2PL4732A for component cooling system Unit 2 train A pump 2A-A on elevation 714'-0". Therefore, a postulated fire on elevation 669'-0" in the area of the unprotected stairway opening could impact the operability of both redundant trains of Unit 2 auxiliary feedwater capabilities, charging pumps, and component cooling system pumps.

In regard to interaction studies nos. 57 and 101, TVA has installed additional closely spaced sprinklers around the perimeter of the south stairway at each elevation. When the sprinkler is actuated, this arrangement will form a water curtain, which should preclude fire propagation from one auxiliary buildng elevation to another. TVA has indicated that cabling associated with the CCS pump 2A-A has been routed out of the interaction area.

Thus, as a result of the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests and completion of the sprinkler water curtain around the south stairway opening, the staff finds there will be reasonable assurance that, if a fire occurred in the area of the stairway, one train of the Unit 2 AFW, CCP, and CCW systems would be maintained free from fire damage.

3.1.2.2 Compliance with 10 CFR 50, Appendix R, Section III.J (SNPP Part 7.2.2)

The new fire shutdown logic (SDL) identified additional plant areas where operator action is required, necessitating additional emergency lights in these areas and in access/egress routes. Some of the emergency lights had 25-watt lamps, whereas 10-watt lamps must be used to ensure there is an 8-hour capacity. As an interim measure, the operations staff had portable battery-powered lighting to use if the normal lighting, standby lighting (onsite powered), and dc lighting (station batteries) systems fail. Design changes were made to replace the 25-watt lamps with 10-watt lamps and to add more than 50 additonal light packs in various plant areas.

During July 7 through 11, 1986, the staff conducted a site visit and verified the adequacy of the emergency lighting. For a fire within the control room, TVA procedure AOI-27 (Control Room Inaccessibility (Revision 5)), lists a number of manual operations required for plant shutdown. Manual operations must be conducted in the following plant areas:

- 6.9-kilovolt shutdown board rooms A and B for each unit
- 480-volt shutdown board rooms (four rooms/unit)
- 480-volt reactor MOV board rooms (four rooms/unit)
- diesel generator building, 480-volt diesel generator auxiliary board rooms (four rooms)

During the site visit, emergency lighting tests were conducted in electrical board rooms 734.0-A2, 734.0-A5, 749.0-A15, and 749.0-A16. Based on these tests, the lighting provided in these rooms met the minimum requirements of Appendix R, Section III.J.

3.1.2.3 Compliance with 10 CFR 50 Appendix R, Section III.O (SNPP Part 7.2.3)

The drain piping between the RCP motor oil collection basins and the containment floor (oil drains to the auxiliary reactor building sumps) is designed to Category I requirements so the piping will not fail during a safe shutdown earthquake and damage nuclear safety-related equipment. This drain piping to the auxiliary reactor building sump has not been designed to maintain its pressure boundary integrity after a safe shutdown earthquake. The RCP motors, the lubricating oil systems, and the auxiliary reactor building sump are designed to seismic Category I requirements so they will not fail during a safe shutdown earthquake. Therefore, random oil leaks are not assumed to occur simultaneously with a design event because of the system design. TVA contends that the total system provides more than reasonable assurance that a RCP motor lubrication oil fire will not occur as a result of a seismic event. Assuming then only a random single failure, the oil collection system would only be required to hold the oil resulting from the largest spill resulting from that single failure.

The sump vents do not require the installation of flame arresters because the high flashpoint characteristics (390°F) of the RCP motor lube oil preclude the hazard of fire flashback.

Based on the above system description and the staff's evaluation and approval (May 29, 1986) of TVA's outstanding Appendix R deviation requests, the staff finds there is reasonable assurance that the existing RCP oil collection system provides an equivalent level of fire safety to that required by the technical requirements of 10 CFR 50, Appendix R, Section III.O.

3.1.2.4 Interim Compensatory Fire Protection Measures (SNPP Part 7.2.4)

In accordance with the NRC's Confirmatory Action Letter issued August 10, 1984, TVA established roving firewatches to provide continued surveillance of selected areas in the auxiliary building, control building, and the turbine building. These firewatches covered areas of the plant that contain cable/safe shutdown system interactions that did not meet the requirements of 10 CFR 50, Appendix R, Section III.G. In addition, these roving firewatches were required to cover their assigned areas at least once an hour and document their actions in accordance with TVA's Operations Section Letter Administrative 73.

As a result of the recent inspection (March 13-17, 1988) the staff found additional interactions which had to be addressed. These interactions and the corrective steps taken are detailed in Inspection Reports 50-327/88-24 and 50-328/88-24. The corrective actions resulting from this inspection included the addition of sprinkler heads to the pre-action system in the Reactor Building Annulus and the continuation of a fire watch in areas of the reactor auxiliary building where cable interactions between the VCT isolation valve and the B train centrifugal charging pump exist. Fire watch coverage was beeing maintained there because of an interaction for the source range nuclear instrumentation. This was to be corrected during the next refueling outage. Since TVA was already covering the pertinent areas for this interaction, the use of additional fire watches was not necessary.

3.1.3 Conclusion

Based on its evaluation, the staff has concluded that upon completion of the fire modifications and implementation of the procedural corrective actions associated with TVA's deviation requests as identified in the staff's SERs of May 29 and October 6, 1986; and modifications and procedures as identified in Inspection Reports 50-327/88-24 and 50-328/88-24. TVA's Appendix R program will provide an acceptable level of fire protection, equal to that required by 10 CFR 50, Appendix R, Sections III.G, III.J, III.L, and III.O.

3.2 Environmental Qualification of Electric Equipment Important to Safety

3.2.1 Compliance with 10 CFR 50.49

3.2.1.1 Introduction

A licensee must demonstrate that equipment that is used to perform a necessary safety function is capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement (which is in General Design Criteria (GDC) 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50) is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this electrical equipment capability are in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; in NUREG-0588, "Interim Staff Position on Environmental Qualification on Safety-Related Electrical Equipment" (which supplements IEEE Standard 323 and various NRC regulatory guides and industry standards); and "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (Division of Operating Reactors (DOR) Guidelines).

On February 8, 1979, the NRC Office of Inspection and Enforcement (IE) issued to all licensees of operating plants (except those included in the systematic evaluation program (SEP)) IE Bulletin (IEB) 79-01, "Environmental Qualification of Class 1E Equipment." This bulletin, together with IE Circular 78-08 (issued on May 31, 1978), required the licensees to review the adequacy of their environmental qualification programs.

On January 14, 1980, NRC issued IEB 79-01B, which included the DOR Guidelines and NUREG-0588 as attachments 4 and 5. Commission Memorandum and Order CLI-80-21, issued on May 23, 1980, stated that licensees must meet the DOR guidelines and portions of NUREG-0588 regarding environmental qualification of safety-related electrical equipment to satisfy those aspects of GDC 4. Supplements to IEB 79-01B further clarified and defined the staff's needs. These supplements were issued on February 29, September 30, and October 24, 1980.

In addition, the staff incorporated license conditions into the license for Sequoyah Unit 1 requiring that TVA (1) provide a report, by November 1, 1980, documenting the qualification of safety-related electrical equipment, (2) establish, by December 1, 1980, a central file location for the maintenance of all equipment qualification records, and (3) comply with NUREG-0588 by June 30, 1982. Item (3) also was included in the licensee for Unit 2 which was issued in 1981.

The staff issued an SER on environmental qualification of safety-related electrical equipment to TVA on June 23, 1981. This SER directed TVA to "either provide documentation of the missing qualification information which demonstrated that safety-related equipment meets the DOR Guidelines or NUREG-0588 requirements or commit to a corrective action [requalification, replacement (etc.)]." TVA was required to respond to NRC within 90 days of receipt of the SER. In response, TVA submitted additional information regarding the qualification of safety-related electrical equipment. This information was evaluated for the staff by the Franklin Research Center (FRC) to (1) identify all cases where TVA's response did not resolve the significant qualification issues, (2) evaluate TVA's qualification documentation in accordance with established criteria to determine which equipment had adequate documentation and which did not, and (3) evaluate TVA's qualification documentation for safety-related electrical equipment located in harsh environments required for implementation of TMI Lessons Learned. FRC issued a Technical Evaluation Report (TER) on March 31, 1983. The staff issued an SER on April 26, 1983, with the FRC TER as an attachment.

A final rule on environmental qualification of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, specifies the requirements for electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment for Sequoyah Units 1 and 2 may be qualified to the criteria specified in either the DOR guidelines or NUREG-0588, except for replacement equipment. Replacement equipment installed after February 22, 1983, must be qualified in accordance with 10 CFR 50.49, using the guidance of Regulatory Guide (RG) 1.89, unless there are sound reasons to the contrary.

The staff met with each licensee for whom FRC had prepared a TER to discuss all remaining open issues regarding environmental qualification, including the acceptability of the environmental conditions for equipment qualification purposes, if this issue had not yet been resolved.

On May 10, 1984, the staff and TVA met to discuss TVA's proposed method to resolve the environmental qualification deficiencies identified in the staff's SER and FRC's supporting TER transmitted on April 26, 1983. Discussions also included TVA's general methodology for compliance with 10 CFR 50.49. The minutes of the meeting and proposed method of resolution for each of the environmental qualification deficiencies are documented in TVA submittals by letters dated March 26, December 23, 1985 and January 29, 1986.

On August 21-22, 1985, TVA shut down both Sequoyah units because of concerns that documentation at TVA nuclear sites might be inadequate for environmental qualification of electrical equipment within the scope of 10 CFR 50.49. This decision was based on the results of a TVA management review of the environmental qualification activities for compliance with 10 CFR 50.49 (conducted by TVA staff and Westec Services, Inc.). Based on this decision and the results of the review, TVA initiated an in-depth program to ensure that environmental qualification of all electrical equipment within the scope of 10 CFR 50.49 was established at Sequoyah and all other TVA nuclear sites.

3.2.1.2 Evaluation

Summary of Review

The staff evaluation of TVA's electrical equipment qualification program is based on the results of a review of (1) TVA's proposed resolutions of the equipment qualification deficiencies identified in the SER and TER; (2) TVA's compliance with the requirements of 10 CFR 50.49; (3) TVA's Corporate Nuclear Performance Plan, Revision 4, and Sequoyah Nuclear Performance Plan, Revision 1; and (4) the staff's equipment qualification audit November 18-22, 1985, and the staff equipment qualification inspections January 6-17, February 10-14, June 23-27, and December 8-12, 1986, and April 6-10, 1987.

Proposed Resolutions of Identified Deficiencies

TVA described its proposed resolutions for the equipment environmental qualification deficiencies identified in the SER and the TER in submittals dated March 26 and December 23, 1985, and January 26, 1986. During its May 10, 1984, meeting with TVA, the staff discussed the proposed resolution of each deficiency for each equipment item identified in the TER and found TVA's approach for resolving the identified environmental qualification deficiencies acceptable. The majority of deficiencies identified were documentaion, similarity, aging, qualified life, and replacement schedule. All open items identified in the SER also were discussed, and the staff found TVA's resolution of these items acceptable.

TVA's approach for addressing and resolving the identified deficiencies includes replacing equipment, performing additional analyses, using additional qualification documentation beyond that reviewed by FRC, obtaining additional qualification documentation, and determining that some equipment is outside the scope of 10 CFR 50.49 and need not be environmentally qualified (equipment located in a mild environment). The staff discussed the proposed resolutions in detail, on an item-by-item basis, with TVA during the meeting of May 10, 1984.

Replacing or exempting equipment, for an acceptable reason, is an acceptable method for resolving environmental qualification deficiencies. More lengthy discussions with TVA concerned the use of additional analyses or documentation. Although the staff did not review the additional analyses or documentation during the meeting, the staff did dicusss how analysis was being used to resolve deficiencies identified in the TER and the content of the additional documentation to determine the acceptability of these methods. During November 18 through 22, 1985, the staff and a consultant from EG&G Idaho, performed an audit of the Sequoyah electrical equipment environmental qualification binders, and inspected selected equipment. During January 6-17, February 10-14, June 23-27, and December 8-12, 1986, and April 6-10, 1987, the staff and its consultants from Sandia National Laboratories inspected the Sequoyah equipment qualification (EQ) binders and selected equipment and reviewed Sequoyah's implementation of the 10 CFR 50.49 program.

On the basis of its discussions with TVA, the review of the submittals, and the audit and inspections, the staff finds TVA's approach for resolving the identified environmental qualification deficiencies acceptable.

Evaluation of Compliance with 10 CFR 50.49

All equipment that is located in a potentially harsh environment and is required to mitigate the consequences of a design-basis event (DBE) at Sequoyah has been identified in accordance with 10 CFR 50.49(b)(1). TVA identified the equipment by reviewing all systems on which the safety analysis in the FSAR is dependent. Other systems or equipment necessary to support these systems were also identified by TVA.

From the safety systems identified above, TVA conducted a survey of the safety-related equipment within the potentially harsh environment that resulted from a DBE. This survey was conducted using electrical instrument tabulations, mechanical piping drawings, mechanical heating and ventilation drawings, instrumentation and control drawings, electrical equipment drawings, and conduit and grounding drawings to identify the safety-related components. TVA verified the equipment qualification by a field survey of the installed components to certify proper correlations between the qualification documents and the in situ equipment.

TVA determined that DBEs in the area covered by 10 CFR 50.49 are high-energy line breaks (HELBs) both inside and outside of containment and loss-of-coolant accidents (LOCAs). Equipment in the 10 CFR 50.49 program was evaluated for the harsh environments through which it must function and/or not fail. These environments include flooding both inside and outside containment as a result of a DBE.

TVA also evaluated other accidents in Chapter 15 in the Sequoyah FSAR that did not fit the 10 CFR 50.49 DBE definition as interpreted above, but that have the potential to produce environments more severe than those encountered during normal operation or anticipated operational occurrences. These accidents are the waste gas decay tank rupture (WGDTR), the fuel handling accident (FHA), and the steam generator tube rupture (SGTR). The WGDTR and SGTR do not produce unusual temperature or pressure environments, and the radiation environments associated with them are not significant. Radiation doses to equipment necessary for mitigation of these events are less than 104 rads. The FHA results in relatively mild radiological consequences that are restricted to zones-of-influence about the auxiliary building gas treatment system (ABGTS) charcoal beds in both units. The only equipment in the scope of 10 CFR 50.49 affected by the FHA is reflected in the category and operating times document for Sequoyah and is qualified to more harsh environments than that produced by the FHA.

In summary, the 10 CFR 50.49 DBEs at Sequoyah that produce harsh environments are those events which are LOCAs and HELBs inside containment and outside containment. The FHA, occurring in the fuel handling area, is the only other Sequoyah FSAR Chapter 15 event which produces a harsh environment.

TVA environmental data drawings are design output documents that identify and define the conditions of all harsh zones that contain 10 CFR 50.49 scope equipment. These harsh zones result from the DBEs. All environmental parameters necessary for design, procurement, and qualification of equipment in accordance with 10 CFR 50.49 are specified on these drawings. These parameters include normal, abnormal, and accident values for temperature, pressure, relative humidity, radiation (expressed as a 40-year integrated dose and an accident dose), flooding level (from a LOCA and HELB including contribution from spray), and spray chemistry. LOCA and HELB pressure, temperature, and relative humidity profiles are provided. The environmental parameters shown on the drawings are derived from a number of supporting calculations that are referenced on the drawings.

TVA's approach for identifying equipment within the scope of 10 CFR 50.49(b)(1) is in accordance with the requirements of that paragraph, and, therefore, is acceptable.

The paragraphs below summarize the method used by TVA to identify electrical equipment within the scope of 10 CFR 50.49(b)(2), "Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions...."

Electrical equipment that is not safety related and is exposed to harsh accident environments must not fail in a manner that can prevent safety-related electrical equipment from performing its safety function. In response to IE Information Notice 79-22, TVA evaluated devices that are not safety related for their potential to adversely affect safety-related devices as a result of environmentally induced failures. Flow, control and logic diagrams for all safety-related process systems were reviewed to determine all interfaces with equipment that is not safety related. Detailed wiring diagrams were used if the nature of an interface was not clear from the control and logic diagrams. Each interface with equipment that is not safety related was evaluated for its potential to adversely affect safety functions, and the results were documented.

The result of this study showed that six devices (three per unit in the residual heat removal (RHR) system) that are not Class 1E have the potential to adversely affect RHR. However, a failure modes evaluation of these devices concluded that the devices would not adversely affect RHR if their cables were environmentally qualified. These cables are environmentally qualified and have been added to the appropriate binders and the "10 CFR 50.49 List" to ensure their continued qualification. The evaluation also identified cases where disruptive signals could be generated, but in each case the operator has sufficient indication of the event and sufficient time to take corrective action.

TVA performed separate evaluation of the Class 1E power system to investigate the effects of environmentally induced failures. The design basis of the Class 1E power systems includes protective features for coordinated, selective clearing of single random faults and overloads. Most failures of non-qualified equipment from environmental causes will occur in a random fashion. The Class 1E power system is therefore adequately protected by its own design for most environmentally induced failures. The operation of this electrical protection was examined in analyses done to verify the protection of primary containment electrical penetrations and in analyses done to identify associated circuits as defined for 10 CFR 50, Appendix R. The protection has been shown to satisfy its design basis. Submergence and spray effects may, however, cause multiple non-qualified electrical equipment and cable termination faults. This type of failure is outside the design basis of the Class 1E power system. Devices and junction boxes exposed to containment spray or to submergence inside containment or to submergence outside containment that are not qualified for these conditions have been identified. Evaluations of the effects of multiple faults from these circuits on the ability of the Class 1E power system to provide power to essential equipment show that unacceptable degradation of the Class 1E power system would not occur.

The staff finds the methodology being used by TVA acceptable because it provides reasonable assurance that equipment within the scope of 10 CFR 50.49(b)(2) has been identified.

With regard to 10 CFR 50.49(b)(3), TVA evaluated existing system arrangements and identified equipment for the variables defined in RG 1.97, Revision 2. TVA has submitted for staff review a report outlining the results of the review and schedules for modifications. Because the review is not complete, some of the equipment items jointly within the scope of NUREG-0737 and RG 1.97 have not been included in the 10 CFR 50.49 scope. When the RG 1.97 report and equipment lists contained therein have been finalized and accepted by the staff, appropriate equipment not already in the 10 CFR 50.49 scope will be added in accordance with the RG 1.97 implementation schedule.

TVA will complete environmental qualification of the applicable FSAR Class 1E-designed instrumentation and the FSAR post-accident monitoring (PAM) instrumentation before plant restart. For those instruments already added to the plant because of a commitment to meet post-TMI requirements (NUREGs-0578 and -0737), TVA will complete its environmental qualification in accordance with its responses to those NUREGs or any extension granted with respect to those responses.

For instrumentation that is not considered operable or not installed but that will be complete by startup from Unit 2, Cycle 4 refueling outage in accordance with the implementation schedule for RG 1.97 or post-TMI NUREGs, environmental qualification will be complete when the equipment is installed and operable. For that instrumentation that exists at the plants but that was not included in the original PAM instrumentation set but that will be Catagory 1 or 2 RG 1.97 instrumentation, TVA will complete environmental qualification in accordance with the implementation schedule for RG 1.97.

TVA has investigated whether proper consideration of the equipment used in execution of emergency operating instruction (EOI) requirements has been given in the development of the 10 CFR 50.49 equipment scope. The following were considered:

(1) Does the plant operator have reliable instruments to identify and mitigate the consequences of DBEs?

(2) Have those instruments been marked to indicate their importance to the plant operator?

TVA's installed PAM indicators are specifically identified to the main control room operator. The indicators are marked either P1 or P2, which indicates the function these indicators fulfill as PAM channel 1 or PAM channel 2. This method of marking the indicators on the main control room boards shows their importance (rather than requiring that they be singled out in the plant procedures as being environmentally qualified and safety related).

These installed PAM indicators are served by instruments (e.g., transmitter) that are qualified to meet the 10 CFR 50.49 requirements. When other activities are implemented (in accordance with NUREG-0737 and RG 1.97), instruments presently installed but not requiring specific identification and qualification may have to be upgraded.

TVA has concluded that the PAM equipment that will be installed and qualified at plant restart will give the operator the information necessary to identify and mitigate DBEs and will be appropriately marked to indicate its importance.

The staff finds TVA's approach to identifying equipment within the scope of 10 CFR 50.49(b)(3) acceptable because it is in accordance with the requirements of that paragraph.

3.2.1.3 Conclusions

On the basis of its evaluation, the staff has reached the following conclusions with regard to the qualification of electric equipment important to safety within the scope of 10 CFR 50.49:

(1) The Sequoyah electrical equipment environmental qualification program complies with the requirements of 10 CFR 50.49.

(2) TVA's proposed resolutions for each of the environmental qualification deficiencies identified in the staff's SER and the FRC's TER are acceptable.

The staff's findings regarding compliance with 10 CFR 50.49 rely on certain modifications/replacements that must be completed for the affected equipment to be qualified. By letter dated February 27, 1988 TVA provided certification that all restart work is complete.

3.2.2 Superheat Transient (Main Steam Temperature Issue)

TVA designed Sequoyah to withstand an unisolable break in a main steam line either inside containment or in the main steam valve vaults (MSVVs) located outside containment. As part of this design the electrical equipment used during this accident would be required to operate in the high temperatures generated by such a line break. After the plant was completed, the information on which the design was based was changed by Westinghouse. This resulted in increased accident peak temperatures in containment and the valve vaults. As a consequence, the design of the equipment located in these areas required re-evaluation. This issue is discussed in Section III.6 of the SNPP.

3.2.2.1 Main Steam Line Break in Main Steam Valve Vaults

This is an issue generic to recirculating steam generators and is not peculiar to Sequoyah. The issue arises from the consideration that during certain postulated line break accidents, portions of steam generator tubes will be uncovered. This uncovering would result in the release of superheated steam rather than saturated steam. This issue of higher temperatures during a main steam line break (MSLB) was initially considered for inside containment; however, TVA also identified it as an issue in the MSVVs. The valve vaults are adjacent to the containments for Units 1 and 2. Each unit has two vaults (east and west valve vaults).

TVA considered three options in resolving this issue and chose the option of having Westinghouse re-analyze the MSLB in the valve vault using an updated containment/subcompartment computer code, COMPACT. This code models buoyancy due to steam temperature, which is an important model for the vaults because it accounts for the chimney effect which is physically present in the vaults. The code shows that outside air is pulled into the vault, which produces a significant temperature reduction. By letter dated August 13, 1986, TVA submitted a report, "Main Steamline Break Environmental Qualification Study for TVA Sequoyah Units 1 and 2 Main Steam Valve Vaults."

The mass and energy release data from Westinghouse Topical Report WCAP-10961, Revision 1, were used as input to the Westinghouse computer code COMPACT for calculating the temperature profiles in the valve vaults. TVA then performed a thermal lag analysis to obtain the component temperature response.

<u>Mass and Energy Release Data</u>

The mass and energy release data for Sequoyah are in "Category 2" of WCAP-10961, which was prepared under the auspices of the Westinghouse Owners Group High Energy Line Break/Superheated Blowdown Outside Containment subgroup program.

The Westinghouse computer code LOFTRAN was used for this calculation. The code was modified to account for heat transfer to the steam during steam generator tube bundle uncovery. (This modification is described in WCAP-8822, Supplement 1, which the staff acknowledged as acceptable by letter dated May 27, 1986.)

TVA postulated a spectrum of breaks, including a double-ended 1.4-square-foot rupture of the steam line, a 0.9-square-foot break upstream of the main steam line check valve, and a 0.9-square-foot break downstream of the main steam line check valve. The 1.4-square-foot break results in automatic isolation of the main steam isolation valves (MSIVs) and the most rapid uncovering tube bundle, and, therefore, the earliest onset of superheat. The 0.9-square-foot break upstream of the check valve is similar to the 1.4-square-foot break except that the blowdown rate is lower and the duration of blowdown is longer. Even though automatic isolation of the MSIVs does not occur, the check valve prevents the other three steam generators from blowing down. The 0.9-square-foot break downstream of the check valve does not initiate MSIV closure, and, therefore, all four steam generators blow down. As a result, the tube bundle is uncovered late in the transient. The total blowdown energy from the four steam generators is significantly higher than that from one steam generator. The results of the analyses indicate that the 0.9-square-foot break downstream of the check valve is the limiting case.

Compartment Temperature and Component Thermal Lag Analyses

In calculating compartment temperature profiles using the COMPACT computer code, the buoyancy force due to temperature stratification and the density of the steam are represented by the gravity term in the momentum equation. TVA found that buoyancy initiates a natural circulation pattern that pulls cold outside air into the vault and pushes hot air out through the blowoff roof panel. Natural circulation significantly reduces the temperature in the vault. The natural circulation phenomenon and its effects were originally identified in the COMPACT code calculations and later confirmed by a TVA calculation using the RELAP5 computer code. They were also confirmed by the staff's consultant, Battelle Pacific Northwest Laboratory (PNL), using the COBREE computer code.

In the calculation of the valve vault temperature response, the concrete walls and steel structures were counted as heat sinks. Condensation heat transfer based on the Uchida correlation was modeled until the surface temperature reached the saturation temperature corresponding to the pressure in the vault. Afterwards, natural convective heat transfer was modeled. For the components, different heat transfer coefficients were used to maximize the component surface temperature responses. Four times the Uchida correlation and forced-convection, heat-transfer coefficients were used in modeling the condensing mode and saturation mode, respectively. This approach is conservative and in accordance with the staff guidance in NUREG-0588. It is, therefore, acceptable.

Results of the Analysis

Westinghouse analyzed six cases for the two valve vaults using the COMPACT computer code. The rapid blowdown of the steam generator for the 1.4-square-foot and 0.9-square-foot breaks upstream of the check valve cause natural circulation to occur early in the transient. Therefore, the cooling effect of natural circulation mitigates the temperature rise in the valve vaults. However, the 0.9-square-foot break downstream of the check valve results in all four steam generators blowing down and delays the natural circulation effect. This delay results in a higher vault temperature. The results in the TVA submittal indicate that the 0.9-square-foot break downstream of the check valve in the west valve vault is the worst case. For this case, the vault air temperature rises to 302°F from 140°F in the first 10 seconds after the break. Thereafter, the vault air temperature slowly rises to 323°F by 250 seconds. At that time, the tube bundles start to uncover; the vault temperature increases to 430°F at about 510 seconds, and stays at about 430°F for 70 seconds. At 543 seconds, the mass release rates have dropped enough for natural circulation to begin. Natural circulation and the termination of the blowdown at 600 seconds cause a rapid cooldown of the vault to temperatures below 200°F.

A sensitivity study showed that the results are not sensitive to the nodalization model chosen for the valve vault. A blowoff roof flow area sensitivity study also showed that the compartment air temperature rise is only slightly sensitive to the flow area.

The resulting surface temperature profile for a MSIV is shown in Figure 6.3-5 of the TVA report submitted August 13, 1986. The peak temperature is 365°F. The resulting surface temperature profiles of an ASCO solenoid valve and conduit are shown in Figures 6.3-11 and 6.3-19, respectively, of the TVA report. The peak temperature is about 380°F in both cases. These peak component surface temperatures are higher than the qualification temperature limit of 325°F.

Confirmatory Analyses Performed by TVA and PNL

Westinghouse performed the analyses discussed above for TVA, using the COMPACT computer code. TVA performed an independent, confirmatory analysis using the RELAP5 computer code. The results based on RELAP5 are similar to those obtained using COMPACT with respect to the shape of the temperature profiles and the phenomenon of natural circulation. The predicted timing of the temperature spike and the onset of natural circulation cooldown were in close agreement in the two calculations. The predicted peak temperature and steady-state temperature values also were close, with the RELAP5 results being somewhat higher.

Using RELAP5, TVA analyzed additional cases assuming a smaller break size (0.3 square feet) and different initial power levels (102 percent and 70 percent). The effect of initial power on the vault temperature response was insignificant, and the temperature response for the smaller break size was less severe. Therefore, TVA believed that the spectrum of break sizes chosen in the Westinghouse COMPACT analysis was acceptable. The staff agrees with TVA on the adequacy of the break spectrum analyzed.

At the staff's request, PNL performed an independent confirmatory analysis using the COBREE computer code. (This code has previously been used for the calculation of compartmental pressure/temperature response following a postulated HELB.) The results of the PNL analysis show good agreement with the shape and timing of the temperature profiles obtained for the three cases analyzed in the Westinghouse COMPACT analysis (the 1.4-square-foot break, the 0.9-square-foot break upstream of the check valve, and the 0.9-square-foot break downsteam of the check valve in the west valve vault). The PNL results confirm the effect of the natural circulation phenomenon identified in the TVA analysis. Quantitatively, the COBREE calculations predicted higher room temperatures but lower component surface temperatures. One of the main reasons for this is the way in which the COBREE code models heat transfer. The current version of the COBREE code used the same heat transfer coefficient for strucural heat sinks and safety-related components. The COMPACT code, however, minimizes heat transfer to the structural heat sinks and maximizes the heat transfer to the safety-related components. This approach is more conservative for component surface temperature calculations and is consistent with the guidance in NUREG-0588. Therefore, the staff finds the component surface temperature profiles calculated with the COMPACT code to be acceptable for equipment qualification.

Internal Heat Transfer

TVA analyzed the thermal response of electrical components to the surface temperature profiles to show that the internal temperatures reached during the MSLB are bounded by the internal temperatures from the quantification testing.

This modeling methodology was the subject submittals to the NRC as well as several meetings with the NRC concerning the acceptability of using the methodology for establishing environmental qualification of equipment. A detailed review and technical evaluation of the licensee's submittals on this issue was conducted by Franklin Research Center (FRC) under contract to the NRC. The results of that work were reported in FRC Technical Evaluation Report TER-C5506-658, "Review of Thermal Analysis of Electrical Equipment for Main Steam Line Break Environmental Qualification, Sequoyah Units 1 and 2," dated May 8, 1987. This TER is included as Appendix C to this SER. NRC staff has reviewed the TER and the staff agrees with the conclusions in the FRC TER that there is reasonable assurance that the heat transfer modeling accurately reflects component temperatures during a MSLB. Where assumptions were required during the modeling, TVA maintained a conservative approach, providing additional assurance that the predicted component temperatures during the MSLB approach a worst-case scenario. Therefore, TVA has effectively demonstrated that the components located in the MSVVs identified in Table 1 of the TER would not exceed their qualified temperature profile during a MSLB and are considered qualified for this condition. The staff further concludes that this methodology would be acceptable (with proper application) for demonstrating qualification of equipment which was not included in Table 1 of the TER and was located in the valve vaults.

3.2.2.2 Main Steam Line Break Inside Containment

Westinghouse, on behalf of TVA and Duke Power, modified the LOTIC III computer code to include the cooling effects of the ice melt water spraying out of the ice condenser drains. A test program that included full-scale modeling of the spray out of a drain was undertaken to support the changes to the LOTIC code. A COBRA NC analysis was also performed to provide a very detailed analysis of the containment temperature transient. This work is contained in two topical reports, WCAP-10986 and -10988. These analyses showed that the spray effects of the ice melt water totally offset the energy addition due to superheated steam after tube bundle uncovery. The peak temperature inside Watts Bar containment was reduced from 327°F to 315°F. Duke Power saw similar results for its Catawaba plant.

TVA reviewed the Watts Bar analysis for applicability to Sequoyah and determined that a Sequoyah specific analysis was necessary. This additional analysis was required because of the minor differences between the two plants in structural arrangements inside containment. The analysis used Sequoyah-specific steam line break masses and energy releases. The results of this analysis indicated that the current FSAR steam line break temperature profiles were conservative and additional analysis was not required.

The staff concludes that the containment temperature profile is acceptable contingent on the verification that the analysis contained in the Westinghouse Reports WCAP-10986 and -10988 is accurate. The staff's review of these reports is being conducted on a generic basis and the results of the generic review will be addressed separately.

3.2.2.3 Summary

The staff finds that this issue is resolved on the basis of the NRC staff review of (1) the TVA main steam temperature issue discussion provided in Part III, Volume 2, SNPP Revision 1, March 1987; (2) the FRC TER-C5506-658, May 8, 1987; and (3) the documentation evaluated during the April 6-10, 1987, NRC environmental qualification inspection report 50-327/328 87-22.

3.3 Piece Part Qualification (Procurement)

3.3.1 Introduction

TVA Nuclear Safety Review Staff (NSRS) reports R-84-17-NPS and R-85-07-NPS identified deficiencies in TVA's practices for the procurement of safety-related replacement items. NRC Inspection Report 50-327/328 86-61, dated November 14, 1986, cited related deficiencies which were classified as a potential enforcement item (50-327/328 86-61-01) for failure to take corrective action. Specifically, the TVA program could allow previously qualified equipment to be degraded by purchasing replacement components and parts as commercial-grade, without documentation of its qualification and without adequate dedication of the items by TVA.

While TVA has taken corrective action to improve the procurement program, TVA had no programmatic requirements for the dedication of commercial-grade items and had failed to address the effect that past procurement may have had on the quality of installed equipment.

3.3.2 Evaluation

The staff evaluation of TVA's component and piece part qualification program is based on a review of Section 12.0, Component and Piece Part Qualification," of Part III, "Special Programs," of Volume 2, SNPP, Revision 1, and of an April 1, 1987(b) TVA submittal.

TVA has established the Sequoyah Replacement Items Project (RIP); the three primary goals of this project are to

(1) verify that previously qualified equipment (seismic and environmental) has not been degraded through the use of spare and replacement items

(2) establish programs and practices that will ensure that previously qualified equipment (seismic and environmental) will not be degraded in the future through the use of spare and replacement items

(3) involve the Division of Nuclear Engineering (DNE) in the procurement process as an integral function

The major activities of the RIP project follow.

(1) Before restart TVA will review the plant's maintenance history to identify the activities where safety-related components or items have been replaced.

(2) Before restart TVA will perform an evaluation on previously installed 10 CFR 50.49 (environmentally qualified) replacement items and on seismically sensitive components that are installed within the Phase I DBVP boundary.

(3) Before restart TVA will establish a conditional release program for Quality Level II items. This conditional release program permits these items to be issued and installed before the dedication process for those items is complete. These items will be tracked from the time they are issued through their specific application to ensure future evaluation.

(4) After restart TVA will dedicate commercial-grade material installed or currently in stock for use in safety-related applications.

(5) After restart TVA will evaluate commercial-grade items located in the power stores warehouse. The purpose is to determine what may be released and used for present maintenance.

(6) After restart TVA will perform an engineering evaluation of the other safety-related replacement items.

(7) After restart TVA will develop pre-engineered specifications detailing technical and quality requirements, source audit and inspection requirements, receipt inspection requirements, part conditioning requirements, and, if applicable, post-maintenance test requirements

Through its RIP, TVA will establish a maintenance history of plant replacement activities by reviewing maintenance requests, preventive maintenance activities, surveillance instructions, and work plans. Replacement items are sorted with respect to their application (e.g., 10 CFR 50.49, critical systems, structures, and components). DNE will perform a documented engineering review and evaluation to establish the suitability of replacement items for their intended application.

TVA has revised the Sequoyah site procedures to require dedication of new procurements of commercial-grade items used as basic components. A contract engineering group has been established to provide the technical and quality requirements for new procurements.

The NRC inspection of the RIP is discussed in Inspection Report 87-40. All restart commitments have been completed. An issue was raised regarding the screening process used by TVA for replacement parts in seismically qualified equipment. In some cases, TVA used the historical data base of equipment operating experience in earthquakes to conduct its review of the seismic adequacy of replacement parts. The staff concluded that this was not an acceptable approach for long-term resolution of this issue at Sequoyah as discussed in an October 29, 1987 letter to TVA. However, the staff further concluded that this process could be used to support plant restart. TVA responded to the staff concern by letter dated December 8, 1987(a); TVA provided an acceptable long-term program plan by letter dated February 10, 1988.

3.3.3 Conclusion

On the basis of its review, the staff finds that, with proper implementation of the plans, this special issue should be satisfactorily resolved.

3.4 <u>Sensing Line Issues</u>

3.4.1 Line Slope

3.4.1.1 Introduction

Issues were raised through the employee concerns program concerning the instrument line slope. It was determined that the actual configuration did not match the requirements for line slope indicated on plant drawings at Sequoyah. Erroneous instrument line slope can affect instrument sensor accuracy and lead to an instrument error in detecting process conditions outside the safety limits. Instrument lines act as a coupling between processes and sensors and, to be effective, they must be filled with a known fluid. Insufficient line slope can cause gas to be entrapped with the liquid medium or may cause gas to condense to liquid and cause a degradation in instrument accuracy. Some designs allow the use of high point vents, along the sense line, for venting where the slope cannot be maintained to ensure that no gas is entrained. Some designs also allow the use of condensate collection chambers, for instrument lines where slope cannot be maintained, to collect condensed liquid from the gaseous medium. The employee concerns noted that some instrument lines had either no slope or reverse slope without high point vents.

There appears to be a number of different problems with different solutions. Some instrument lines have insufficient positive slope while others have a negative slope. Some instrument lines, such as those within the auxiliary feedwater system, have been relocated to ensure system functionality, while others in the effluent gas treatment system (EGTS) require the addition of condensate collection chambers. TVA has submitted a report that contains technical details of such observed problems and the the corrective actions it has taken. TVA has submitted this information by letters dated April 2, July 20, December 8, 1987(b) and January 22, 1988. In the letter of December 8, 1987, TVA issued a six-volume report titled "ECTG Slope Closure," Rev. 0, dated October 27, 1987 (RIMS B25 871027015). As a result of this review, TVA has taken the actions listed below.

(1) TVA expanded the identified concern of upward sloping liquid filled lines to also include condensation entrapment in downward sloping gas filled lines.

(2) Based on various calculations (SQN-ISL-002), TVA has developed criteria for determining instrument line walkdowns where process and ambient conditions could cause unacceptable instrument performance for reactor trip, engineered safety features actuation, or accident monitoring functions.

(3) Based on these criteria, TVA physically walked down 57 instruments and 83 instrument lines and measured for instrument line slope. TVA recorded all observed discrepancies on the instrument line slope sketches and each individual discrepancy was evaluated, dispositioned, and verified by a second individual for technical adequacy.

(4) TVA issued Electrical Design Standard DS-E18.3.7 to be used for instrument line slope criteria for future Sequoyah modifications.

(5) TVA conducted a series of tests to determine the velocity of entrapped air as a function of instrument line slope to determine acceptable slope criteria (Norris Lab report WR28-1-85-124-R1).

(6) TVA issued calculations to determine the amount of entrapped air in closed instrument lines under various temperature and pressure conditions in order to permit the sizing of the high point vent reservoir (VENTRES 001 JAN, B 43 870123 901).

(7) TVA issued two design change notices (DCN) to add a number of condensate collection chambers in EGTS (System 65) instrument lines (DCN-X00007 and DCN-X00014).

(8) TVA issued a DCN (DCN-X00004) to revise RHR (System 74) instrument line for slope and to eliminate a number of high point vent valves.

(9) TVA issued a DCN (DCN-X00009) and two ECNs (ECN-7171 and ECN-7172) to revise auxiliary feedwater (system 3) and containment spray (System 72) instrument lines for slope and rotate the pressure switch tap for the auxiliary feedwater system to 120° from top of the suction header.

(10) TVA has revised and issued an instrument maintenance instruction for filling of sealed instrument systems (IMI-118, Rev. 7).

(11) TVA has prepared and issued maintenance instructions (MI) for backfilled instrument lines for various systems (MI 19.1 series).

(12) TVA has prepared and issued surveillance instructions for verification of essential instrument operability (SI-604).

3.4.1.2 Evaluation

TVA prepared a list of all instruments that either detect or mitigate those events in FSAR Chapter 15, the reactor protection system, provide an input to the reactor protection system, or perform engineered safeguard functions. A number of instrument lines were eliminated from physical walkdown on the basis of the criteria listed below.

(1) all instruments mounted by vendors on a vendor supplied package or skid

(2) all instruments where pressure at the root valves remains above 100 psig (based on calculation VENTRES 001 JAN)

(3) instrument lines that are sealed

(4) ambient temperature is low and pressure excursion will not drain the instrument line during an accident condition

(5) all gaseous filled sense lines that are not subject to condensation.

The staff has reviewed these criteria and found them reasonable. Based on these criteria, 57 instruments and 83 sense lines were identified which required the physical walkdown.

The staff has also reviewed the Norris Laboratory test report (WR28-1-85-124.R1) that indicated that entrapped air in instrument lines sloped at 0.125 inch per foot or more have no effect on the static transmission of pressure in liquid filled lines, even though some air may become entrapped in socket weld fittings. However, the dynamic transmission of pressure may cause significant oscillation at the transmitter over a transient period of time. TVA has calculated that an instrument line that tends to be oscillatory during DBA conditions because of entrapped air will exhibit oscillatory behavior during normal operation and testing. Therefore, this provides the opportunity for corrective actions for the instrument lines that tend to be oscillatory as a result of entrapped air.

The Norris Laboratory test results did not address the migration of entrapped air bubbles within horizontal sections or in downward sloping sections following upward sloping portions. However, TVA calculations indicate that air bubble formation is a concern only in instrument lines operating below 100 psig. This analysis also provided the methodology for sizing of a high-point vent reservoir to ensure that the instrument lines remain liquid filled.

TVA has applied these test results and conclusions to the 57 instruments and 83 instruments lines that were physically walked down. Based on this review, the following findings were identified:

(1)	instrument lines that are acceptable met acceptance criteria	12
(2)	instrument lines that are acceptable met acceptance criteria after minor adjustment	4
(3)	instrument lines that did not meet the acceptance criteria but are acceptable, because of the justification provided	47
(4)	instrument lines that require rework before restart	20

For the 20 instrument lines that required rework, DCNs (X00004, X00007, X00009 and X00014) and ECNs (7171 and 7172) were issued. TVA has dispositioned these DCNs and new slope values were recorded on the revised diagram. These 20 instruments covered the wide range of plant systems including auxiliary feedwater, residual heat removal, containment spray, and effluent gas treatment systems. For the instrument lines that did not meet the acceptance criteria, TVA has evaluated each discrepancy individually on the basis of system requirements, response time, accident environments, operating experience, industry experience and Norris test results.

The NRC staff assisted by its consultant, Science Applications International Corporations, has reviewed the information submitted by TVA and has also met with the personnel who performed the walkdown and who were responsible for disposition of the individual findings.

TVA has issued an electrical design standard to be used for instrument line slope criteria in future modifications. TVA also is planning to issue in the near future an instrumentation engineering requirements specification that specifies the design standards and the required QA inspections. The staff has reviewed the new electrical design standard and believes that design standard together with the instrument specification will prevent the future recurrence of the problem.

3.4.1.3 Conclusion

The TVA study has adequately considered the needed accuracy requirements for safety-related instruments and the technical justification contains the rationale for allowances in instrument inaccuracies. Based on its review of test results, analysis, and design standards for instrument line slope, the staff finds the instrument line slope issue is adequately resolved for Sequoyah.

3.4.2 Compression Fittings

Compression fittings from multiple manufactures are in stock at Sequoyah. Many of them are similar in appearance, but not interchangeable in design. Issues arising from the employee concerns special program were that there are mixed fittings and improper installation resulting from lack of training and inadequate quality assurance. Tests were performed at Singleton Materials Engineering Laboratory of various configurations of compression fittings. The report concluded that regardless of different manufacturers or installation techniques, a compression fitting that successfully passes hydrotesting will serve its intended purpose.

TVA has initiated corrective actions that include periodic training for craft personnel and a procedure defining requirements for installation of compression fittings. Sequoyah will also stock and emphasize the use of one type of fitting, except for equipment interfaces with special types of fitting connections. On the basis of its review of Element Report C017304 and the above information, the staff concludes that the concerns regarding compression fittings are resolved.

3.4.3 Teflon Tape

Teflon tape has been used as a sealant in pipe thread fittings at TVA plants. Under high temperature or radiation conditions, the teflon tape may release flourides that would induce stress corrosion cracking of the stainless steel fitting. Although Sequoyah plant procedures prohibit the use of teflon under high temperature/radiation conditions, a concern at Watts Bar led to an inspection at Sequoyah. Two cases not conforming to the procedural requirements were found and repaired. This issue was tracked as Finding A-5 of the Nuclear Manager Review Group findings, Element Report OP30901, and in Section III.9.3 of the SNPP. As discussed in NRC Inspection Report 87-37, actions for plant restart are complete. As a long-term action, corporate guidance on the use of teflon tape and a single-defined tape replacement plan will be issued.

3.5 Welding

3.5.1 Introduction

In Section III.8 of the SNPP, TVA discusses the welding project program to evaluate the adequacy of the TVA welding program for all of the TVA plants and the suitability of welded structures and systems for service. In addition, approximately 30 percent of the safety-related employee concerns pertain to various aspects of the TVA welding program. Of these concerns, 26 pertained specifically to the Sequoyah plant and 119 were judged to be generic, thus may be applicable to the Sequoyah site. TVA efforts to resolve welding issues were directed first at the Sequoyah site.

By letter dated January 17, 1986, TVA formally submitted its program plan to address employee concerns related to welding for staff review. In essence, TVA formulated its program to evaluate the welding program at each TVA nuclear power plant in two separate work phases. The Phase I effort consisted of a review of the written TVA welding program (design documents, policies, and procedures) to ensure that the welding program correctly reflects TVA's licensing commitments and regulatory requirements. The Phase II effort consisted of actual reinspection of selected welds and the inspection results were used to evaluate the implementation of the written welding program. The sampled welds evaluated to determine whether the welds made by TVA in the field meet the applicable code requirements and are adequate for service.

In both phases of the program plan, TVA was to identify and categorize any deficiencies in the existing program, correct the problems, and implement changes to prevent recurrence.

3.5.2 Evaluation

Phase I Program Plan

The Phase I program consisted of the following subtasks:

° review TVA commitments to NRC

° verify that the written program reflects those commitments

° determine that weld-related commitments are reflected in design output

° determine that the programs implemented by the Offices of Construction and Nuclear Operations, as applicable, reflect design output and quality documents

° assemble employee concerns by type and plant

° analyze and evaluate quality indicators that may have impacted on the programs

- issue an adequacy statement regarding written programs to implement/control welding activities

As a result of the evaluation of the Sequoyah related employee concerns, TVA concluded that there were five problem areas of a programmatic nature which are to be addressed. These five areas concerned (1) box anchor design deficiencies (2) Nuclear Operations (NO) programmatic defieiencies regarding compliance with ANSI N45.2.5 where a required inspection was performed by someone other than the QC inspectors, (3) inadequacies in the inservice inspection (ISI) program, (4) a specific case of poor welder performance, and (5) minor implementation deficiencies in the NO welder qualification continuity program. None of these problems involved hardware deficiencies. The most significant recommendation is to stop the practice that allows welders to update their welder performance qualifications by running a bead on plate rather than making a full-penetration weld.

The staff found that TVA's Phase I effort of this program required a review of its requirements and commitments and search for the specific TVA document (e.g., specification, procedure, or instruction) that provided for implementation of these commitments or requirements. However, TVA had so many tiers of documents with overlapping requirements that were produced by different TVA organizations that it made it almost impossible to understand and verify that all of TVA's own requirements were implemented.

For example, in the FSAR TVA stated that structural steel welding would be conducted in accordance with the American Welding Society (AWS) D1.0-69, "Code for Welding Building Construction," or later versions, up to AWS D1.1-Rev. 2-74, "Structural Welding Code." Section 6 of all these codes specifies: "The inspector shall examine the work to make certain that it meets the requirements of Section 3...." The requirements for fit up are specified in Section 3.

The staff recognizes that fit up inspections for fabrications that are not safety related may be waived, but for safety-related fabrications, fit up requirements must be met in these codes to meet Appendix B of 10 CFR 50. If an unacceptable fit up is incorporated in a welded fabrication, the effective weld size may not be adequate for structural integrity. The results of the TVA welding project revealed that fit up inspections were not performed as a quality control function because they had not been incorporated in the drawings. TVA's proposed actions to resolve these problems are addressed in Section 3.5.3 below.

PHASE II Program Plan

The Phase II program consisted of the following subtasks:

- contract with an outside consultant, APTECH Engineering, to assess plant fitness for service

- contract with an outside consultant, Bechtel Power Corporation, to perform independent audits of the welding programs of TVA's Office of Construction and the Office of Nuclear Operations

- evaluate the need for reinspections based on the result of an evaluation of quality indicators

° implement any additional reinspections and deficiency resolutions

The results of the Phase II efforts of TVA's welding program are discussed below.

The APTECH Engineering review consisted of a review of (1) historical records and activities related to the production of welds under Sequoyah's welding and inspection program, (2) preservice and inservice inspection records of welds, and (3) licensee event reports (LER) relating to weld quality. APTECH concluded that (1) the welding program contained the necessary controls to ensure high quality welds, (2) the rate of significant indications detected during the preservice and inservice inspections is low, and (3) no LERs were generated that are related to poor quality field welds. In summary, there is no evidence that the quality of welds at the Sequoyah plant is such that they are not fit for their intended service.

The Bechtel audit concluded that TVA had an effective program related to welding and NDE at the Sequoyah site. However, the auditors noted that some of the program documents were confusing, overlapping, repetitive, and unclear. The Bechtel audit team recommended that the quality control program be centralized to one level of authority for uniformity and consistency.

The Bechtel audit provided an outside evaluation of TVA's approach to meeting its FSAR commitments. The auditors selected the weld joints for the systems selected by TVA and reviewed the documentation. The audit team reviewed each weld document package for the 17 key elements listed below.

° implementation of technical and welding program requirements
° adequacy of design output document (not in terms of technical adequacy)
° initial welding operator qualifications
° maintenance of welding operator qualifications
° renewal of welding operator qualifications
° initial welding inspector qualifications
° maintenance of welding inspector qualifications
° renewal of welding inspector qualifications
° use of appropriate welding porcedures
° use of appropriate inspection procedures
° use of appropriately trained and qualified personnel
° use and control of welding filler materials
° in-process control of welding
° documentation of the above activities
° nonconformance reports and corrective actions
° adequacy of the training programs

The Bechtel audit resulted in one audit finding concerning procedural errors in the use and control of filler materials by the Office of Construction. The effect of the errors (the post weld heat treatment temperature and time were less than specified and yield strength not recorded as specified) was minimal on the hardware produced. The code requirements (FSAR commitments) were met, but this indicated that TVA did not follow its own procedures.

The most significant recommendation made by the Bechtel auditors is that TVA, wherever possible, should centralize the quality assurance program to one level of authority for uniformity and consistency.

The staff found that the APTECH Engineering review of preservice and inservice inspection results did not appear germane to the employee concerns. Because of the attributes visually inspected and because the operating stresses were so small compared with the seismically induced stresses or stresses induced by postulated design events, the staff does not attach any significance to the study except to indicate that defects and deficiencies great enough to have resulted in failure during normal plant operation probably do not exist.

The Bechtel audit of records was performed in Phase II after TVA had reviewed its records. TVA's review and resolutions of discrepancies are reported in the Welding Project Generic Employee Concern Evaluation Reports WP-03-SQN, WP-06-SQN, and WP-07-SQN. Because of this sequence of review, it is understandable that the Bechtel audit did not find any discrepancies of significance.

TVA Welding Reinspection

The Sequoyah Welding Reinspection Plan specified, among other elements, a reinspection of (1) 333 piping welds in 7 systems, (2) 15 welds in spiral welded duct, and (3) 403 joints (1394 welds) in 50 structures.

This reinspection scope was purposely skewed towards areas where less stringent criteria were specified and, thus, fewer QC checks were required and applied during construction. The reasoning behind this approach was that, if there were welding problems, these are the areas where the problems would most likely be reflected in the plant hardware. The results of the TVA reinspection effort are summarized below.

(1) Pipe Welds

Table 3.1 presents the results of TVA's reinspection of piping welds. In terms of components, the rejection rate is about 55 percent (184/333). In terms of deficient weld attributes contained per weld, the rate of deficient welds is about 4 percent (184/4566). Obviously, both numbers are misleading in that the first number tends to magnify the severity of the problems, particularly when one considers that 104 out of 184 are in the arc strike/spatter category. The weld spatter/arc strike indications are superficial indications and should have been reportable, but they should not be a cause for rejection. The superficial arc strikes and spatters should have been removed by light grinding, as required by TVA's internal procedures. The second number (4 percent rejection rate) is also misleading; it tends to obfuscate the fact that these indications are generally indicative of poor quality and should have been detected and properly addressed during construction.

Cracking is an important attribute for inspection and no cracks were found. Five welds required additional surface rework to remove NDE surface indications. Grinding encroached upon the manufacturer's minimum wall thickness in one of these five welds; however, the remaining wall thickness was more than twice the design wall thickness. It should be noted that the paint removing techniques used (rotary wire brushes and flapper wheels) also changed the original inspection surface and presented an altered surface for reinspection. These slightly altered surfaces will provide different reinspection results.

Discrepancies other than those related to size, shape, location, undercut, and contour/transition that were discovered by visual examination were accepted based on NDE results, that is, by magnetic particle or liquid penetrant testing. The engineering evaluations showed that all of the visually detected indications for all attributes were acceptable; i.e., they met the applicable design stress limits.

The reinspection results for piping welds are shown in Table 3.1. Table 3.2 is a rearrangement of the same data in Table 3.1, which was provided by TVA in its August 1, 1986(b) response to a staff request for additional information. The table shows that most of the welds reinspected were made by the Office of Construction (OC), and that the reportable indication rate was significantly higher for OC-made welds.

(2) Structural Welds

The reinspection results of structural welds are summarized in Table 3.3. Table 3.4 is a recompilation of the same data in Table 3.3, as provided in TVA's August 1, 1986(b) response to a staff request for additional information.

The rejection rate on the basis of deficiencies per inch of weld is about 16 percent (1194/7369), even though the components containing these deficiencies are suitable for service by engineering calculations. The rejection rate on the component basis is about 15 percent (211/1394). On deficient attributes

Table 3.1 Piping weld reinspection results

Attribute	No. of Welds Reinspected	No. of Welds Accepted/Rejected		Percent of Welds Accepted/Rejected	
Contour/Transition	333	317	16	95.2	4.8
Offset/Alignment	333	331	2	99.4	0.6
Undercut	333	331	2	99.4	0.6
Reinforcement	333	326	7	97.9	2.1
Weld spatter/ Arc strike	333	229	104	68.8	31.2
Weld Location	333	333	0	100.0	0.0
Weld Size	333	320	13	96.1	3.9
Weld Metal/ Base Metal	333	333	0	100.0	0.0
Weld convexity	333	333	0	100.0	0.0
Incomplete Fusion	333	328	5	98.5	1.5
Weld Overlap	333	325	8	97.6	2.4

Underfilled	333	321	12	96.4	3.6
Surface Porosity	333	318	15	95.5	4.5
Surface Slag	333	333	0	100.0	0.0
Total/Average:	4,662	4,478	184		

Table 3.2 Reportable indication for pipe welds

Type of Weld	No. of Welds Reinspected	No. of Welds With Reportable Indications	No. of Welds Rejected by Code
Socket Welds			
Office of Construction (OC)	204	78	0
Nuclear Operations (NO)	34	6	0
Butt Welds			
OC	68	46	0
NO	22	6	0
Attachment to Pipe Wall			
OC	5	3	0
NO	0	0	0
Total Welds			
OC	277	127	0
NO	56	12	0

Table 3.3 Structural welds reinspection results

Attributes	Inches of Weld Examined	Weld Attribute (Inches) Acceptable	Rejectable	Percent Acceptable	Rejectable
Size	7369	6604	765	89.62	10.38
Incomplete Fusion	7369	7351	18	99.76	0.24
Overlap	7369	7366	3	99.96	0.04
Craters	7369	7362	7	99.91	0.09
Profile	7369	6999	370	94.98	5.02
Undercut	7369	7338	31	99.58	0.42
Correct Filler Metal Type	7369	7369	0	100.00	0.00

Totals:	51,583	50,389	1,194

Table 3.4 Reportable Indications for Structural Welds

Type of Weld	No. of Welds Reinspected	No. of Welds With Reportable Indications	No of Weld Joints* not Meeting Design Requirements
Fillet Welds			
Office of Const.	1080	160	0
Nuclear Ops.	148	21	0
Butt Welds			
Office of Const.	50	4	0
Nuclear Ops.	0	0	0
Other (specify) - Flare			
Office of Const.	92	24	0
Nuclear Ops.	24	2	0
Totals:	1394	211	0

* Weld joints were evaluated, not individual weld segments.

per linear inch basis, the rejection rate is about 2.3 percent. Again, these numbers could be misleading. For welds made by the OC, these rejectable welds should have been detected and disposed of either by analysis or repair during the original construction.

No crack or reportable porosity indications were found. The reinspection results also showed nine missing welds. No underlength welds were identified. The number of reported attributes for size and profile are rather high for the number of welds inspected; however, the engineering evaluations demonstrated that, as constructed, none of the structural welds, including the structures with missing welds, required weld repair.

The staff found that the TVA reinspection effort probably provides the most direct measure of the degree of control exercised by the welding program at the Sequoyah site. The rejection rates cited in TVA's letter of August 1, 1986(b), illustrate a general lack of control or sloppiness during implementation of the welding program in some instances during plant construction. This statement is made on the basis of high rejection rates in piping welds for contour/transition, weld size, underfilling and surface porosity and, in structural welds, for size, undercut, incomplete fusion, and profile. Despite these discrepencies, no weld repairs are required to meet Code requirements.

Employee Concerns

The NCR staff categorized all of the concerns related to welding to identify the issues that may affect the quality of welds at Sequoyah. The first five categories represent elements that the staff believes to be essential for a successful welding program. The categories are listed below.

(1) welding procedures
(2) welder qualification/training
(3) welding inspection and inspector training/qualification
(4) weld design and configuration
(5) filler material control
(6) miscellaneous/one of a kind

Each individual employee concern was assigned to one of these categories. Within each category, the concerns were evaluated as to whether they affected hardware quality or were a programmatic deficiency. The staff review was concentrated on information pertaining to these elements. The information was provided by TVA, as the result of its contractors' programmatic reviews and by its sample reinspections of plant hardware, and by independent inspections conducted by the NRC. The NRC then evaluated this information against either TVA's licensing commitments or industry standards in each of the above six essential elements of an effectively implemented welding program.

There are 41 final element reports of employee concerns primarily involving welding. The staff grouped these reports into five essential element categories that the staff believes are necessary for a welding program and a sixth category, miscellaneous/one of a kind, was created for those concerns which did not fit easily into any of the five essential categories. Each of these essential categories were addressed separately. Of the 145 employee concerns involving welding (specific and generic) applicable to Sequoyah, all except one are addressed in one of these six SERs. The exception, potentially generic concern 2850162005, discussed in TVA's Final Element Report WP-25-SQN, "Effect of Weld Repairs Not Meeting ASME Code," is addressed by the staff in another SER. The conclusions of the staff's SERs are summarized below; these SERs will be discussed in detail in Volume 2, Part 2 of this report.

For the first element, welding procedures, there was only one employee concern expressed for the Sequoyah site which involved a standard fabrication operation with a welding procedure that was not referenced on a particular drawing. The staff team inspections did not find any problems in this area.

For the second element regarding Welder Qualification/Training, there are 27 employee concerns. Most had to do with irregularities in the dating of welder certifications. A welder is required to renew his/her qualification every 90 days, and this may be done by the welder's use of the welding process certiied by his/her employer. The time between taking the test and the handling of a welder's paperwork and actual signing by the responsible authority often gives the appearance of the 90-day requirement being violated, and that backdating or updating occurred. In instances where it may have occurred, the safety significance is rather minimal because the welder's skill would not be that much different between not welding for 90 days versus 100 days. It would be a cause of concern when someone like a foreman who had not done any welding on the job and maintained his qualification by falsification for lengthy periods. However, its safety significance would be rather minimal as long as the individuals in question did not make actual production welds; and there is no evidence, nor employee concerns, to indicate that this was practiced at the Sequoyah site. In addition, the welds would have been inspected and those welds that demonstrated a lack of electrode manipulative skill by the welder would have been rejected. The TVA and NRC reinspections showed that welds with defects indicative of poor electrode manipulative skill by the welders were usually rejected by the original TVA acceptance inspections.

The results of the TVA reinspection, the Bechtel audit, and the staff's independent examinations indicate that the level of workmanship was adequate for the structures and systems involved. No instances of unsatisfactory workmanship significant to the degree that required weld repair were identified. Workmanship type flaws/defects were found, but these were either removed by filing and grinding or an engineering evaluation was performed and the systems or structures were demonstrated to meet applicable code requirements. However, these types of defects/flaws should have been found and disposed of during construction by the QC inspectors under an effectively implemented QA program. The overall quality of welds showed that the welders at the Sequoyah site had the capability to make sound welds and, by definition, were qualified. The impact on the produced plant hardware by welders updating/backdating qualification records was found to be insignificant.

TVA has committed to standardize among all nuclear plant sites the means of maintaining welder qualifications. This will be accomplished by having the QC inspector or the welder foreman initialling the welder's rod issue slip indicating that the welder has maintained qualification by the use of the process.

The third element regarding welding inspection and inspector training/qualification had the largest number of employee concerns (45). The results of the reinspections and audits indicate that the welding inspectors performed their duties in a generally acceptable manner, although they may not have been fully qualified to perform visual inspections. The adherence to code requirements for addressing weld discrepancies should have been more stringently applied. The high rejection rates revealed by the reinspections of welds that were accepted by the original TVA inspections demonstrate that TVA had not performed the original acceptance inspections in accordance with their licensing commitments. As no repairs are necessary to meet the code requirements that TVA had committed to in their licensing application, the significance of these violations is rather slight.

The fourth element, weld design and configuration, had seven employee concerns for the Sequoyah site. Five of the concerns related to a particular box anchor design for piping. These concerns are adequately addressed for the Sequoyah plant because of the special care and drawing changes for these installations. The other two concerns were individually investigated by TVA and the responses are adequate for closeout. Accordingly, the staff does not believe there are major problems under this element.

The fifth element regarding the filler material control had 29 concerns. Many of the concerns related to no portable rod ovens and the lack of material accountability. These issues were adequately addressed by TVA. There were concerns alleging that welders kept unused electrodes and used them later for welding without baking to remove moisture. However, the reinspections should have detected some cracking in weldments if this was a pervasive, common occurrence. The employee concerns regarding the poor quality electrodes were investigated by TVA and the responses are reasonable. The two instances of incorrect electrodes being used were investigated by TVA and the responses are adequate. The results of the reinspections and audits found no signs of inadequate filler material control. Even if there were deficiencies in the filler material control, they did not appear to have impacted the produced hardware.

For the miscellaneous/one-of-a-kind category, there are 35 employee concerns, 27 of which are addressed in WP-19-SQN, "WBN Concerns with No Generic Application to SQN." The TVA Welding Task Group had evaluated all of the employee concerns assigned and had determined, based on further investigations as reported in the various element reports, that these 27 employee concerns were not applicable to Sequoyah. The remaining employee concerns had issues pertaining to unpainted welds, inadequate welding machines, and that the results of the TVA Internal Report QAE-80-2, "Review and Evaluation of the OEDC Welding and NDE Program," were not applied to the Sequoyah site. The uncoated welds are being addressed by TVA under a corrective action report. Although the welding machines might not have all features and aids a welder would like, the machines were adequate to perform the weld when used by a qualified welder. The QAE-80-2 Report was completed after the construction of the Sequoyah plant was completed and, therefore, is not really pertinent.

NRC Team Inspections

Between January 20 and July 11, 1986, the NRC staff conducted three team inspections of TVA's activities related to the welding at the Sequoyah site.

These team inspections have been conducted in accordance with established procedures and with predetermined areas for inspection. The second team inspection, conducted February 18 through 28, 1986, also included independent examinations by the NRC Region I NDE Van, of welds randomly selected by the NRC inspectors. Listed below are the summary results of the NRC inspections.

(1) Inspection Report 50-327/328 86-09

The qualifications of the personnel performing the Bechtel audit, organization, internal procedures, and policies were reviewed and were found satisfactory. The selection process for determining which welds were to be included in the samples and other procedures were reviewed. The sample selection was based on engineering judgment and the availability of records.

The Bechtel audit determined only if the records were present and correct; it did not address the technical suitability of the documents which were audited.

This inspection report also summarizes the staff's review of the TVA Reinspection Program in the areas listed below.

- TVA inspectors qualifications/certifications and nondestructive evaluation procedures
- performance of TVA reinspections
- records of reinspections that TVA had already performed

° possible bias of the sample by determining when the selected items were originally fabricated and comparing them to the level of effort of construction in the past

° distribution of welds reinspected between Units 1 and 2

° TVA's reinspection of at least the minimum number of welds in each group as specified in the Welding Project Program Plan

TVA's reinspection effort identified various weld deficiencies, undersized fillet welds being the major problem. TVA's engineering calculations of these deficient welds found them to be acceptable "as is" and adequate for their intended application. These deficiencies should have been identified during construction and disposed of in accordance with the governing procedures and specifications. However, there are no records to indicate whether or not these deficient welds were identified during construction. Most deficiencies for ASME fabricated pipe welds were of a surface nature, that is, arc strikes and spatters. These too should have been removed during construction by light grinding.

(2) Inspection Report 50-327/328 86-13

To further assess the overall TVA welding program and to evaluate the results of the TVA reinspection effort at Sequoyah, the NRC staff and the NRC NDE van reviewed a sample of the TVA reinspection weld data packages and independently examined a selected number of welds. There were some minor problems in the reinspection weld data packages that required TVA action to resolve. However, no violations or deviations were identified during this inspection of TVA current activities. The staff concluded that the TVA reinspection results were accurate.

(3) Inspection Report 50-327/328 86-33

This inspection report summarizes the NRC team inspections conducted during June 2-6, June 16-20, and July 7-11, 1986, at the Sequoyah site. The NRC welding team reviewed eight followup items that had been identified during previous NRC inspections; the team was able to close seven of those items. The licensee resolved the remaining open item and it was reported as closed in Inspection Reports 50-327/328 86-59 and 50-327/328 87-21.

The NRC staff found the hardware and documentation for the inspected welding activities were generally in accordance with requirements and licensee commitments. The staff noted a number of weld discrepancies, most of which had been identified and evaluated as a result of the TVA reinspection effort. Thus the staff concluded that the current TVA welding project reassessment program was effective in identifying weld deficiencies. However, the staff did identify a number of irregularities, which in most cases related to the accuracy of weld documentation. These irregularities are summarized below.

° The inspection guidance provided in drawings and specifications was confusing for supports of instrumentation, electrical, and heating, ventilating, and air conditioning installations as well as pipe supports. The team could not clearly identify which supports required Quality Level 1 inspection and which required Quality Level 2 inspection. Quality Level 1 inspections require documentation for each weld while Quality Level 2 inspections only require documentation for the completed support.

- A number of welds were found to deviate from the requirements of the applicable design drawings. For instance, the drawing required a full-penetration weld while the hardware was installed using a flare bevel weld.

Section III-3 of TVA's revised SNPP provides an action plan that will improve the design control program for Sequoyah when implemented. This plan includes the reconcilation of "as constructed" and "as designed" drawings to achieve a single set of plant drawings. This plan should address the irregularities identified above to ensure that the welds and welding requirements stated on the "as designed" drawings match the installed hardware.

Expert Consultant Team Evaluation

The NRC staff was assisted by the Brookhaven National Laboratory (BNL) in conducting this review and evaluation. The Technical Evaluation Report (TER) provided by BNL is incorporated as part of this evaluation (Appendix D). The TER evaluates specific employee concerns in more detail and is incorporated as part of this staff safety evaluation. The principal finding of the Expert Consultant Team is that, although there were discrepancies, these discrepancies were not significant or extensive enough to conclude that the plant was not ready or unsafe to start up. Since much of this review was performed in 1986, the staff consultants also reviewed the final element reports on welding late in 1987. However, no new issues were identified that would require resolution before restart.

3.5.3 Conclusions

On the basis of its evaluation, the staff has reached the specific conclusions listed below.

(1) During construction of both Sequoyah units, TVA's implementation of the QA/QC program in the area of welding, while generally effective, was ineffective in certain instances. For example, a significant number of deficient welds were found that required engineering calculations to demonstrate their suitability for service. These calculations should have been performed during construction. In addition, discrepancies between the design drawings and the actual hardware installed were identified. Notwithstanding these findings, the fact that no welds required repair to meet design code requirements indicates an overall effective implementation of the QA/QC program in the area of welding.

(2) The effectiveness of TVA's process for QC inspector training and qualification/certification to visually inspect welds during plant construction and after operation is questionable. The welding deficiencies discussed above should have been detected and corrective actions should have been taken.

(3) In spite of the deficiencies found in the implementation of the QA/QC program for welding activities, including some that were of a programmatic

nature, the staff finds that these deficiencies have not significantly affected the suitability for service of plant hardware.

(4) With the exception of QC inspectors' training and qualification/certification, the staff finds that other essential elements (i.e., welding procedures, welder qualification and training, weld design and configuration, and filler metal control) of a sound welding program were functioning and the resultant hardware is suitable for service.

Therefore, the staff concludes that TVA's welding re-evaluation program has been carried out adequately and that TVA has demonstrated that the hardware as constructed is suitable for service, that is, the design load limits for welded connections have been met. The staff further concludes that restart of both Sequoyah units will not endanger the public health and safety.

For an overall improvement of the welding program at Sequoyah, the staff endorses the following TVA proposed changes in its internal control documents contained in the SNPP:

(1) Combining the requirements of General Construction Document G-29 and Process Specification N73M2 into a single document.

(2) Replacing the general construction specifications for each unit with specific specifications.

(3) Maintaining indirect quality control of fit up inspection by monitoring processes as provided in 10 CFR 50, Appendix B (1) by having the welder and his foreman document that fit up is suitable for the QC inspector to verify weld size during final inspection and (2) by having the QC inspector selectively inspect a sample of fit ups to verify this documentation.

(4) Consolidate inspector training and certification into one program under the control of a certified Level III NDE examiner.

(5) Provide training or orientation to engineers, designers, technical supervisors, and engineering managers on the content and use of the internal control documents.

(6) Standardize the process of maintaining welder's certification by having the QC inspector or welder foreman initial the rod issue slip indicating that the specific welder has used the process.

In a letter dated January 30, 1987, TVA committed to an augmented and accelerated inservice inspection as recommended by NRC staff. The inspection program will include the elements listed below.

(1) A 100-percent examination of the ASME Class 1 and 2 piping field welds will be completed in the first 10-year in-service interval. Those welds that remain to be examined will be scheduled for examination in the next two consecutive refueling outages following the submittal of the revised plan and the restart of any unit.

(2) A 100-percent examination of the ASME Class 1 and 2 pipe support field welds will be completed in the first 10-year in-service interval. Those

welds that remain to be examined will be scheduled for examination in the next two consecutive refueling outages following the submittal of the revised plan and the restart of any unit.

(3) Major component support welds made in the field on the reactor vessel, steam generator, pressurizer, and reactor coolant pumps that have been identified to be examined in the first 10-year program will be completed. Those welds that remain to be examined will be scheduled for examination in the next two consecutive refueling outages following the submittal of the revised program and the restart of any unit.

(4) Where possible, the percentage of welds examined during the program will be maintained as required by the code in the Tables IWB-2412-1 and IWC-2412-1 (Inspection Program B). Note that the required percentages may not be met for all categories of specific systems, or item numbers, because certain systems contain a large number of socket welds that are field welds and the majority of pipe support welds are also field welds. Where conflicts arise with the percentage requirements, the revised augmented/accelerated program will identify specific requirements for relief.

Credit for program examination will be taken for all examinations performed and no additional Class 1 and 2 field welds will have to be re-examined in the remaining time of the first 10-year interval, with the exception of the Code required additional examinations resulting from unacceptable indications in the initial or required successive examinations. Future 10-year interval examinations will follow their original schedule and will not be required to meet the accelerated program.

Because the first refueling outage is scheduled to occur approximately 4 to 6 months after restart of Unit 2, the short duration of the operating time may not provide the needed time for the increased planning and scheduling, staffing and craft support required to perform the increased inspections of items 1, 2, and 3 above. In this case, the implementation of any accelerated program would be deferred to the second and third outages following restart of Unit 2. Scheduling parts of the actual inservice inspection for Unit 2 for the second and third refueling outage after restart rather than the first and second refueling outage after restart is acceptable to staff.

Further, the staff recommends that TVA consider the following:

(1) using industry-generated standards where possible, particularly using American Welding Society (AWS) standards for certifying the AWS scope weld inspectors

(2) amending relevant FSAR sections to reflect changes in commitments and to formalize the intent as stated above

(3) training personnel in the application of the standards adopted

3.6 Containment Isolation

3.6.1 Containment Isolation System Design

3.6.1.1 Introduction

General Design Criteria (GDC) 54 through 57 of Appendix A to 10 CFR 50 contain NRC design requirements for isolation of piping systems penetrating containment. In particular, GDC 54 contains general provisions for leak detection, redundancy, and reliability. GDC 55 requires each line that is part of the reactor coolant pressure boundary (RCPB) and that penetrates the containment to have isolation valves as listed below, unless it can be demonstrated that the provisions for a specific class of lines are acceptable on some other defined basis.

(1) one locked closed valve inside and one locked closed valve outside

(2) one automatic valve inside and one locked closed valve outside

(3) one locked closed valve inside and one automatic valve outside

(4) one automatic valve inside and one automatic valve outside

A simple check valve may not be used as the automatic valve outside containment. GDC 56 contains similar provisions for lines that connect directly to containment atmosphere and that penetrate containment. GDC 57 addresses systems that penetrate containment but that do not communicate with either the RCPB or containment atmosphere and requires at least one valve (not a simple check valve).

The rationale for allowing a demonstration of acceptability on "some other defined basis" (i.e., a deviation from the explicit requirements of the GDC) is that in certain instances (e.g., lines in essential systems that are required to operate following an accident) compliance with the explicit requirements of the GDC would be detrimental to safety.

Isolation designs which are adequate on "some other defined basis" are described in the Standard Review Plan (SRP) Section 6.2.4, "Containment Isolation System," and American Nuclear Standards Institute (ANSI) Standard N271-1976, "Containment Isolation Provisions for Fluid Systems." For containment spray line penetrations, as well as for other essential systems, the SRP and the ANSI standard identify the use of remote manual valves in lieu of automatic valves as acceptable. TVA, on the other hand, has traditionally relied on the closed system outside containment rather than identify an outboard remote manual valve as an isolation valve.

This was considered by TVA to be an acceptable isolation design on another defined basis. The staff SER for the SQN license, NUREG-0011, Section 6.2.4, issued March 1979, concluded that the design of the containment isolation system was acceptable, but did not specifically address the acceptability of "other defined basis" for any containment isolation figure. The present staff position, particularly following development of the TMI Action Plan, is that a closed system outside containment is not generally acceptable as an isolation barrier for lines covered by GDC 55 or 56.

The staff identified apparent discrepancies in system compliance with containment isolation requirements during an inspection conducted at Sequoyah on March 3-14, 1986. Specifically, Inspection Report 50-327/328 86-20 documents five containment penetrations of the chemical and volume control system (CVCS) that did not appear to meet 10 CFR 50, Appendix A GDC for containment isolation. The penetrations cited in the inspection report are penetration X-16, the normal charging supply, and penetrations X-43A, -43B, -43C, and -43D, the four reactor coolant pump seal injection lines.

The staff requested TVA to provide its position on the design bases for the isolation system, as well as a complete description of the isolation provisions for all penetrations that do not meet the explicit requirements* of GDC 55, 56, and 57. TVA by letter dated May 30, 1986, provided a complete evaluation of containment penetration isolation provisions against the licensing requirements of GDC 55, 56, and 57. On the basis of this information, the staff concluded that, in addition to the five CVCS penetrations, there were numerous penetrations whose isolation provisions as described in the FSAR were in noncompliance with the explicit requirements of the applicable GDC.

TVA and the staff discussed the particular isolation capabilities for the five CVCS penetrations, the designated isolation design and the isolation capability for numerous essential system lines, and the isolation design logic in general. The staff advised TVA that while the designated isolation design for a number of penetrations in essential systems was unacceptable, adequate isolation capability existed in the form of existing remote manual valves that had not been identified as isolation valves. Therefore, in most instances involving isolation of essential systems, the isolation design could be made acceptable per the GDC by designating certain available valves and subjecting them to the operability, surveillance, and testing requirements appropriate for isolation valves. As part of these discussions with the staff, TVA agreed to re-evaluate the isolation capability for all penetrations, identify and describe those penetrations whose isolation provisions complied with the explicit criteria of the GDC, and identify and describe those penetrations that satisfy the GDC on "some other defined basis." Furthermore, TVA agreed, where applicable, to designate certain available valves as containment isolation valves, subject to appropriate operability, surveillance, and testing requirements, to comply with the GDC.

By letter dated September 24, 1986, TVA provided information reflecting agreements reached between TVA and the NRC on August 13, 1986, and in particular, discussion of the original design provisions, responses to NRC questions, and re-evaluation of the isolation provisions for the five CVCS penetrations and for additional specific penetrations identified by the staff. During the course of reviewing this submittal, the staff identified a number of items requiring additional information or clarification. By letter dated January 2, 1987, TVA provided additional information clarifying several issues, including

*"Explicit requirements" refer to the specific containment isolation valve arrangements listed in the GDC without need for a demonstration of acceptability on "some other defined basis" as allowed by GDC 55 and 56.

(1) evaluation of the isolation system regarding design criteria specifications for seismic Category I, Quality Group B, and protection from missiles and pipe

whip; (2) administrative controls over certain local manual valves; (3) position indication for motor- or air-operated isolation valves; and (4) leak detection capability to allow the operator to identify and isolate essential systems that have become leak paths. These items are discussed later in this section.

Additionally, during the process of reviewing the Sequoyah containment isolation system design, the staff determined that, although in most instances the system met the GDC or could be acceptably modified by designating additional existing valves as containment isolation valves to satisfy the GDC, there were eight penetrations whose isolation provisions, even after modification by designation of additional existing isolation valves, would not satisfy the GDC. More significant design modifications would be necessary to bring the isolation design for the subject penetrations into compliance with the appropriate GDC. The eight penetrations involve the four reactor coolant pump seal injection lines, the reactor heat removal (RHR) discharge line, and the three containment vacuum relief lines. In response to the staff determination, TVA accordingly submitted, by letters dated January 23 and February 3, 1987, requests for exemption to the requirements of 10 CFR 50 GDC 55 and 56 for the penetrations in question. Supplemental information to these requests was submitted by TVA on April 8, 1987.

In the evaluation below, the staff discusses each penetration not meeting the explicit GDC requirements as identified by TVA in Table 2.2 of its submittal of January 2, 1987.

3.6.1.2 Evaluation

Reactor Coolant Pump Seal Injection Lines (Penetrations X-43A through X-43D)

The containment isolation for these lines provided a check valve inside containment and a closed system outside containment. GDC 55, which applies to penetrations that serve as part of the RCPB, is the applicable criterion for these penetrations. GDC 55 requires either automatic or locked closed isolation valves, one inside and one outside containment. However, as discussed earlier, the GDC allow for a demonstration of acceptability on "some other defined basis," principally in order to avoid situations in which compliance with the GDC is counterproductive to overall safety. For certain transients and accidents, it is desirable that the reactor coolant pump seal injection lines remain in service to protect the reactor coolant pump seals; thus these lines are not automatically isolated or locked closed.

It is acceptable and common practice, therefore, to satisfy the requirements of GDC 55 on "some other defined basis" for the reactor coolant pump seal injection lines by providing a remote manual containment isolation valve outside containment, in addition to a check valve inside containment. However, the Sequoyah design is of an early vintage and remote manual valves are not installed in those lines. Since the staff indicated that the originally designated isolation design for these penetrations did not satisfy GDC 55 explicitly and was not acceptable on "some other defined basis," TVA re-evaluated the options for improving the isolation design. As a result of its evaluation, TVA selected the local manual globe valves in the seal injection line header as the outboard containment isolation valves. After an accident, the globe valve at the seal water filter outlet is accessible from the standpoint of dose assessment.

As a related issue, the staff requested TVA to address the matter of leak detection for the seal injection lines because local manual isolation imposes an additional burden in post-accident management. The reactor coolant pump seal injection flow is provided by the centrifugal charging pumps. A leak in either pump room can be associated with the pump involved and action taken to isolate the affected equipment. From the pump room the seal injection line is generally routed through common pipe chases. However, the leak detection system does not provide detection for the lines running through a common pipe chase. Leak detection for the seal injection lines basically consists of flood detection, which provides non-specific indication of leakage from a variety of sources. Isolation of leaks will be accomplished by arbitrarily selecting and isolating subsystems and evaluating the response of the flood detector system. In the event a leak in the seal injection filter valve packing should occur, drains in the cubicles carry spillage to the tritiated drain collector tank. The drains are sized to accommodate a maximum leak rate of 50 gpm, corresponding to the leak rate estimated for failure of a reactor heat removal (RHR) pump shaft seal. Valve packing leaks should be substantially smaller; therefore, the drains would accommodate valve packing leakage, thus allowing access to the cubicles housing the seal injection line filter valves.

By designating the local manual globe valves in the seal injection line header as containment isolation valves, TVA has provided a design that satisfies the redundancy requirements of GDC 55 in that an inboard and outboard valve are included. However, reliance on the local manual valve does not satisfy the valve actuation requirements of GDC 55, nor does it meet the criteria (as outlined in the SRP Section 6.2.4 or ANS Standard N271-1976) to satisfy the GDC on some "other defined basis." The use of local manual valves in lieu of power-operated valves with remote manual action is a degradation of design criteria that, in this instance, precludes compliance with the GDC.

After being apprised of the staff position on this matter, TVA requested an exemption from the requirements of GDC 55 for the four reactor coolant pump seal injection lines. TVA has noted that in addition to the inboard check valves and the outboard local manual valves, there are other isolation barriers that provide additional protection against leakage to the environs from these penetrations. First, each of the seal injection lines has another check valve inside containment, albeit located inside the missile barrier and therefore not considered missile protected. Secondly, the system outside containment is a closed system designed to seismic Category 1 standards and meets at least Safety Class 2 design requirements. Furthermore, these lines are normally in service under normal, transient, and accident conditions, with at least one centrifugal charging pump providing a water seal at a pressure sufficient to preclude containment atmosphere leakage. The piping is leak tested by visual inspection relative to NUREG-0737, Position III.D.1.1, and is included in the ASME Section XI inservice pressure test program.

The staff concluded that the proposed containment isolation provisions for the seal water injection lines, with the newly designated containment isolation valves, are adequate and that an exemption from the requirements of GDC 55 with respect to valve type could be granted for those reactor coolant pump seal injection lines. The exemption was issued on December 4, 1987.

Charging (Penetration X-16)

TVA stated in the FSAR that the containment isolation design for this line consisted of a check valve inside containment and a closed, seismically qualified, safety class system outside containment. The use of a check valve inside containment and a closed system outside containment is not acceptable for meeting staff guidelines with respect to the requirements of GDC 55. Therefore, the staff requested TVA to identify an outboard containment isolation valve. TVA identified the available outboard automatic isolation valve closest to the containment as the outboard containment isolation barrier. This valve automatically closes on a safety injection signal and was provided in the original design. Its new designation as a containment isolation valve, subject to appropriate operability, surveillance, and testing requirements, renders the isolation design for this penetration acceptable and in compliance with the explicit requirements of GDC 55.

Emergency Core Cooling System (ECCS) Injection Lines (Penetrations X-20A, -20B, -21, -22, -32, -33, -108, 109)

The containment isolation provisions for the ECCS injection lines as described in the FSAR, consist of a check valve inside containment and a closed, seismically qualified, safety class system outside containment. In accordance with the staff's request, TVA has identified (submittal of January 2, 1987) additional outboard remote manual valves for these penetrations and has designated those valves as containment isolation valves, subject to the operability, surveillance, and testing requirements associated with containment isolation valves. These newly designated containment isolation valves were provided in the original design but were not identified as containment isolation valves. The use of remote manual valves, in lieu of automatic valves, in conjunction with a closed system is acceptable for meeting the requirements of GDC 55 on another defined basis, for essential safety systems which are intended to operate following an accident.

RHR Discharge (Penetration X-17)

The containment isolation provisions for the RHR discharge line (penetration X-17), as described in the FSAR, are identical to that for other ECCS lines (i.e., it utilizes a check valve inside containment and a closed system outside containment). However, when the staff requested TVA to identify an outboard isolation valve, TVA responded that there was no suitable outboard remote manual isolation valve because the Sequoyah design called for the motor-operated (remote manual for containment isolation) valve for this system to be located inside containment upstream of the check valve. Thus TVA has proposed to designate the inboard remote manual valve as a containment isolation valve, subject to appropriate operability, surveillance and testing requirements. This will satisfy the redundancy requirements of GDC 55. While the proposed designation of the additional motor-operated valve as a containment isolation valve is acceptable and necessary, this modification to the design does not bring the isolation design into compliance with the requirements of GDC 55 concerning valve location.

TVA has designated the remote manual valve in the RHR discharge line to the loop 1 and 3 hot legs as a containment isolation valve. This line has multiple isolation provisions: a remote manual valve and two missile-protected check valves inside containment and a closed system outside containment.

The staff concluded that the containment isolation provisions for the RHR discharge line are acceptable. The exemption for valve location was issued on December 14, 1987(a).

Relief Valve Discharge (Penetration X-24)

The containment isolation provisions for the relief valve discharge line (discharging to the pressurizer relief tank), as described in the FSAR, consist of a check valve inside containment and a closed system outside containment. Again, TVA evaluated the system configuration to identify a second isolation valve and concluded it was appropriate to identify the three parallel relief valves outside containment as the outer isolation barrier. The staff found it acceptable to use relief valves outside containment as isolation valves in this instance because containment pressure is applied opposite to the direction the valves relieve and acts as a closing force on the valve. Therefore, the staff concluded that the designation of the outboard relief valves as isolation valves, in conjunction with the closed system outside containment, renders the isolation design acceptable for meeting the requirements of GDC 55 on another defined basis.

Component Cooling Water Supply and Return to Excess Letdown Heat Exchanger (Penetrations X-035 and X-053

These lines are subject to the requirements of GDC 57, and isolation is provided by a closed system inside containment and an automatic valve outside containment. A relief valve is provided on the system inside containment. Since the containment pressure would act in the direction opposite to that in which the valve relieves, the staff found this acceptable.

Chemical and Volume Control System Letdown (Penetration X-015)

The CVCS letdown meets the requirements of GDC 55 with automatic isolation valves inside and outside containment. One of the inboard valves is a pressure relief valve, which relieves to the pressurizer relief tank inside containment. However, because containment pressure would act opposite the direction that the valve relieves, thereby acting as a closing force, the staff considered this configuration acceptable.

Residual Heat Removal Suction (Penetration X-107)

The suction line from the loop hot leg to the RHR pumps is isolated by two motor-operated valves in series, which are closed with power removed while the plant is at power. The valves are interlocked to prevent opening when the reactor coolant system (RCS) is at high pressure. Both valves are located inside containment. The staff considered this configuration acceptable on another defined basis in accordance with ANSI Standard N271-1976.

The relief valve inside containment that discharges to the pressurizer relief tank inside containment is also acceptable per the ANSI standard.

Containment Spray and RHR Spray Lines (Penetrations X-48A, B, X-49A, B)

TVA has indicated in the FSAR that the isolation design consists of a check valve inside containment and a qualified, closed system outside containment. GDC 56 is the applicable criterion for these penetrations because these lines communicate with the containment atmosphere. Since certain penetrations, including the containment spray and RHR spray, are part of systems required to operate following an accident, it is imprudent to follow the explicit requirements of GDC 56 and automatically isolate or lock closed the isolation valves. In those instances where post-accident operation is required, remote manual valves are acceptable for meeting the GDC as described by SRP Section 6.2.4 and the ANSI standard. For the containment spray and RHR spray line penetrations, TVA has identified additional outboard valves that have remote manual closure capability as containment isolation valves. The designation of those valves as containment isolation valves brings the isolation design for these penetrations into compliance with the staff guidelines for meeting GDC 56 contained in the SRP.

Vacuum Relief Lines (Penetrations X-111, 112, 113)

TVA states in its FSAR that the containment isolation design for the vacuum relief penetrations consists of a single automatic isolation valve located outside containment. However, the FSAR also identifies spring-loaded vacuum relief (check) valves in series with the containment isolation valves. By its letter of January 2, 1987, TVA has identified redundant isolation valves for these penetrations, including the air-operated automatic isolation valve and the spring-loaded check valve, both located outside containment. Thus, while TVA has provided a design that complies with the requirements of GDC 56 in terms of the number of valves, the staff found that there is a deviation from the explicit GDC requirements with regard to valve location. TVA, therefore, requested an exemption from the requirements of GDC 56 for the isolation provisions on the containment vacuum relief lines. Specifically, an exemption is required from the requirements of GDC 56 regarding valve location: the isolation design satisfies the redundancy and valve actuation requirements.

With regard to the adequacy of isolation, the staff concluded that with both the spring-loaded check valves and the automatic butterfly valves cited as containment isolation valves, the design is adequate for assuring containment isolation. Another consideration is the fact that the first outer isolation valve, the automatic butterfly valve, is bolted directly to the containment penetration sleeve. The penetration sleeve between primary containment and the butterfly valve has been evaluated by TVA to demonstrate that stresses in the penetration sleeves are well below allowable values in accordance with Branch Technical Position MEB 3-1. Therefore, the staff found that an exemption to the requirements of GDC 56 in the case of the containment vacuum relief lines was justified. An exemption for valve location was issued on December 14, 1987(b).

Another related issue for the containment vacuum relief line isolation design that was considered by the staff in this re-evaluation was the failed position of the isolation valves, specifically the butterfly valves.

The butterfly valves in the vacuum relief lines are normally open valves that are designed to fail-open. This design feature was chosen because the valve-open position has been evaluated as providing for the greatest safety for the plant. In the event of an inadvertent actuation of containment sprays or air return fan operation, a failure of the vacuum relief system to perform its intended task could result in the collapse of the containment. Since the valves are normally open, each of the three butterfly valves in the vacuum relief system is provided with two solenoid actuators powered from redundant air supplies. Thus, a single failure will not prevent closure of the valve, if needed, except if a mechanical failure occurred in the butterfly valve itself. Both the butterfly valve and the check valve have position indication in the main control room. The staff concludes that for Sequoyah, due in part to its low capability to sustain reverse differential pressures, the fail-open position of the butterfly isolation valves is acceptable.

Blind Flanges (Penetrations X-003, -040D, -054, -079A, -079B, -088, -117, -118)

The containment isolation design for the hydrogen purge line penetration consists of a blind flange equipped with double O-ring seals. The flange is located outside containment in the auxiliary building. The staff originally expressed concern over this design because it significantly deviates from the requirements of GDC 56. TVA responded that there was no intent to use this penetration following an accident; post-accident hydrogen control is accomplished by redundant safety-grade recombiners or, in the case of degraded core accidents, by the hydrogen igniter system. Therefore, this penetration is inactive and is prevented by technical specifications from being opened except during cold shutdown or refueling modes of operation. Under these circumstances, the staff concludes that the isolation design is acceptable.

Several other penetrations also are equipped with blind flanges, including those for shutdown maintenance access, ice blowing and layup water treatment. These penetrations are only used in Modes 5 and 6; therefore, the staff finds this acceptable.

Spare Penetrations (X-008, -018, -028, -031, -036, -037, -038, -039C, -039D -040C, -055, -084B, 084C, -084D -085C, 085D, 086D, 087A, -087C, -089, -096A -096B, 105, -116B, -116C, -116D, -119, -120, -125E, -130E,-135E)

Spare penetrations are seal-welded and thus are part of the passive barrier of the containment structure itself. The staff finds this acceptable.

Equipment Hatch (Penetration X-001)

The hatch is provided with a double O-ring seal as its isolation barrier. The staff finds this acceptable.

Personnel Airlocks (Penetrations X-002A, 002B)

The two airlock doors each have double resilient seals and a mechanical interlock to prevent both doors from being opened at the same time. The staff finds that this design provides acceptable isolation for airlocks.

Main Steam (Penetrations X-013A, B, C, D)

The main steam system piping is subject to the requirements of GDC 57. The safety relief valves form part of the outside containment barrier. The set point for the valves is greater than 1.5 times the post-accident containment pressure; therefore, the staff finds these valves are acceptable as isolation valves in accordance with SRP Section 6.2.4.

Sump Supply to ECCS (Penetrations X-109A, 019B)

For the lines from the sump to the RHR pumps, a single remote-manual valve (outside containment) and a closed system outside containment provide the isolation barriers. This system has an essential post-accident function and its reliability could be adversely affected by the presence of additional or automatic isolation valves. In accordance with SRP 6.2.4, the staff finds this configuration is acceptable on another defined basis for conformance with GDC 56.

Hydrogen Analyzer (Penetrations X-092A, B, -099, X-100)

TVA has modified the hydrogen analyzer penetrations to provide fail-closed solenoid-operated valves inside containment and solenoid-operated valves outside containment. This satisfies the requirements of GDC 56, and is therefore acceptable.

Delta P Sensor (Penetrations X-025B, -026A, -027A, -27B, 85B)

These sensor lines provide containment pressure inputs to instrumentation. They are missile-protected and designed for accident conditions. Isolation is provided by redundant bellows. Considering the safety function of these lines, the staff finds the isolation provisions are acceptable on another defined basis in accordance with ANSI N271-1976.

Reactor Vessel Water Level Instrumentation (Penetrations X-025C, -26C, -27D, -86A, -86B,-86C

These sensing lines provide indication of reactor vessel water level and are required to function after an accident. The lines are armored, filled with water and sealed. No valves are provided since they could interfere with performance of the system. The sensor inside containment is sealed, and outside containment, a bellows device provides isolation. The staff considers the isolation configuration acceptable for these lines based on the guidance of the ANSI standard.

Electrical Penetrations (X-20E to X-170E)

Electrical penetrations are not subject to the valving requirements of GDC 56. However, the isolation barriers are provided by the epoxy-sealed penetration assemblies. The staff finds that this provides adequate isolation for these penetrations.

Other Issues

As stated previously, as part of a general reevaluation of the Sequoyah containment isolation design prompted by the NRC team inspection, the staff, in addition to GDC requirements, also evaluated other issues related to containment isolation. First, since the containment isolation system is part of the engineered safety feature network in that it serves a vital role in reducing offsite releases, the staff requires the isolation system meet the usual criteria for an engineered safety feature system. In that regard, TVA has confirmed that all containment isolation valves including newly designated containment isolation valves and all associated piping meet the standards of ASME Section III Class 2 and are seismic Category I or the equivalent of those standards. Second, the staff normally requires that all power-operated containment isolation valves have position indication in the main control room. TVA recently confirmed that with the exception of 22 valves, all other power-operated valves have position indication in the main control room. Position indication for the 22 exceptions are provided in either the auxiliary building or the hot sample room. Installation of position indication for the 22 containment isolation valves in the main control room is planned for the cycle 4 refueling outage.

Since the local manual globe valves in the seal water filter outlet lines and filter bypass lines in the reactor coolant pump seal injection system provide the function normally provided by remote manual, power-operated isolation valves, the staff has questioned the provisions for position indication of those valves. TVA has responded that while those manual valves do not have position indication in the conventional sense of power-operated valves, the valve position is recorded in the plant configuration log that is kept in the main control room. The staff concludes that by this method the licensee provides position indication in an appropriate and acceptable manner.

3.6.1.3 Conclusions

On the basis of its evaluation, the staff concludes that with the approved exemptions, the containment isolation design is in accordance with Appendix A to 10 CFR 50; therefore it is acceptable.

3.6.2 Containment Leakage Testing Program

3.6.2.1 Introduction

As discussed above, Inspection Report 50-327/328 86-20 contained open items regarding the containment isolation design for certain containment penetrations. By letter dated September 24, 1986, and January 2, 1987, TVA proposed to partly resolve these open items by redesignating certain valves as containment isolation valves. The acceptability of these proposals is addressed above. TVA also has evaluated the redesignated containment isolation valves in regard to the requirements of Appendix J to 10 CFR 50 concerning local leakage rate testing. The staff's review of this issue follows.

3.6.2.2 Evaluation

Reactor Coolant Pump Seal Water Injection Lines (Penetrations X-43A, -43B, -43C, and -43D) and Normal Charging Line (Penetration X-16)

TVA states that the valves in these penetrations will be sealed with water during an accident by ECCS pumps at pressures greater than 1.1 Pa and with at least a 30-day supply of water, even considering a single active failure. TVA has concluded that these valves are not subject to Type C (local leakage rate) testing.

Based on the above description of the system operation, the staff agrees with TVA that if these penetrations and associated containment isolation valves are closed to perform their containment isolation function, they will be sealed with water via the ECCS pumps with a continuous supply of sealing water from the containment sumps. In accordance with paragraph III.C.3 of Appendix J to 10 CFR 50, because the containment isolation valves of these penetrations will be maintained under a water seal for at least 30 days following the onset of an accident, they are not potential containment atmosphere leak paths; therefore, they do not require a Type C test with air or nitrogen. In addition, a water leakage rate test is not needed since a continuous supply of sealing water is provided from the containment sump.

Emergency Core Cooling System Lines (Penetrations X-22, -33, -32, -21, -20A, -20B, -17, -108, and -109)

For the high-head and intermediate-head safety injection pumps (penetrations X-22, -33, -32, and -21), TVA states that a water seal is provided during an accident at pressures greater than 1.1 Pa and with a continuous supply of water, even with consideration of a single active failure. Therefore, the staff finds, by the same reasoning as stated in the last paragraph above, the valves in these penetrations are not subject to Type C testing.

For the injection line penetrations (X-17, -20A, and -20B) for the low-head safety injection pumps (RHR pumps), a water seal cannot be guaranteed with a single active failure of an RHR pump. Any leakage past the two in-series leak-tested check valves in each line would be into a seismically qualified closed system; testing is performed to demonstrate integrity of the piping. TVA requested an exemption to the Type C test requirements of Appendix J for these lines. An exemption was issued on January 15, 1988.

For the upper-head injection (UHI) lines (penetrations X-108 and -109), a limited supply of water would be available for a water seal during an accident. The water seal is maintained by the water and nitrogen overpressure in the UHI accumulator. If this pressure should be lost, any leakage would be contained in a closed system. Two valves in a test line will be Type C tested with pressure applied in the opposite direction of containment pressure. TVA requested an exemption to the specific provisions of Appendix J for these lines. An exemption was issued on January 15, 1988.

Containment Spray and RHR Spray Lines (Penetrations X-48A, -48B, -49A, and -49B)

The containment spray lines (penetrations X-48A and -48B) are considered by the staff to be water sealed and not potential containment atmosphere leak paths.

A water leg is maintained during normal operation in each riser between a closed valve and the spray ring header. These closed valves now are leakage rate tested with water to verify that there is sufficient inventory in the risers to maintain a water seal for 30 days, even after the containment spray pumps are shut off; this testing is specifically required by Technical Specification 4.6.1.2.g. Therefore, the staff concludes that the present testing of penetrations X-48A and -48B is acceptable.

The RHR spray lines (penetrations X-49A and -49B) are very similar to the containment spray lines, except that no leakage rate testing is performed. The staff would find it acceptable if TVA performed the same type testing as it does for the containment spray lines, or normal Type C testing with air or nitrogen. By letter dated January 2, 1987, TVA has proposed to test the RHR spray valves in the same manner as for the containment spray lines. Thus, the staff finds this is acceptable.

Relief Valve Discharge to PRT (Penetration X-24)

TVA states that all of the redesignated containment isolation valves (which are relief valves) for this penetration are located in closed systems outside containment. These are pressurized after an accident and, therefore, the valves are not subject to Type C testing. These valves are connected to the safety injection system, CVCS, and containment spray system. The staff raised the issue of whether the seals would be maintained with a single active failure. TVA noted that installation of block valves to permit Type C testing would conflict with requirements of the ASME Code for relief valves. Therefore, TVA requested an exemption to Appendix J for this penetration. An exemption was issued on January 15, 1988.

Hydrogen Purge (Penetration X-40D) and Containment Vacuum Relief (Penetrations X-111, -112, and -113)

TVA is not proposing to redesignate any valves as containment isolation valves in these penetrations, nor to otherwise change the isolation provisions for these penetrations. These penetrations presently undergo appropriate local leakage rate testing (Type B or Type C testing) for their current containment isolation barriers, in accordance with Appendix J. Therefore, the staff finds the local leakage rate testing of these penetrations acceptable.

Summary

The staff finds that with the above exemptions, the proposed local leakage rate testing (Type B and C) program for penetrations is in accordance with the requirements of Appendix J to 10 CFR 50, and is therefore acceptable.

3.7 Containment Coatings

The deficiencies found during a TVA review of maintenance records relating to TVA's programs for coatings inside containment are listed below.

- Vendor-coated items had been installed inside containment without being accounted for in the coatings analysis.
- Some inorganic zinc primer was improperly applied and random delamination occurred.
- Coatings were not subject to periodic inspection and a maintenance program.
- Assumptions were not verified for the calculations that established the amounts of coatings that could fail.

° The effects of containment temperature for the main steam line break (MSLB) accident on coatings were not assessed.

Following a loss-of-coolant accident or main steam line break, water from the containment sump is used for makeup to the core and for containment spray. The sump has a 6-inch trash curb around the base with 1/4-inch wire mesh screens that slope upward and outward from the sump to prevent debris from entering.

Failure of coatings during a loss-of-coolant accident or main steam line break could lead to blockage of sump screens, thus an inadequate recirculation flow to the core or blockage of spray systems.

As a result of these weaknesses, TVA undertook corrective actions, which included physical repair of coatings, erection of screens to prevent transport of material, and implementation of a program to establish and maintain a log of the status of coatings and their qualification. As part of this effort, TVA has proposed to establish a new basis for operability of the plant with respect to the amount of coatings that could fail in a design-basis accident and how that material is treated in the transport analysis. TVA discussed this approach in its submittal of September 16, 1987.

The original basis for qualification of coatings was the accident conditions resulting from a design-basis LOCA.

The containment temperature profile for the LOCA does not bound the temperature profile expected from an MSLB. Thus, approximately 12,000 square feet of top-coated steel and 7500 square feet of concrete inside containment, which were previously qualified, would not be qualified for the MSLB conditions. Therefore, the debris from coatings following an MSLB would be more severe than following a LOCA.

Staff discussions with the licensee and the material manufacturer provided information about the containment coatings. Carbozinc II, also known as Carboline CZ II, was the inorganic zinc primer used on the steel. The topcoat on the steel was a phenolic, Carboline 305. For the concrete, Carbozinic 295, a waterbase polyamine or polyamide was used without the need for a primer. The topcoat was also Carboline 305. The primer coating on the RCP motor, which was the vendor-supplied component that was recoated, was Ameron D6. It was given a polyamine topcoat. The topcoat has performed satisfactorily in radiation resistance and decontamination testing. The licensee has qualified the organic coating materials for conditions up to and including the design basis loss-of-coolant accident.

As a result of this information, TVA re-evaluated the licensing basis for the containment sump screen blockage. In the FSAR, an arbitrary blockage of 50 percent of the screen area had been assumed.

Westinghouse performed a physical transport study to determine if the containment spray and emergency core cooling systems could be operated safely if debris were present from coating failures. The Westinghouse study examined the effects on net positive suction head (NPSH) of sump screen blockage caused by coating and insulation debris. The study focused on a near-sump region that would be affected by post-accident flow fields and addressed the potential effects of the return of containment spray flow through the refueling canal drains. Both reflective metallic insulation and fibrous NUKON insulation were included in the study, as well as other coatings that potentially could fail.

The study indicated that under MSLB and LOCA conditions with sump screen blockages of up to 90 percent, adequate NPSH would be provided for the containment spray and RHR pumps. The study also showed that at least 12 percent of the sump screen area would be protected from blockage by the shielding provided by a 45-inch-diameter crossover pipe located directly in front of the screen, an 8-by-8-inch wide flange material to one side of the screen. In addition to the shielding, the sump screen is designed with an upward and outward side slope from the sump, which further prevents debris from blocking the screens.

On the basis of this information, the staff concludes that a sufficient area of the sump screen would remain unblocked following an MSLB or a LOCA to allow the containment spray and RHR pumps to operate safely. Therefore, the containment coatings issue is considered resolved.

3.8 Moderate Energy Line Breaks

3.8.1 Introduction

In Section III.15.2 of the SNPP, TVA identified the actions it would take before restart to correct the moderate-energy line break (MELB) flooding issue. These corrective actions were originally identified in Sargeant and Lundy Report SL-4424, transmitted by a TVA letter dated July 2, 1987(b). This report defined the scope and design criteria for the evaluation as well as the results and recommendations for corrective actions to achieve safe shutdown during a MELB flood. The evaluation covered plant operating conditions during reactor startup, refueling, testing, operation at power, hot standby, reactor cooldown, and cold shutdown.

In addition to the Sargeant and Lundy report, TVA performed an analysis to determine the effects of internal flooding during different modes of operation. The results of this study were used to determine which recommendations (from the Sargeant and Lundy Report) must be accomplished before restart and which could be delayed until later. The staff reviewed the original TVA analysis, dated March 27, 1987, during an audit. Revision 1 of the analysis was submitted to the NRC on October 9, 1987.

The purpose of the analyses performed by the licensee and its contractors was to demonstrate that safe plant shutdown could be achieved for design-basis MELB flooding events or to determine what modifications to the plant were necessary to achieve safe shutdown. These studies included the elements listed below.

° flood level calculations (including field verification of input parameters)

° structural load assessment

° safe shutdown evaluation (including field identification of submerged Class 1E electrical equipment)

° safe shutdown power supply analysis

° cable submergence analysis

3.8.2 Evaluation

The staff's evaluation of TVA's analyses is discussed below.

Flood Level Calculations

Two important modeling assumptions were made for the flood level calculation analyses: (1) only one piping failure was assumed for each MELB event, and (2) no credit was taken for flow in floor drains. Using these assumptions, flood level calculations were performed for flooding events in 250 flood zones in the auxiliary, control, diesel-generator, and reactor buildings and in the ERCW pumping station. Two flood levels were calculated for each flood zone, one for flooding sources originating within the zone (h1) and one for flooding originating outside of the zone (h2).

The duration of fluid inflow from a postulated MELB was generally assumed to be taken as 60 minutes. This inflow time is significantly longer than for high-energy line break (HELB) events because of the general unavailability of automatic isolation for moderate energy systems. For most zones (approximately 80 percent) calculated flood levels are independent of the assumed inflow duration. These levels represent steady-state levels where inflow is balanced by outflow.

The staff considers the basic assumptions used by the licensee in the calculation of flood levels to be acceptable.

Structural Flood Load Assessment

The structural assessment included a review of the affected slabs, beams, columns, and walls for each zone. The qualification of the slabs, beams, and columns was based on a comparison of postulated flood loads to the allowable floor live loads provided by TVA. Walls were qualified by comparing postulated flood levels to the wall capacities that were generated by Sargeant and Lundy. The staff considers this structural assessment to be adequate.

Safe Shutdown Evaluation

The safe shutdown evaluation examined MELB flooding on a zone by zone basis. TVA conducted field walkdowns to identify Class 1E electrical equipment that was indicated to be submerged by the calculated MELB flood levels. When the field walkdowns verified which essential equipment would be submerged, the ability to achieve safe shutdown was evaluated for flooding events that could submerge that equipment. Other Class 1E electrical equipment that could be submerged concurrently also was considered. Required system controls and instrumentation were examined through use of block diagrams. The licensee has stated that components needed for safe shutdown are not submerged by the MELB flood levels. The staff finds the licensee's safe shutdown analysis to be acceptable.

Safe Shutdown Power Supply Analysis

An evaluation of the auxiliary and the control power systems was made to ensure the availability of required shutdown boards and control circuits. The primary objective of the auxiliary power systems review was to evaluate the likelihood of increased board loading that would result from equipment that is not safety related being submerged and to determine if this increased loading could be sufficient to trip the main breaker. The control circuit study also was performed to determine if the flooding of shutdown equipment that was not safety related could potentially disable required shutdown boards. The staff considers this approach to be acceptable.

Cable Submergence Analysis

Cables in cable trays and in conduits were evaluated to determine which would become submerged if flooding occurred. It was assumed that cable trays routed below the maximum expected flood level would become submerged, as would cable trays routed from floor to floor. Cables in conduits were assumed to become filled if the conduits have openings or fittings that are not water tight and if they are located below calculated flood levels. Cables that may become submerged were identified as requiring flood protection. The staff considers this approach to be acceptable.

Evaluation of Neoprene Seal Modification on Door C-14

The October 9, 1987 submittal also provided evaluation of the modification to fire-rated door C-14, connecting the turbine building floor with the auxiliary instrument room at elevation 685. The modification will consist of placing a small strip of neoprene on the door frame sides and on the bottom of the door, leaving a 1/32-inch gap to the sealing surface. The licensee determined that the neoprene seal will not have a significant impact on the fire rating of door C-14. The staff agrees with this determination.

Summary

The Sargeant & Lundy analysis identified 10 recommendations for corrective action. In addition, Sargeant & Lundy recommended that TVA consider resetting the auxiliary building supply fan breaker to reduce the degraded voltage duration. TVA used the result of this analysis to determine what modifications were needed to ensure full protection of the plant from MELBs flooding in all modes of operation.

TVA broke down these 10 items into 27 separate tasks. Six of these tasks were to be accomplished before Unit 2 restart, and the remaining 21 items would be accomplished before startup from the cycle 4 refueling outage. The justification of delaying these 21 items until after restart is addressed in Calculation SQN-SQS4-0088, "Justification for Continued Operation with Unimplemented Corrective Actions for Moderate Energy Line Breaks." A new item was added to the 21 post-restart items in Revision 1 of the calculation. In its SQN-SQS4-0088 calculation, the licensee examined the effects of such factors as operation of the condenser cooling water pumps, the operability of the annulus sump alarm system, electrical equipment flooding, and the probability of MELB occurrences to establish justification for postponing certain action items until after restart. These factors were used to implement the restart requirement criteria as listed in the SNPP. Effects of cable submergence, conduit sealing, spurious equipment operations, backflow through drains, safety injection test mode, and surveillance on flood alarms also were addressed to justify the chosen restart actions. Revision 1 to the calculation states that degraded bus voltage will be resolved as a generic issue rather than as part of the MELB issue.

The staff has reviewed the logic presented in SQN-SQS4-0088 and accepts the justification for limited deferral of selected tasks. However, the staff believes that possible adverse effects from MELBs can be further limited by requiring appropriate licensee personnel to familiarize themselves with shutoff valves for all moderate-energy lines leading to safety-related areas.

3.8.3 Conclusion

The staff accepts the licensee's procedures and assumptions for evaluating MELB flooding. The staff further accepts the licensee's commitment to complete the actions listed below before restart.

(1) ensure adequate sealing between the turbine building, control building, and the auxiliary building

(2) provide administrative control for possible flooding in the annulus

(3) verify that the electrical equipment and electrical boards on the 734' and 749-foot level are above MELB flood levels

(4) update the previous review of unimplemented ECNs to determine if subsequent ECNs impact the flooding evaluation.

The staff concludes that completion of these actions (which includes all six restart tasks) will be sufficient for restart. However as a post-restart action, the staff recommends that TVA be able to demonstrate quick response to MELBs in safety-related areas.

3.9 ECCS Water Loss Outside Crane Wall/Air Return Fan Operability

3.9.1 Introduction

By letter dated July 8, 1987, and as supplemented August 4, 1987, the licensee identified a condition involving the collection of water from the containment and residual heat removal sprays following a design-basis accident (DBA). Spray water collecting on the operating deck floor could drain directly into areas outside the crane wall through the opening for the containment air return fan A-A. The concerns were that this drainage could result in undesirably low water levels above the sump and in flooding of the air return fan A-A.

3.9.2 Evaluation

The primary purpose of the air return fan (ARF) is to enhance the ice condenser and containment spray heat removal. The secondary purpose of the system is to limit hydrogen concentrations in potentially stagnant regions of containment by ensuring a flow of air from these regions. Two fans are provided.

The operating deck, located above the containment sump, is designed to collect falling spray water and divert it to the inner crane wall region through the refueling canal to the sump. The licensee identified a condition whereby, during containment spray operation, spray water could bypass its intended flow path to the inner crane wall region by draining directly to areas outside the crane wall through an opening for the containment air return fan A-A. Subsequently, the equipment access hatch and personnel access door trenches also were identified as potential inner crane wall bypass leakage paths. These trenches also would direct spray water through the opening for containment air return fan A-A. (The intake for fan B-B is above the floor elevation and this fan is unaffected by water drainage.)

The licensee has determined that the root cause of this condition to be a design deficiency that does not adequately prevent spray water interaction with the ARFs. In addition, the kick plates have not been maintained as required by design drawings. The kick plates on the operating deck were designed and installed to prevent runoff at the personnel access hatch. However, a portion of these kick plates were removed and not replaced because they would have interfered with movement of the personnel airlock door.

It has now been determined that, had the kick plate been maintained as designed, the estimated flow runoff through fan A-A would have been reduced. However, this reduction in runoff would not have been sufficient to preclude failure of ARF A-A. The licensee has since installed kick plates of a different design that prevent the spray water that has collected on the floor from draining into the air return fan and settling outside the crane wall.

Excess moisture in the containment atmosphere can be drawn through the air return fans and then exhausted to the accumulated rooms outside the crane wall. Containment spray is designed to direct spray inside the crane wall only. However, for the purposes of the analysis, the licensee conservatively assumed that they would be a homogeneous distribution of spray throughout the total air volume above the operating deck, including the region outside the crane wall. Using this assumption, the total rate of entrained water that would pass through the two fans has been calculated to be 70 gallons per minute per fan. The containment air return fans have been evaluated by the vendor and found to be capable of performing their intended function with this amount of entrained water in the containment air.

The RHR and containment spray pumps require a 13.2-foot sump water level to maintain the proper net positive suction head (NPSH). Entrained spray water that would pass through the air return fans would be diverted from the sump; thereby, reducing the sump water level and the pump's NPSH. However, to maintain the sump water level and the proper NPSH, the licensee has proposed certain modifications to trap the de-entrained spray water and drain that water back inside the crane wall. The necessary modifications to the drainage areas outside the crane wall (accumulator Rooms 3 and 4) consist of the following:

(1) install 5 inch curbs in each accumulator room, as required

(2) seal penetrations through the accumulator room floors

(3) using 4-inch piping, construct a drain line that runs from each accumulator room floor to inside the crane wall

(4) install orifices on the existing accumulator room floor drain lines to limit the total flow through them to less than 5 gallons per minute

All efforts associated with the curb and drain modifications have been completed on Unit 2; those modifications for Unit 1 will be completed before restart.

3.9.3 Conclusion

The staff concludes that re-design of the containment drainage system will ensure that spray water will not damage the air return fans or bypass the sump; therefore, the design is acceptable.

3.10 Platform Thermal Growth

The SNPP, Revision 1, Section III, item 15.5, "Platform Thermal Growth," deals with thermal loads on structures during a postulated LOCA.

On May 15, 1985, TVA received an employee concern on temperature variation in pipe hangers and supports (IN-85-103-002). As a derivative of this activity, TVA found that some structural and miscellaneous steel structures were designed and installed without proper consideration of thermal loading during a postulated LOCA. The requirements to consider thermal load are specified by license design criteria SQN-DC-V-1.3.3.1, the Standard Review Plan (Sections 3.8.3 and 3.8.4), and industry standards.

The special program contains a summary of the issues, a description of the intent and scope of the program, steps TVA has already taken to correct the issue, and the status of TVA's corrective actions. The staff's evaluation of this program plan is discussed below.

The staff reviewed the licensee's criteria dated August 11, 1980, which was used by TVA as a guide to re-evaluate thermal growths in structures by TVA. In particular, Section 4.2.2.b load combinations specify to combine thermal loads (Ta) with other loads such as dead and live loads and earthquake loads. The criteria are generally consistent with the Standard Review Plan; therefore, they are acceptable.

TVA performed a thorough drawing review to identify the structural and miscellaneous steel structures that appear to be thermally restrained to the point where thermal growth might damage the structure itself or adjacent structures. In the suggested corrective action of the Engineering Report (SCR SQN CEB 86103 R0, Revision 1) a thorough drawing review was recommended and subsequently performed by experienced design engineers who are familiar with thermal evaluation. A thermal evaluation was then performed for each of those structures that were identified in the drawing review. The primary function of the evaluation was to determine if the structure is ductile and if the stresses are secondary and self-limiting. For those structures that were shown to meet these criteria, the thermal load was ignored in accordance with the criteria and no further action was taken.

The staff found that the licensee's outline of the evaluation of the issue is reasonable. However, the staff will inspect the licensee's calculations to see if the criteria are adequately interpreted, if the scope of the evaluation is adequate, and if the quality of the calculations is acceptable. The staff also will review generic implication of the issue: relationship of this issue to the thermal variation in the pipe hangers/supports as described in the employee concern IN-85-103-002/Report Number 220.11(B).

For those structures that do not meet the above design criteria, a certain modification was introduced to the structures to allow thermal growth in key members. The staff believes that this allowance for a free thermal growth will alleviate thermal stresses in the structures. However, too much alteration of the structure may change the basis for the floor response analysis of the equipment since structural natural frequencies may change. This aspect will be reviewed by the staff during an audit.

TVA's review and evaluation identified four miscellaneous steel structures that required modifications. The four structures were the instrument room access platform, the reactor coolant pumps access platforms for loops 3 and 4, and the pipe support framing in accumulator room 4. The applicable drawings have been revised for implementation.

On the basis of its review, the staff concludes that TVA's proposed resolution procedures are acceptable.

3.11 Pipe Wall Thinning Assessment

3.11.1 Introduction

On December 9, 1986, Unit 2 at the Surry Power Station experienced a catastrophic failure of a main feedwater pipe, which resulted from the erosion/corrosion of a carbon steel pipe wall. Although erosion/corrosion pipe failures have occurred in small diameter piping containing a water-steam mixture and in water systems containing solids, there have not been any previously reported failures in large diameter carbon steel piping systems containing high-purity water; thus, the licensee did not have a procedure for the systematic examination of the thickness of the walls of the feedwater and condensate piping.

Main feedwater systems, as well as other power conversion systems, are important to safety. Failure of piping containing high-energy fluids such as the feedwater system can result in complex challenges to the operating staff because of potential interactions of high energy steam and water with other systems, such as electrical distribution, fire protection, and security. The licensee's commitments for the functional capability of systems containing high energy fluids are a part of the licensing basis for the facility; an important part of this commitment is that piping will be maintained within allowable thickness values.

3.11.2 Evaluation

The staff's evaluation is based on Volume 2 of TVA's Nuclear Performance Plan and meetings with the licensee on June 29, September 14 and 30, and October 29, 1987. Information was also obtained from the licensee's response to NRC Bulletin No. 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," which is being evaluated separately. TVA's response of September 18, 1987, included its tests and inspections of piping.

Erosion/corrosion for carbon steel piping is a combination of rusting and loss of material as a result of moving water or steam or both. The licensee selected areas susceptible to erosion/corrosion based on base metal composition, flow velocity, pressure differentials, unusual flow path or geometry, and operating temperature. Inspection was by visual and ultrasonic testing (UT) methods. The five susceptible systems are listed below.

$ condensate (single phase)
$ feedwater (single phase)
$ extraction steam (two phase)
$ heater drains and vent lines (two phase)
$ turbine drain and vent lines (two phase)

The licensee submitted inspection reports detailing the extent of wall thickness testing. In 1983 the licensee replaced a portion of the Unit 1 moisture separator reheater drain tank steel line that had failed as a result of steam erosion. In 1984 the extraction steam lines were examined. There was evidence of wall thinning in some areas, but the thicknesses, as measured by UT, exceeded the calculated minimum wall values. The piping downstream of the level control valves was changed to stainless steel to prevent future problems.

A preliminary report of the suspected erosion/corrosion areas on the condensate and feedwater piping dated January 27, 1987, described the testing procedures and the selection of locations to be tested. Some loss-of-wall thickness was detected on a reducing elbow downstream of a feedwater pump, but this was determined by the licensee to be the result of cavitation. Erosion/corrosion had not occurred in the areas most likely to be damaged.

The recent inspections for Unit 2 are summarized in reports, "Preliminary Evaluation of the Turbine Building Heat Cycle Piping" dated March 6, 1987, and "Wall Thinning Assessment Program Final Report" dated April 8, 1987. Approximately 70 areas were examined by UT and the results were compared with the material specification's minimum wall thickness and the licensee's design minimum wall thickness. Any measurement below specifications was noted, and those areas found below or rapidly approaching the design minimum were targeted for replacement. This data is being used to establish a data base for tracking purposes. Significant thinning was detected in several locations. One 2-1/2-inch high-pressure reheater operating vent line elbow had about 50 percent erosion; this elbow was replaced in all three lines. The elbows in the four 16-inch feedwater lines immediately downstream of 12-inch valves were eroded below the minimum wall value as a result of high local water velocities. These safety-related elbows have been repaired by overlay welding.

Chemical samples were taken of degraded and erosion/corrosion-resistant fittings. As expected, the resistant fittings contained elements known to give corrosion resistance while the degraded fittings did not.

The licensee submitted copies of the UT procedures and surveillance instructions for the wall thinning program. The licensee plans to monitor susceptible areas and trend the results.

3.11.3 Conclusions

The NRC staff concludes that the licensee's inspection and surveillance program is acceptable. The staff finds that monitoring the licensee's implementation of the surveillance program is not necessary at this time.

3.12 Cable Installation

A number of employee concerns were received relating to construction practices at Watts Bar, particularly with respect to cable installation. The evaluation of these concerns was extended to the Sequoyah plants.

The NRC and its consultant, Franklin Research Center (FRC) conducted a review of installation procedures at Sequoyah, plant walkdowns, and interviews with electricians who had installed cables in the plants. The results of this review were transmitted to TVA by letter dated March 9, 1987. In that evaluation, the staff concluded that tests should be conducted for Sequoyah before restart to assess potential damage for three situations: (1) cable pullbys, (2) cable jamming, and (3) vertical cable supported by 90-degree condulets. TVA developed a test program to address the staff's concerns, which was subsequently revised in consultation with the NRC; this program is described in a TVA submittal dated July 31, 1987. TVA has completed its testing of cables for these three issues; the results were submitted to NRC by letter dated November 20, 1987.

During its testing, TVA identified potential insulation deficiencies with silicone rubber insulated cables supplied by three vendors: American Insulated Wire (AIW), Anaconda, and Rockbestos. Some silicone rubber insulated cables have failed in-situ high potential (hi pot) tests and some uninstalled, new, but drop-weight impacted cables have failed laboratory testing that was conducted to ascertain the potential for cable damage from normal stresses expected during shipping, handling, and installation. TVA provided a report on these failures pursuant to 10 CFR Part 21 because it believed the findings could affect other plants. There are about 960 silicone rubber insulated single conductor cables inside containment, totaling about 60,000 feet.

In a letter dated November 13, 1987, the staff informed TVA that based on TVA's information developed up to that time, all the silicone rubber insulated cables at Sequoyah were considered suspect. Although the generic concerns associated with the use of this material in other plants are under review by the staff, it was the staff's position that this issue must be resolved for Sequoyah before restart. TVA was told that if it elects not to replace these cables, then TVA will have to demonstrate, before restart, that these cables will perform their intended safety functions in a harsh environment.

On November 24, 1987, in a meeting between NRC and TVA, TVA presented results from tests conducted at Wyle Laboratory. The results were also submitted to the staff in a letter dated November 24, 1987. The results demonstrate that a significantly lower insulation thickness than originally anticipated is necessary for installed cables to perform their intended function during and after a LOCA.

In a letter dated December 28, 1987, TVA documented its basis for concluding that the silicone rubber insulated cables installed at Sequoyah are adequate to perform their intended function. TVA also informed the staff that, as a result of a decision made before the Wyle Laboratory test results were known, all the

AIW cables in safety-related harsh environment applications and the associated Anaconda and Rockbestos cables mixed in the AIW cable conduits in Unit 2 containment have been replaced. These cables were replaced with cables acceptable to the staff.

The staff has reviewed the TVA test data and concluded that the remaining installed silicone rubber cables--Anaconda and Rockbestos--are acceptable for service. The Wyle Laboratory tests represent partial qualification of the silicone rubber cables for a period of 10 years, which provides sufficient margin for startup. However, TVA will qualify the cables for the expected life of Sequoyah before return to operation from the refueling outage.

3.13 Fuse Replacement

Bussman, the KAZ fuse manufacturing company, informed TVA in early 1986 that KAZ actuator devices cannot be used as a fuse in 6 ampere or lower rated 125-volt dc circuit and 600-volt ac circuit applications. The device can only be used in parallel with a higher rated fuse, so that when the higher rated fuse blows, the KAZ also blows; and the indicator pin actuates the annunciator circuit.

In June 1986, TVA decided to replace the KAZ actuators with MIS-5 indicating fuses manufactured by Bussman. However, Bussman could not provide fuses that had been seismically qualified. Hence, TVA contracted Northern International, Inc., to supply seismically qualified MIS-5 fuses. As of October 1986, TVA had replaced approximately 2,500 KAZ actuators with MIS-5 actuating fuses.

In October 1986, TVA suspected that MIS-5 fuses were defective because of the failures that had occurred, and suspend installation of MIS-5 fuses. A 10 CFR Part 21 report was submitted to NRC on October 29, 1986.

In January 1987, TVA contracted Littlefuse, Inc., to supply indicating fuses, FLAS-5 model. TVA contracted Wyle Laboratories to seismically qualify the FLAS-5 fuses. The FLAS-5 fuse consists of a fuse wire, 560 ohm resistor, spring-loaded indicator pin, and sand-like filler. The indicator pin is mechanically attached to the spring. At the end of the spring, the resistor and the fuse wire are soldered together. The solder material used is a eutectic alloy that has a low melting point. During overcurrent or fault conditions, the solder joint melts and releases the indicator pin. The indicator pin serves to cause annunciation only and does not trigger any safety features.

TVA installed the FLAS-5 fuses by March 1987, and Region II completed the inspection (Inspection Report 50-327/328 87-20) during the week of March 23, 1987 and found the replacement acceptable. However, on June 20, 1987, an FLAS-5 fuse blew in a diesel generator (DG) start circuit that started all four DGs. TVA investigated the problem and found that FLAS-5 fuses from lots 2 and 3 were inadvertently blowing without the component in service or any other activity in progress. Discussion with Littlefuse, Inc. revealed this to be a creep failure problem introduced during the manufacturing of the fuses in lots 2 and 3.

The problem was believed to be corrected by changing the solder material and soldering process during the manufacturing of subsequent lots. TVA submitted a licensee event report dated July 21, 1987, on this problem.

By letter [November 17, 1987(b)], TVA submitted the testing performed on the FLAS-5 fuses to determine the cause of failure in lots 2 and 3 and to demonstrate the reliability of subsequent lots of fuses.

Scanning electron miscroscope photographs indicated partial melting was present in all of the failed fuses. Those photographs showed a large amount of porosity in the solder and one fuse with almost no solder. These problems point out the poor quality control exercised during the manufacturing process of these fuses. This was a preliminary conclusion before the creep failure mechanism was identified by later tests. TVA also subjected four fuses from lot 3 and four fuses from lot 6 to a temperature of 200°F under no electrical load. The first fuse from lot 3 failed within 12 hours. All other fuses from lot 3 failed within 80 hours. The first fuse from lot 6 failed at 44 days and the last fuse from lot 6 did not fail even after 71 days.

It should be noted that bismuth was included in the solder for fuses in lots 2 and 3 while cadmium was used on fuses from lots 4 and higher. Bismuth, because of its low melting point, is believed to be the cause of failure of the fuses in lots 2 and 3.

TVA also subjected these fuses to a long-term current test. Fuses from lot 3 were subjected to 2 ampere and 4 ampere circuits. Out of 20 fuses, 11 fuses failed in the 4 ampere circuits within 33 days. The first fuses failed within 5 days. In the 2 ampere circuits, only one fuse failed (at day 26) during a test of 40 days. No fuses from lots 4 and 6 failed in the long-term current test.

TVA, at its Singleton Materials Engineering Laboratory, has performed tests on the FLAS-5 fuses with cadmium bearing solder (lots 4 and upward) to evaluate the temperature dependence of the creep rate. During these tests, TVA conducted limited stress rupture tests on the fuses at 100, 120, and 143°F. These data combined with results of tests performed by Littlefuse, Inc. at 78 and 200°F provide the predicted service life of the FLAS-5. These tests prove that soldering material used in these fuses are expected to undergo a time-dependent increase in length (creep) under a constant load at elevated temperatures. TVA also has measured the temperature rise above ambient at various points of the FLAS-5 fuses. Based on the expected life tests and fuse temperature rise tests, together with knowledge of fuse loading and ambient temperature, TVA has predicted the service life of the solder junctions to be 80 months average and 25 months minimum.

TVA also has performed short circuit tests on samples of both types of fuses in which the bismuth solder fuse indicating mechanism operating in 37.15 seconds whereas the cadmium indicating mechanism operated in 37.45 seconds, an insignificant time difference.

TVA has committed to replace bismuth solder FLAS-5 fuses from lots 2 and 3 with cadmium solder fuses before operating at mode 4.

On the basis of these tests, it can be reasonably determined that the failure of the fuses had been caused by a creep problem. These tests also prove that cadmium solder fuses from lots 4 and higher are more reliable and will have less tendencies of failure because of the creep problem than bismuth solder fuses of lots 2 and 3.

TVA has informed the staff that cadmium fuses (FLAS-5 lots 4 and higher) have blown because of short circuit conditions and not creep failures as experienced with lots 2 and 3. Based on the test results and experience with the FLAS-5 cadmium solder fuses from lots 4 and higher, the staff finds the replacement fuses acceptable. However, because the analysis performed by TVA on the service life of the solder junction is predicted to be 80 month on the average and 25 month minimum, TVA should either replace these fuses every 25 months or extend the life of these fuses with further testing and analysis based on the ambient conditions and failure rates of these fuses.

4 RESTART READINESS

There are a number of programs necessary for safe conduct of nuclear activities at Sequoyah. These include management performance, maintenance, quality assurance and training. The management controls, initiatives and procedures related to these activities are discussed below. Numerous inspections of the effectiveness of these programs have also been conducted and will continue.

4.1 Operational Readiness

4.1.1 Introduction

TVA has historically demonstrated weaknesses in performance of nuclear activities as has been discussed in previous Systematic Assessment of Licensee Performance (SALP) reports. On September 17, 1985, on the basis of continued poor performance as described in the fifth TVA SALP, the NRC issued a letter delineating their concerns pursuant to 10 CFR 50.54(f).

Enclosure 2 to the staff's 10 CFR 50.54(f) letter posed certain questions to TVA regarding

(1) equipment qualification (Questions 1 and 2)
(2) operational readiness (Question 3)
(3) cable tray support (Question 4)
(4) design control (Question 5)

Items (1), (3), and (4) are discussed in Sections 3.2, 2.5, and 2.1, respectively, of this report. Operational readiness will be discussed in this section.

TVA has undertaken a significant effort to address and correct operational readiness issues. A special Sequoyah Task Force was established by the Manager of Nuclear Power on March 19, 1986, to identify problems and initiate those actions necessary to resolve the problems before restart of either Sequoyah unit. The Sequoyah Nuclear Performance Plan (SNPP), Revision 1, provides the assessment and plans for resuming operation of the Sequoyah units and Section V discusses those topics related specifically to operational readiness.

TVA has stated that the overall purpose of operational readiness is to provide the Site Director with verification that activities, programs, and commitments required for restart are completed. This is to be accomplished by designating an Operational Readiness Manager who reports to the Manager, Office of Nuclear Power (ONP) and an SQN Operational Readiness Manager who reports to the Site Director. The Operational Readiness Manager provides independent oversight of the development and implementation of the operational readiness program and assists the site in ensuring the program adequacy while also providing independent assessments and evaluations to the Manager of Nuclear Power. The Site Director will use the results of the operational readiness program and other status reviews to make his recommendation for Unit 2 restart to the Manager, ONP. The Manager, ONP, will not approve restart of Unit 2 until he is satisfied that all preparations for restart have been satisfactorily completed.

The SQN Operational Readiness Manager assesses whether corrective action plans have been established to address the underlying causes of deficiencies or problem areas, evaluates the adequacy of corrective action, reviews the close-out practices and provides comments to improve the process and program content. The SQN Operational Readiness Manager is responsible for working with the site and line organizations to obtain verification of program implementation, to obtain verification of organizational readiness through the evaluation of performance objectives, and to develop the restart prerequisite checklist. The checklist will be used to verify that hardware issues directly impacting system operability are closed before applicable mode changes.

4.1.2 Evaluation

Success of the operational readiness program is contingent upon the successful implementation of the three program elements: the SNPP completion of Volume 2 programs, the establishment and assessment of performance objectives, and the restart prerequisite verification (Restart Test Instruction 1.1-Master Test Sequence).

Implementation of the first element will be to verify (1) that restart activities as defined in the Sequoyah Activities List (SAL) have been completed, (2) that SNPP Volume 2 text statements of intention have been completed, and (3) that major projects, having broad impact on other plant activities, have been completed prior to restart. Some long-term program enhancements will be open at restart and will be tracked through routine NRC observations of the TVA corporate commitment tracking system.

The purpose of the performance objectives evaluation is to ensure that site organizations function effectively and are prepared for plant restart and operation. Generic performance objectives and criteria have been established and assigned to site organizations so that they may address the areas of procedures, staffing, supervisory involvement, internally and externally identified findings, housekeeping, and readiness of support organizations during restart. Additional performance objectives and criteria have been developed for the functional areas of organization and administration, document control, maintenance, training, licensing, engineering, and configuration control. Performance objectives in these functional areas also have been assigned to the appropriate site organizations.

TVA's performance objectives are based on the guidance provided by "Performance Objectives and Criteria for Operating and Near Term Operating License Plants," INPO 85-001, Institute for Nuclear Power Operations, January 1985.

This interim operational readiness evaluation will include the following:

- establishing appropriate objectives and criteria
- evaluating readiness against established criteria
- assessing impact of deficiencies identified

- developing and implementing additional corrective actions for identified deficiencies
- verifying that performance objectives have been met and readiness is assured

TVA has established plant instructions and tracking systems to ensure that hardware issues directly impacting system operability are closed before mode changes. To ensure that these hardware issues are complete, a restart prerequisite checklist has been developed. This checklist was developed by the SQN operational readiness staff and serves to consolidate hardware operability issues, including those listed below.

- maintenance or work request backlog
- outstanding clearances
- modification status
- outstanding temporary alteration control forms (TACFs)
- outstanding preventive maintenance packages
- instrumentation availability
- outstanding hardware-related PROs and SCRs

The restart prerequisite checklist will be provided to the Sequoyah Restart Test Manager for inclusion in the plant restart test sequencing instruction. This instruction will provide for PORC review and plant manager approval of results prior to leaving specified hold points. In addition to incorporating the restart prerequisite requirements, this instruction will address the completion of required special testing during the restart of SQN.

TVA will provide two reports, an initial report and a final report to document the operational readiness program.

The initial report provided

- the status of each SAL item
- the status of each Volume 2 restart text intention
- closure criteria approved by the principal manager for each defined major project/issue
- the status of the performance objective/criteria evaluations
- a copy of the current draft restart checklist

The final report will provide

- a revised update of the initial report to document operational readiness
- a detailed description of the remaining open items
- the specified mechanism for ensuring closure and the method by which closure will be documented for open items
- the final restart prerequisite checklist as submitted to the SQN Restart Test Manager

A parallel, independent assessment of operational readiness was performed by the ONP Operational Readiness Manager. This review was conducted by senior personnel with plant experience from both inside and outside TVA. The team provided its findings and recommendations to the Manager of ONP in a letter dated January 5, 1988. This managerial group may be augmented from time to time by additional senior personnel within or outside TVA to provide special expertise in particular areas. Further, the Manager, ONP, has requested that the SQN Nuclear Safety Review Board (NSRB) review the SNPP Volumes 1 and 2 and the actual status of preparation for restart of Sequoyah units from a safety perspective. The NSRB has reviewed and accepted the overall approach outlined in the SNPP. The Board also has reviewed the special programs and certain secondary hardware issues and the onsite safety review process, maintenance planning and procedure development. The Board will review the restart test program, and, on the basis of these reviews, it will provide NSRB recommendations to assist the manager in his restart decision.

The initial report has been reviewed by the staff. The NRC staff will review and evaluate the final report and the Independent Readiness Review as part of the ongoing staff evaluation of the implementation of the Operational Readiness Review Program.

4.1.3 Conclusions

Initially, the staff believed that TVA needed to clarify the meaning of hardware issues in the paragraph describing the restart prerequisite verification element. In addition, provisions needed to be included to ensure that TVA assesses hardware operability for the cumulative effect on system performance. Overall the staff has concluded that the implementation portion of the operational readiness program represents a realistic and systematic format to ensure that plant activities, programs, and commitments required for restart are completed.

On the basis of its review, staff finds that this program is acceptable. As designed the program should provide the Site Director and Manager of Nuclear Power verification that activities, programs, and commitments required for restart are completed.

4.2 Management

4.2.1 Introduction

TVA's SNPP states that in the past there has been a lack of clear assignment of responsibility and authority to managers and their organizations. To correct this weakness, TVA has reorganized the Sequoyah site organization. TVA also has taken specific actions to clarify each manager's authority and area of responsibility and to establish accountability. TVA also has programs under way to improve the level of plant knowledge of plant managers and supervisors.

The staff has reviewed several efforts by TVA to improve the management and organization at Sequoyah and agrees with the type of programmatic changes being made. The staff inspected some of these programs during Inspection 50-327/328 87-59; the purpose of which was to evaluate the management systems at Sequoyah by focusing on the following specific functional areas: operations, maintenance, quality assurance, modifications, engineering, and licensing. The inspection looked at the process by which TVA was implementing the commitments in Volume I and Volume II of the TVA Nuclear Performance Plan.

4.2.2 Evaluation

4.2.2.1 Organization at the Sequoyah Site

Sequoyah site organization is organized into functional departments that generally parallel the functional departments in TVA's headquarters Office of Nuclear Power. The functional alignment of the Office of Nuclear Power is discussed in the staff's SER of the Corporate Nuclear Performance Plan (NUREG-1232, Volume 1). In that SER, the staff stated that corporate functional area managers are responsible for the technical direction in each functional area at each of the nuclear sites. The Sequoyah site organization showing this functional alignment is presented in Figure 4.1. Each site functional department is responsible for a discrete type of function.

The Sequoyah Site Director, through his organization, approves and controls all activities conducted on site. The Site Director is responsible for planning, scheduling, coordinating, and providing direction for the activities of the site organizations. The Plant Manager, Site Services Supervisor, Manager of Projects, Planning and Scheduling Supervisor, Financial Planning Supervisor, Radiological Assesssor, and Personnel Services Supervisor report directly to the Site Director. The site Project Engineer, Licensing Manager, Site Quality Manager, Site Procedures Staff Manager, and Modification Manager report to the Site Director for day-to-day functional supervision, but each of these individuals reports administratively and technically to his director in the corporate office. The Site Director maintains an interface with the Directors of Nuclear Engineering, Nuclear Licensing and Regulatory Affairs, Nuclear Quality Assurance, Nuclear Construction, and other TVA organizations to ensure effective implementation of corporate goals and objectives.

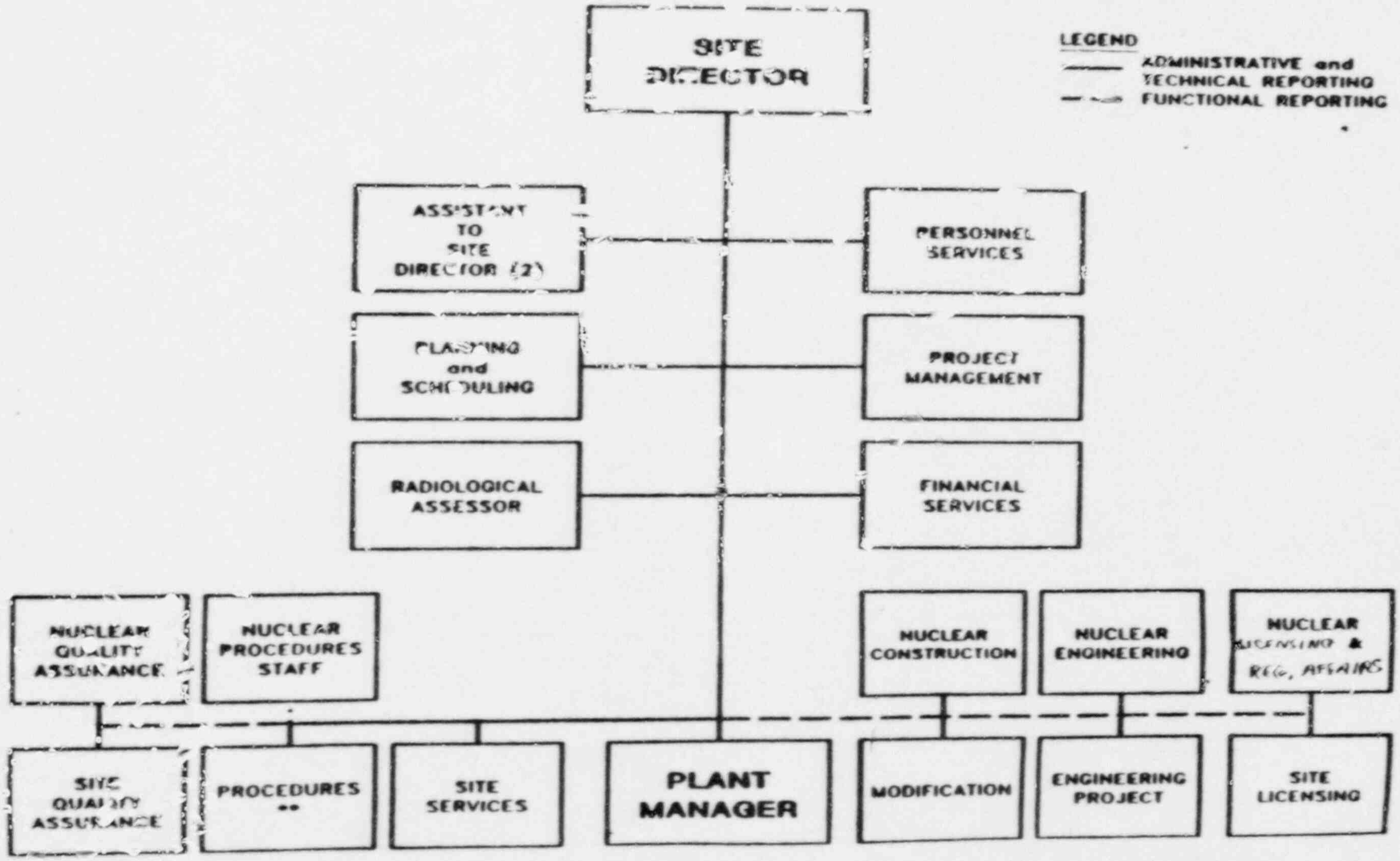
The Sequoyah plant organization is shown in Figure 4.2. The Plant Manager is responsible for conducting the day-to-day plant operations in compliance with licensing regulatory requirements. A plant management organization has been implemented with a unit superintendent assigned to operations and a unit superintendent assigned to maintenance.

In summary, the staff considers the site organization acceptable and in accordance with the guidance of Section 13.1.2 of the Standard Review Plan, NUREG-0800.

4.2.2.2 Responsibility, Accountability, and Authority

As described in the revised Corporate Nuclear Performance Plan (CNPP) the lines of authority and responsibility have not always been clear. To correct this problem, TVA is revising the position description program.

The position description program is a continuing program that is constantly being updated. After the organization charts and functional statements are approved, a great many of the position descriptions will need to be rewritten. NRC recognizes that this will be a large effort. Position descriptions have been written for each manager within the Office of Nuclear Power (ONP). Position descriptions define the functions, responsibilities, reporting relationships, and qualification requirements for each management position. Each



** Temporarily reports to Site Director; Long-term function will report to Site Services

Figure 4.1 Sequoyah site organization

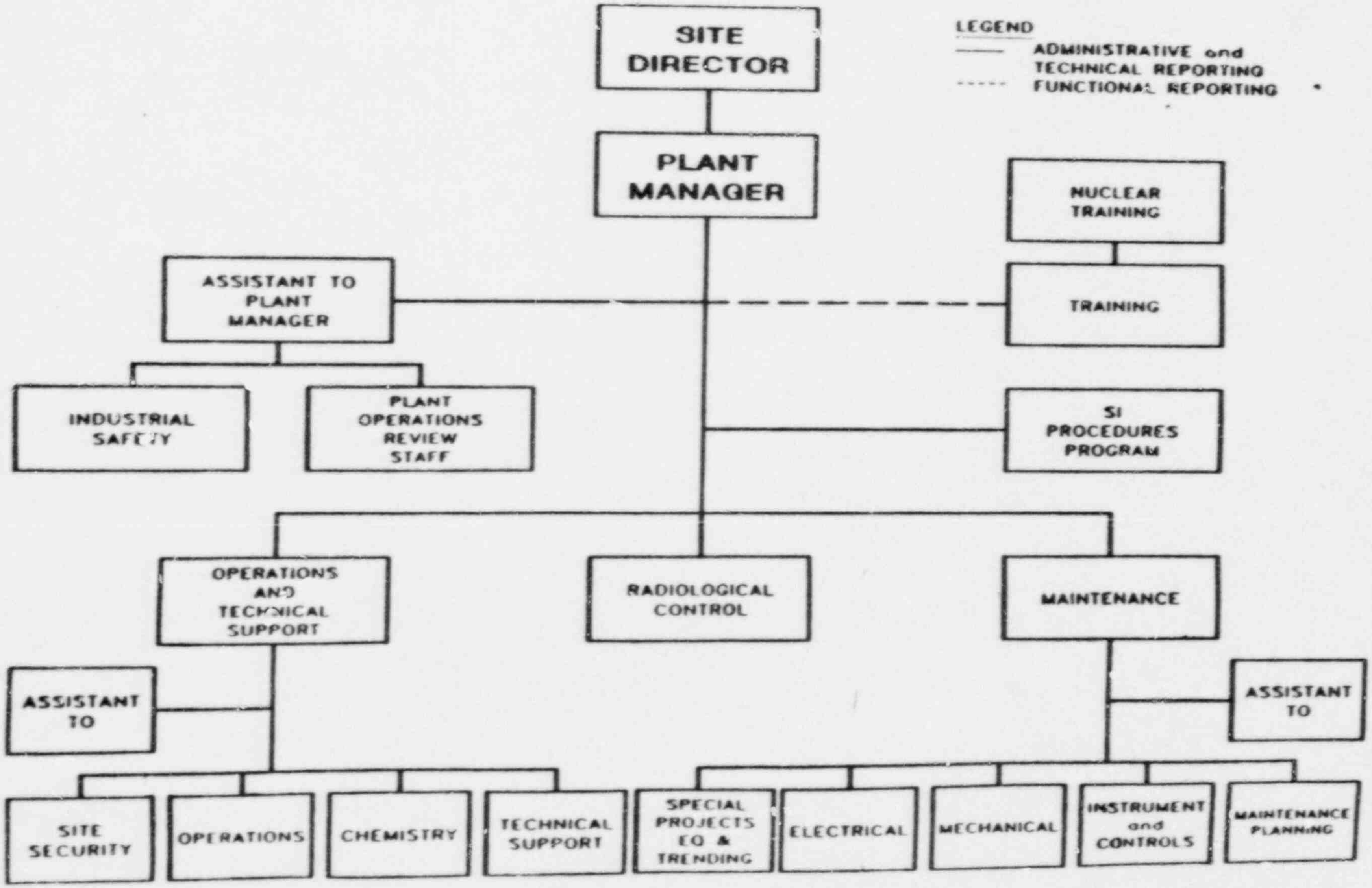


Figure 4.2 Sequoyah nuclear plant organization

position description will eventually be written according to TVA Procedure No. 0906.01, "Position/Organization Control Process." The line manager is responsible for the position descriptions for managers and the job descriptions for non-managers. Job descriptions are essentially the same for non-managers and managers.

Organizational charts will include functional statements for each group depicted on the chart. Interface agreements between organizations will define accountability and will be part of the organizational chart approval process. Each organizational chart is being signed by the Director of ONP. The process provides strong centralized control of the organization development process.

The Organization Charts Manual will be a controlled document that contains approved charts, thereby providing control. A position control system will provide a number for each position within ONP.

In summary, the position/organizational control process establishes the controls necessary to develop and maintain position descriptions, job descriptions, organization charts, and staffing plans. The process has very strong corporate management direction. However, because of the large number of organizations and individuals involved, the process is moving slowly.

4.2.2.3 Management's Level of Plant Knowledge

TVA has taken action at SQN to increase the level of plant knowledge in its line managers. Figure 4.3 shows the staffing qualifications necessary for key Sequoyah plant managers and supervisors. In addition, many other site supervisors have received the systems portions of either managers and engineers certification training or STA training.

The Managers and Engineers Certification Program provides an opportunity for individuals with a degree, who are considered to be potential candidates for upper plant management positions, to receive training necessary to gain simulator certifications. This program is designed to provide the trainee with an extensive knowledge of plant theory and operations. Included in the program are 15 weeks of systems and theory training along with 7 weeks of simulator/operations training. Candidates must pass comprehensive written and oral examinations similar in nature to SRO certification examinations before receiving their simulator certification.

Technical training for technical staff and managers is one of the TVA training programs accredited by INPO. Sequoyah Procedure 202.17 describes the requirements for the TVA Technical Staff and Manager Training Program, which is designed to provide general technical training needed by plant technical personnel. It is not intended to supply discipline-specific training. The Management Training Program provides management and supervisory skill training.

The first phase of the technical staff and managers training is called the orientation phase. The orientation phase is normally accomplished within the first 18 months of holding a technical staff position. The following training is included:

° General Employee Training
° Plant Reference Material

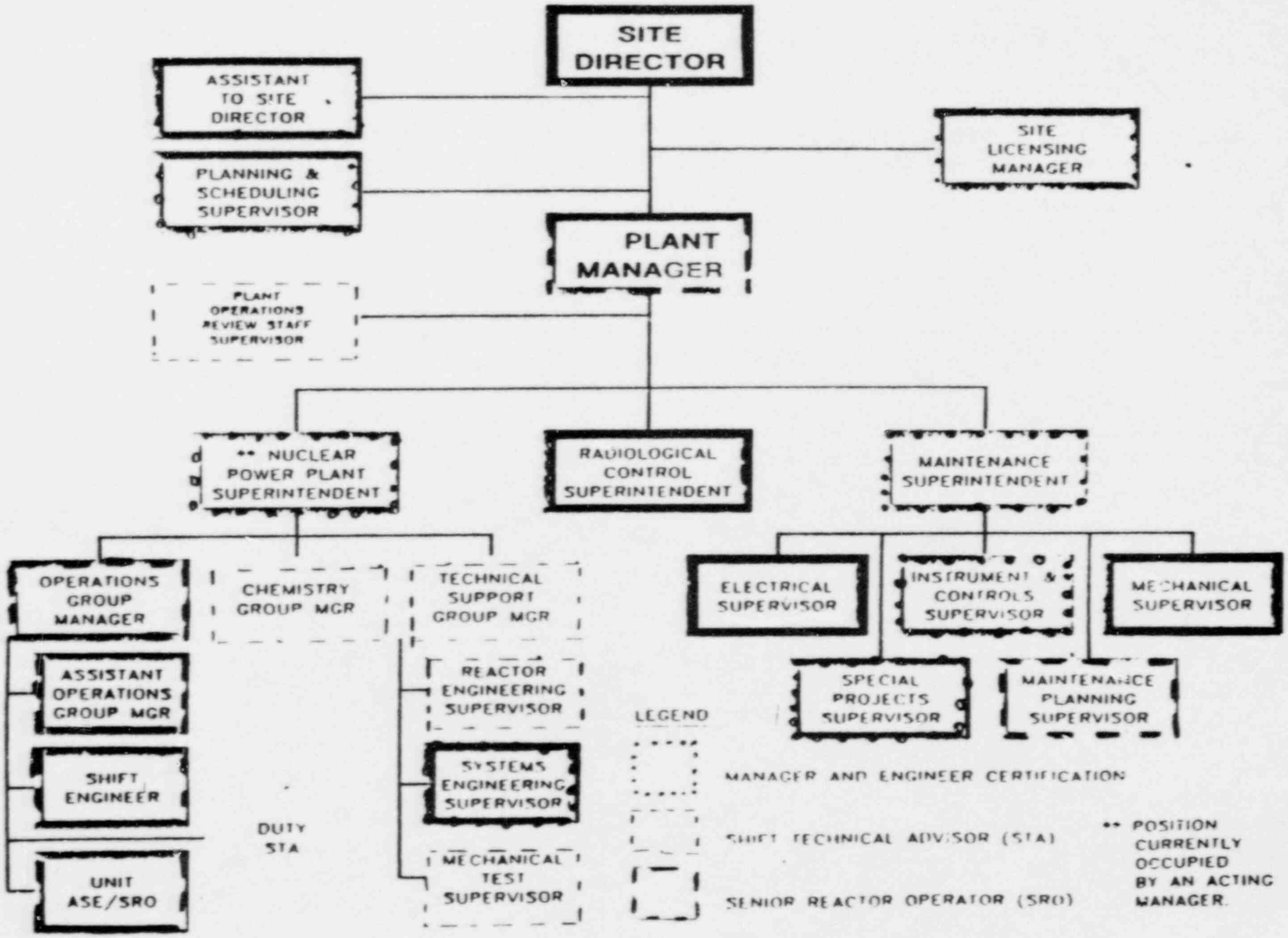


Figure 4.3 Sequoyah staff qualifications

- Nuclear Codes, Standards, and Regulations
- Plant Modification and Work Control
- Plant Systems and Components

After the orientation phase, several types of advanced phase training are available. Procedure 202.17 outlines the typical contents of Segment 1 and Segment 2 and portions of the advanced phase training.

These programs have resulted in increased management involvement in technical training. The staff believes that these programs should contribute to the overall technical and managerial capabilities of the Sequoyah management, thereby enhancing plant safety.

4.2.2.4 Management Goals and Objectives

The level of management involvement in controlling work practices has been inconsistent at Sequoyah in the past. To address this problem, the Manager of Nuclear Power established new goals and objectives, as listed below.

- ensure that Sequoyah has a strong, effective management team with clear lines of responsibility, authority, and accountability

- fully implement required prerequisites for safe operation of Sequoyah

- bring Sequoyah back into operation expeditiously

- conduct operation of Sequoyah in a safe and efficient manner

- create a working environment built on trust and confidence that will permeate the entire organization

Each of the stated goals is supported by several objectives to achieve the goal.

The staff endorses the goals established for Sequoyah. While achievement of these goals would not, by itself, resolve all past problems that have been identified, it would produce an atmosphere conducive to resolving the control of work practices.

4.2.2.5 Communications

Since every employee has responsibility for safety, employees must receive and understand the relevant information. Therefore, the staff endorses the importance of communication channels within SQN organizations as well as between Sequoyah organizations and coprorate offices. Inspection Report 50-327/328 87-59 addresses these issues. In that report, the staff found that the communication channels at SQN are adequate.

4.2.2.6 Procedures

TVA has a program to upgrade all of its procedures to correct documented deficiencies, incorporate organizational changes, and reflect plant modifications. A short-term effort will focus on the technical content and clarity of TVA's nuclear operation and surveillance procedures. TVA intends that the long-term procedures-upgrade program will ensure that recent industry and NRC concerns, such as human factors considerations, are properly addressed.

Long-term and short-term actions are under way to improve the plant procedures. Procedure development or revision is necessitated by (1) documented deficiencies or weaknesses in the existing procedures, (2) results of completed plant modifications and system walkdowns, and (3) changes in responsibilities and authorities as a result of the organizational changes that have been made. The short-term effort will consist of the development or revision of those procedures necessary to support plant restart. Changes that are not necessary prior to plant restart will be handled as part of the long-term procedure upgrade program.

The long-term procedure upgrade program is a corporate-wide effort that will extend beyond restart of a Sequoyah unit. As part of this program, the Sequoyah plant procedures will be incorporated into an overall five-tiered package of policies, directives, standards, procedures and instructions that will govern the operations of TVA's entire Office of Nuclear Power. A Site Procedures Group has been established on a permanent basis at Sequoyah to participate in this long-range program. The group will assist the line organizations in developing and revising site procedures and instructions and will be responsible for scheduling, tracking, editing, verifying, incorporating good human factors practices, and coordinating the review and approval of site procedures.

An interim directive or plan has been issued that provides a description of the overall implementation plan for the TVA Nuclear Procedures System. This plan includes requirements that control both the transition period and the implementation process. TVA has indicated that its nuclear management has placed increased emphasis on compliance with procedures and will monitor compliance. Supervisors must ensure that there are proper procedures in their areas of responsibility. Personnel performing the work must follow the appropriate procedures or initiate management approval for a temporary change to the procedures. The nuclear headquarters staff and the site QA manager are to monitor compliance with procedures when they conduct their plant performance assessment activities.

On the basis of its review, the NRC staff finds TVA's proposed actions acceptable.

4.2.2.7 Management Training

Management training is conducted by the Management Training Branch, which is part of the Division of Nuclear Training, but also reports directly to the Manager, Nuclear Power. The primary functions of the Management Training Branch are listed below.

- develop productive supervisors
- increase utilization of appropriate and efficient supervisory skills
- assist supervisors/managers in moving from a reactive management style to a proactive management style

° facilitate the development of consistent management throughout the ONP

The Management Training Branch personnel document and track training performance of supervisors and managers within ONP. In addition, the Branch evaluates the training as it is conducted and provides feedback to line management.

Each of the core courses is taught by TVA although vendors may be involved in such things as the printed materials. The core courses are Orientation to Nuclear Supervision, Supervisor Development Course, and Managing for Excellence.

On the basis of its preliminary review, the NRC staff finds the management training program acceptable.

4.2.3 Conclusion

On the basis of its evaluations, the NRC staff concludes that TVA has acceptably addressed the Sequoyah-specific management concerns and weaknesses.

4.3 Quality Assurance

4.3.1 Conditions Adverse to Quality (CAQs)

TVA has acknowledged that it had not always taken timely action to resolve conditions adverse to quality in its nuclear activities. This problem included a lack of upper-level management involvement and a lack of timely processing of conditions adverse to quality involving multiple organizations.

TVA took actions to improve performance, including those listed below.

- ° standardization of CAQ reporting and of the method used for determining significance

- ° automatic escalation to higher levels of management when the timeliness or responsiveness at lower levels is inadequate to resolve the CAQ

- ° training of personnel on use the of new CAQ process

- ° frequent status meetings

- ° procedure changes requiring prompt assessment of safety significance when a CAQ is identified

The staff finds that the measures described in the SNPP (Section II.2.5) for handling CAQs are acceptable. NRC inspections (see Inspection Report 50-327/328 87-55) have shown that significant management attention is being directed to this program but that problems still exist that will take time to fully resolve. These problems include additional employee training, accurate problem tracking, and general procedure compliance.

4.3.2 Quality Assurance Program

The TVA organization for quality assurance (QA) that has been in use since mid-1976 is described in a Topical Report TVA-TR75-1 entitled, "QA Program

Description for Design, Construction, and Operation of TVA Nuclear Power Plants." This report contains organization charts, a description of the organization, and the QA responsibility assignments. The staff has reviewed and approved program revisions that have been submitted by TVA. However, although the staff accepted each QA program described by TVA, problems were encountered in program execution, and the staff's systematic assessment of licensee performance (SALP) reports for TVA nuclear activities from 1980 through mid-1985 showed a need to improve QA.

As noted in the revised SNPP, TVA's nuclear QA and quality control (QC) functions had not been effectively unified in a single department. One nuclear QA organization was responsible for conducting corporate-level audits, a separate nuclear QA group within the construction division was responsible for inspecting construction activities, and a third nuclear QA group within engineering was responsible for conducting audits of engineering activities. To further compound the problem, each nuclear site had its own QA group responsible for QA/QC activities at that site. As a result, TVA's nuclear QA activities were not conducted according to a consistent set of programs and procedures, and the QA groups reported to various management groups within TVA, thereby diminishing the visibility and importance of these activities to top-level management. As a result, the staff believes that the QA program has not always been implemented on an effective, consistent basis.

The staff evaluation of TVA's Sequoyah Quality Assurance Program is based on a review of SNPP Section 2.6, "Quality Assurance."

Under the new organization, the responsibility for all nuclear QA/QC functions has been consolidated under the Director of Nuclear Quality Assurance, who reports directly to the Manager of Nuclear Power. This responsibility includes all QA/QC activities related to engineering, construction, and operations, as well as QC inspections of construction and maintenance/modification activities.

A standarized TVA QA program, nuclear quality standards and directives, and model QA procedures for the sites are being developed. The standard nuclear QA program is to be implemented at each site, with site-specific adjustments allowed only if (1) they do not degrade the level of quality provided by the standard program and (2) they are approved by the Director of Nuclear Quality Assurance.

The staff concludes the overall revisions to the TVA nuclear quality assurance program as generally described in the revised SNPP represent QA programmatic improvements and, if properly implemented, are acceptable.

TVA submitted to Region II (May 1986) a revised and upgraded version of its QA topical report for NRC review. The report described the then-current organization and QA procedure system. After a review of the report and a meeting with TVA representatives, the staff forwarded a request for additional information to TVA. TVA revised the topical report to address these staff questions and to fully reflect the organization of the Office of Nuclear Power.

Determining if the changes in the TVA QA topical report will resolve past problems can only be done by observing TVA's performance over an extended period. As noted above, the problems in TVA's nuclear activities occured under a previously approved QA program; however, that program was not implemented the way it was described. Thus, it is important to note that the staff's review and acceptance of the QA topical report means only that TVA's commitments meet the programmatic requirements of 10 CFR 50, Appendix B, as described in Section 17 of the NRC Standard Review Plan (NUREG-0800). The staff will assess whether these commitments are fully and effectively met in its ongoing oversight and inspection of TVA's technical and QA programs. Because of TVA's past problems in the QA area, the Region II staff approved this revision (Revision 9) to the QA topical report on January 30, 1987, for a period of 2 years. The staff's decision on extending the approval of the topical report will depend on how effectively TVA implements the program.

On the basis of its review, the staff finds that the Quality Assurance Program is acceptable as described.

4.4 Operating Experience Improvement

Item C.3 of Enclosure 2 to the 10 CFR 50.54(f) letter requested a detailed description of the Sequoyah Operational Readiness Plan. In response to this request, TVA described operating experience actions (in terms of enhancements made through reactor trip reduction, limitation of spurious engineered safety features actuations, review of the Davis-Besse event for lessons learned, and review of nuclear operations experiences) in the SNPP. Each of these enhancements is addressed below.

4.4.1 Reactor Trip Reduction

From initial criticality to the present shutdown, the number of reactor trips for Units 1 and 2 has been 83 and 53, respectively. To reduce unnecessary challenges to the reactor protection system and increase plant reliability, TVA established a reactor trip reduction program using input from vendor and other nuclear industry organizations.

The staff reviewed Sequoyah's reactor trip reduction program during a special NRC team inspection (Inspection Report 50-327/328 85-46). The program consisted of an evaluation of the areas identified in the Institute for Nuclear Power Operations (INPO) report, "Scram Reduction Practices" (INPO 85-11), dated November 21, 1985. The TVA evaluation addressed in detail each of the INPO items and identified which were being implemented and which were standard practice.

TVA identified 27 trips at Sequoyah that have occurred since January 1, 1984, and categorized them as follows:

° equipment malfunction or failure	13 trips
° manual feedwater control of steam generators	8 trips
° personnel error	5 trips
° inexperience with single element controller for steam generator feedwater bypass regulating valve	1 trip

The root causes of the 13 trips associated with equipment malfunction or failure were identified. Long-term corrective actions consisting of preventive maintenance, design reviews, and posting of warning signs to prevent recurrence were taken for five trips. (No long-term corrective action was felt appropriate for the remaining eight trips.) Structured to reduce these types of equipment malfunction/failure-induced trips, TVA's preventive maintenance program includes the following:

- Critical plant equipment that can cause scrams is inspected and tested during each refueling outage.
- Vendor simulators are used for testing systems.
- Preventive maintenance on important equipment is minimized while the plant is operating.
- Instrumentation and control (I&C) technicians verify that control systems are functioning properly by stroking components through their full range.
- Major equipment performance is monitored so anticipatory corrective action can be taken before a scram.

A design change to provide automatic control of feedwater bypass regulating valves was installed to reduce the trips that occurred from manual control during startup and shutdown evolutions. Additional feedwater system modifications made as a result of the Davis-Besse event will improve the auxiliary feedwater system reliability.

To address those trips caused by personnel errors, TVA has implemented the following additional training:

- I&C technicians receive a half day of systems training each week as part of the continuing training program.
- Simulator training is provided for I&C technicians, engineers, and certain maintenance personnel based on availability of stimulator.
- Newly hired technicians must complete a certification program that includes procedures, policies, system training, and practical factors. Certification must be completed satisfactorily before a technician performs unassisted testing.
- On-the-job training is conducted by a foreman as part of the training/qualification process.
- Vendor training programs are used for critical plant equipment (e.g., electro-hydraulic control, governors, and motor-operated valves).
- Operations personnel receive training on plant modifications before new equipment is placed in service. (Single element feedwater controllers have been added to the Sequoyah simulator and are used during operator retraining.)

- Trainees, including available auxiliary operators, observe and, in some cases, receive hands-on experience during such plant evolutions as startup, synchronization, and shutdown in the control room.

- Operations personnel are given additional in-depth training on balance of plant equipment.

TVA has implemented the following practices to reduce plant trips through increased personnel responsibility and enhanced root-cause determinations:

- beginning to assign a system engineer to be responsible for each plant system

- performing a comprehensive post-trip review for each reactor trip

- delaying startup until a multi-discipline committee reviews the trip to determine the cause and implementation of corrective actions (A historical data base is maintained to allow analysis and trending by scram cause codes.)

- participating in the Westinghouse Owners Group, which has a program for investigating each scram

As documented in the SNPP and the special NRC Inspection Report 50-327/328 85-46, TVA has taken positive steps to improve plant reliability through trip reduction. Based on its review of the SNPP and the information gained from the special inspection, the staff has concluded that the actions taken by TVA to reduce reactor trips are acceptable.

4.4.2 Limitation of Engineered Safety Features (ESF) Actuations

To reduce unnecessary challenges to safety systems and maintain system availability, TVA has an established program to limit spurious/unnecessary ESF actuations.

In 1985, the number of spurious unnecessary ESF actuations was significantly reduced from the number that occurred in 1984. The main contributors to the number of ESF actuations historically have been containment ventilation isolations and auxiliary building isolations caused by spurious and inadvertent radiation monitor high radiation trips. To reduce the number of isolations, TVA initiated several actions.

One of the actions taken by TVA was to have the Chemical Engineering Section revise the sampling instructions to coordinate activities with operations, and to block the applicable radiation monitor channel before changing filter paper or obtaining air samples. Additionally, proper sample flow on monitors is verified once per shift, thereby limiting spurious high radiation actuations due to sample flow switch actuation from low flow conditions.

Other actions have included raising the set points for the noble gas channels of the upper and lower containment monitors from 20 percent to 40 percent of the allowable value of the technical specification. NRC has approved a technical specification change to raise the set point of the fuel pool radiation monitors to further reduce the number of auxiliary building isolations caused by movement of contaminated trash and elevated background radiation levels.

Time delay relays have been incorporated on the vent monitors in the containment, control room, auxiliary building, and fuel pool to reduce the impact of short-duration electrically induced spikes on these radiation monitors. The General Atomic RP-30 radiation analyzer has been modified on the noble gas and air particulate channels to operate with an upper level and a lower level discriminator, and radiation monitor signal cables have been installed in conduit on all ESF and effluent radiation monitor channels.

The spurious and unnecessary ESF actuation reduction program has been effective in reducing the number of actuations caused by electrical noise. Although the program has been less effective in reducing personnel errors during testing activities, continued upgrading of the implementation of this portion of the program will help to increase its effectiveness.

On the basis of its review, the staff finds TVA's program to limit spurious and unnecessary ESF actuations acceptable.

4.4.3 Review of Findings From Davis-Besse Event

TVA assigned a task team to evaluate NRC Generic Letter 85-13, which transmitted NUREG-1154 in response to the staff's findings of the June 9, 1985 Davis-Bessie event, and an INPO report entitled "The Operational Performance of Auxiliary Feedwater (AFW) Systems in U.S. PWRs 1980-1984."

A special NRC inspection team (Inspection Report 50-327/328 85-46) reviewed TVA's evaluation of NUREG-1154 and the INPO AFW report. TVA's evaluation addressed the significance of the Davis-Besse loss of main and auxiliary feedwater event with respect to Sequoyah. TVA used the INPO report to review the Sequoyah AFW system for problems that have been experienced by other utilities. As discussed in the SNPP, the nine major topics from the Davis-Besse event that were evaluated are listed below.

° interaction of plant security features and operator actions

° availability of shift technical advisors (STA)

° reliability of the AFW containment isolation valves and other safety-related valves

° reliability of AFW pump turbines

° reliability of power-operated relief valves

° adequacy of control room instrumentation

° adequacy of plant procedures

° adequacy of safety system testing

° acceptability of current safety assessment methods

The NRC inspection team confirmed that the interaction of plant security features and operator action problems that occurred at Davis-Besse would not have occurred at Sequoyah. Additionally, the STA would be available at Sequoyah during such an operational event.

Unlike Davis-Besse, Sequoyah's AFW system does not have any containment isolation motor-operated valves (MOVs). However, reliability problems with other MOVs in the AFW system, as well as with the main feedwater isolation valves, have occurred as a result of improper limit switch settings. TVA is implementing increased MOV maintenance, and the motor-operated valve and test system (MOVATS) is being used to adjust limit switch settings.

Operator training sessions have been conducted with the Unit 1 turbine driven auxiliary feedwater pump covering problems experienced by operators during the Davis-Besse event, and a laminated sign has been installed near the turbine throttle valve with a drawing of mechanical overspeed trip. Management has indicated that all operators will receive training of a similar nature before startup of either unit, and annual simulator training on a complete loss of feedwater (normal and emergency) has been implemented.

Sequoyah surveillance programs provide some assurance of operational readiness of the power-operated relief valves (PORVs). However, TVA does not support the automatic block valve closure suggested in NUREG-1154 as a potential remedy for PORV failures. The acoustical monitoring instrumentation for both units is located in the common area of the control room, approximately equidistant from the Unit 1 and Unit 2 controls. TVA has evaluated the adequacy of the location of the acoustic monitors and the pressurizer tail pipe temperature indicator during the detailed control room design review. TVA will relocate the acoustic monitors to the panels that contain the tail pipe monitors. The staff's safety evaluation report assessing the adequacy of the control room design review and TVA's corrective actions was issued on August 27, 1987.

TVA's evaluation of NUREG-1154 shows that the Davis-Besse event should not occur at Sequoyah because of several differences. Sequoyah's design provides two motor-driven and one turbine-driven AFW pump per unit, as opposed to Davis-Besse's two turbine-driven pumps. Also, Davis-Besse only has two steam generators where Sequoyah has four, with only one required for decay heat removal. Additionally, Sequoyah does not have an automatic system like Davis-Besse's steam and feedwater rupture control system, which could allow a single operator error to totally isolate AFW. Total isolation of AFW at Sequoyah requires several deliberate manual operations for each AFW pump and could not be accomplished by a single operator error.

On the basis of its review, staff finds the TVA actions in response to Generic Letter 85-13, combined with its AFW reliability improvement program, are acceptable.

4.4.4 Review of Nuclear Operations Experience

In January 1985, TVA transferred the responsibility for experience review to the site. At that time, Sequoyah assigned the Site Services Group the function of handling such items and made several program improvements.

The staff reviewed the nuclear operations experience feedback program during a special NRC team inspection (Inspection Report 50-327/328 85-46). Improvements to the program include the following:

- The procedure covering experience review was rewritten.
- A formal computer system was initiated for the tracking experience review items at the site until they are closed out.
- A system was set up for putting together "packages" on Sequoyah events and making this information available to other utilities and TVA plants.
- Provision was made for consolidating site experience review information into one file. This will facilitate the tracking of evaluations and other actions on individual review items.
- Provision was made for Sequoyah items to be routinely communicated to other TVA nuclear facilities for their experience review, including nuclear network releases on Sequoyah events, licensee event reports, and studies that may be applicable to other plants (e.g., auxiliary feedwater study). When items are received from other TVA nuclear facilities, they are evaluated on site for applicability to Sequoyah.
- Provision was made, since September 1985, for the Sequoyah plant manager to participate in a regular weekly conference call with the Browns Ferry and Watts Bar plant managers. The purpose of the call is to exchange information on operating experience, programs initiated, and other activities at this management level.

Inspection findings have indicated that the revised procedure is vague on how the operating experience received outside TVA is being processed to different departments within the Sequoyah organization. However, TVA's Division of Nuclear Services receives operating experience information from outside the TVA system, such as NRC generic letters, information notices, and bulletins and INPO reports and vendor letters. This information is then routed to various departments, including the TVA Training Center, and to the training shift engineer. A sampling of this process indicates that the information is being provided to the operators.

TVA's SNPP addresses measures to improve dissemination of information on operating experience. A site nuclear experience review program (NERP) has been established as part of the corporate program managed by Nuclear Licensing and Regulatory Affairs. The site licensing organization interfaces with the corporate NERP to disseminate information to operations and engineering departments. The training department reviews operating experience items to incorporate them into the training programs.

Based on a selective sample review of TVA's operator experience feedback process, it is apparent that the necessary information (e.g., operating experience reports and plant modifications is being provided to the operators. These program improvements should enhance the program. Accordingly, the staff considers the feedback program acceptable.

4.5 Post-Modification Testing

Past NRC inspections have identified problems with respect to the adequacy of testing on systems and components following modification. TVA assembled a task force to review the Sequoyah post-modification testing (PMT) program. The task force examined 124 completed engineering change notices (ECN) to check the testing that was performed. In addition, TVA committed to review all Unit 2 or common ECN packages associated with the systems that are within the scope of Phase I of the design baseline verification program (DBVP) that have been issued since Unit 2 received its license. These TVA programs are discussed in Section III and Appendix 2 of the SNPP.

The staff inspected modification testing July 28 through August 1, 1986 (Inspection Report 50-327/328 86-43). Two violations were identified with respect to failure to specify a required surveillance test in the work package and improper changing of PORC-approved procedures. In response to the PMT task force review and the NRC notices of violation, TVA has improved its plant procedure on PMT. Training also was conducted on specification of correct testing in the work plans.

TVA conducted a review of all work plans issued after the post-modification task force and the DBVP reviews and identified a total of 117 modifications that will need additional testing to document functional operability. The staff is following the scheduled testing as discussed in Inspection Report 50-327/328 87-30.

The staff subsequently conducted a reinspection which examined 16 DBVP system evaluation reports for adequate screening by TVA of work plans and ECNs (Inspection Report 50-327/328 87-18). While isolated deficiencies were identified, the staff's overall conclusion was that the licensee had adequately determined testing requirements for previous modifications.

The staff concludes that the programs instituted by TVA to address post-modification testing are acceptable.

4.6 Surveillance Instruction Review

4.6.1 Introduction

Staff reviews and audits of Sequoyah surveillance instructions (SIs) identified technical and administrative weaknesses in these instructions. To remedy these weaknesses, TVA has undertaken a comprehensive and disciplined program to review and revise these instructions. The program has undergone several evolutions since it was initiated in the summer of 1986. These changes have resulted in increasing the technical and administrative depth of reviews, the scope of reviews, the independent evaluations of the process and its products, the field verification of SIs and their supporting instructions, and the technical content and specificity of SIs. The staff has evaluated the program that has existed since January 1987, which includes the improvements and was discussed in TVA's March 24, 1987 submittal and in Section II.2 of the SNPP.

4.6.2 Evaluation

The staff assessment of the descriptive material providing the basis for the TVA program to review and revise certain Sequoyah Unit 2 SIs that implement technical specification surveillance requirements before restart included the scope, methodology and organization of TVA's surveillance review and revision program. The staff also conducted inspections in this area as discussed in Inspection Reports 50-327/328 87-36 and 87-50.

The basic objective of the SI program is to ensure all technical specification requirements are addressed and that the SIs and their supporting instructions covered by the program scope are technically adequate to fulfill the surveillance requirements of the technical specifications, have an appropriate level of dependence on the skill of the performer of the instruction, and comply with basic administrative requirements that make performance of the SI reliable. This program will be completed before Unit 2 restart.

Although the staff concurs with TVA's objectives, TVA should define the skill level required to write, revise, and review the surveillance instructions and supporting procedures and TVA should describe, including starting and completion dates the long-term program which will be undertaken to ensure complete administrative consistency, achieve standard format and organization and make other improvements and enhancements as are determined to be needed.

The scope of TVA's review program includes those technical specification SIs and supporting instructions that are required for startup, operation, and safe shutdown of Sequoyah Unit 2 to the point of the next refueling. The licensee noted that the criteria for determining which instructions would not be included in the SI program prior to restart were provided in a memorandum separate from the SI review program. During NRC inspections, the staff reviewed those procedures not in the restart scope and did not identify any cases which were considered necessary for restart.

TVA has indicated that some instructions that are not required for startup and operation will be reviewed using the latest SI-1 Appendix F (Part 1) checklist to confirm that the instruction was adequate for its last performance; this review will be completed before restart. If this review indicates that the instructions are not technically adequate to verify equipment operability, these instructions will be revised and another review performed before restart.

The program methodology and the governing organization, required training and qualification, and instruction validation and verification are discussed below with staff comments, as required.

The program is under the control of the Plant Manager, and it is implemented by the established plant organization under the day-to-day direction of the Instruction Review Project Manager.

The site procedures staff performs the functions of typing (word processing), process control to move the revised instructions through the various parts of the cycle, and process tracking to maintain visibility of progress. The Plant Manager approved list of instruction to be reviewed and identifies any deficiencies for tracking to ensure that they are resolved.

Whichever section is responsible for a particular instruction performs the review, produces the revised instruction, and validates it. In some cases the validation of the instruction is performed by a section other than the section responsible for instruction preparation because the second section is normally responsible for performance of the instruction.

The Technical Support Section (TSS) ensures that personnel are appropriately trained to perform the review in accordance with the established checklist and that the review is properly documented. This section ensures that there is an SI to satisfy each applicable technical specification surveillance requirement; in cases where the surveillance requirement is satisfied by more than one instruction (each instruction partially satisfying the requirement), this section ensures that the group of instructions fully satisfies the surveillance requirement.

The staff concludes that since most of the personnel performing the review had previously approved questionable instructions, it is appropriate for TVA to specify the training/screening process used to ensure that reviewing personnel have adequate systems knowledge and expertise in their assigned areas.

An independent review group (IRG) is responsible for verifying that the checklists used to determine the need for instruction revision have been properly completed and for verifying that the reviews are performed by trained and qualified individuals. The IRG ensures that updated drawings are appropriately reflected in the SIs whenever these affect the instruction. The IRG also conducts independent technical reviews of a sample of the revised procedures to ensure that program objectives are being met. The IRG selects the instructions to be reviewed so that representative instructions are sampled, but the IRG also may perform the functions of qualified individual reviews while performing such independent reviews. As of the first week in March 1987, the IRG had performed 186 independent reviews for the primary purpose of identifying deficiencies in the detailed process and approach being used by the responsible section. These reviews were conducted at various stages in the section revision and approval process so the problems could be remedied at the earliest possible time. The IRG provided written comment to each responsible section.

The future activity of the IRG will concentrate on review of instructions after they have been released by the responsible sections into the approval cycle. The IRG will review about 10 percent of the instructions introduced into the approval cycle.

The site quality assurance (QA) organization reviews instructions during the Plant Operations Review Committee (PORC) approval cycle and performs program surveillance. In addition, QA is performing technical reviews of selected instructions to ensure that the program is achieving its objective. Since the program began in the summer of 1986, QA performed technical reviews of instructions in various stages of the revision and approval process and determined that program changes were necessary; its comments were provided and changes were implemented. To provide additional assurance that the program objective is being achieved, QA will perform a technical review of at least 10 percent of the instructions that have been submitted for PORC approval.

The program calls for a detailed checklist to be used during the technical review of an instruction to identify technical deficiencies. Part I of this checklist focuses on the technical adequacy of the instruction, with an operability evaluation being performed only if the instruction is found to be technically inadequate. Part II of the checklist focuses on the administrative adequacy of the instruction, but all items within this section do not need to be fulfilled to ensure instruction adequacy. Part II of the checklist does not have to be completed for this program. Certain items in Part II of the checklist, such as SRO approval to perform the test and verification or double verification signoffs, stem from other documents and are checked to ensure necessary compliance.

A number of sources identified instruction deficiencies that needed to be remedied. These sources include INPO reviews, NRC inspection reports, employee concern reports, QA deficiency reports, corrective action reports, conditions-adverse-to-quality reports, and audit reports. These deficiencies are listed and tracked by the site procedures staff in a temporary tracking system. This staff ensures that these deficiencies are satisfactorily resolved, as appropriate, when the instruction is revised. Such deficiencies include correct identification of site organization and organizational responsibilities.

The developed checklist is used during the training for personnel performing the reviews. Most of the involved individuals (about 80 percent) received this training on December 10, 1986. The remaining personnel received training using the training package at other times. The list of trained personnel is maintained by the IRG and is used to ensure that the evaluations of instructions using the checklist are performed by these personnel. This appears inconsistent with the description of the duties of the IRG in the organization description where it is stated that the TSS will ensure appropriate reviewer training takes place and is documented. It is not clear why the roles of the two groups are indicated this way. The staff believes that IRG also should verify that training has taken place to ensure that the reviewers are indeed trained.

Reviews of the procedures revised by the responsible sections are performed as part of the onsite independent review, as specified in Sequoyah Nuclear Plant Standard Practice SQA21. SQA21 lists the organization members of the PORC and identifies the qualifications of individuals who may function as qualified individuals in the performance of review. The appendix to SQA21 lists the individuals (by name) who meet the requirements and have been approved by the Plant Manager as qualified to perform qualified-individual reviews.

Validation and verification are important activities that help to ensure that the program objective is accomplished. The fundamental purposes of the validation and verification activity are to ensure that the instruction is correct and accomplishes the intended purpose, that the instruction is clear to the performer, that it is written to a sufficient level of detail, that the plant equipment and instruction identifications are consistent, and that the instruction can be accomplished by the performer without reference to information or consultation with personnel not indicated in the instruction.

The technical specifications do not permit full performance of a surveillance instruction that involves manipulation of equipment and changes in critical safety system components (CSSC) configuration until the instruction has been approved. TVA has reviewed the plant conditions and technical specifications and has not found a reasonable justifiable approach to satisfy this interpretation and constraint. In addition, there are some instructions that cannot be validated by performance until applicable plant conditions and configuration are attained.

TVA has adopted a progressive validation and verification approach that obtains the best validation and verification permitted by plant conditions and the approval status of the instruction. During the latter stages of instruction preparation, the responsible section will perform or has performed nonmanipulative walkdowns to confirm that the instruction is correct. Once the instruction has been through the approval cycle and appropriate plant conditions are attained, the responsible section will perform or has performed a validation by actual performance. TVA anticipates that performance may involve temporary changes in the instruction because some deficiencies may not reasonably be discovered without performance at requisite conditions. Any such changes will be made according to approved procedures. This would only be acceptable to the staff if, after the problems were resolved, those temporary changes necessary for performance of the surveillance instruction were permanently incorporated into the affected instruction, the revised instruction is approved by PORC, and the newly revised/approved instruction is then performed satisfactorily out in the field.

In addition to the validation and verification activities described above, this program involves an independent sample review of SIs by personnel not involved in their preparation, review for approval, or performance. These personnel will review a 20-percent sample of the SIs for clarity and completeness, and they will observe the validation (walkdown or performance) of at least 10 percent of the instructions in the field to help ensure that they are performed as written. The guidelines for this activity are drawn from "Procedures Evaluation Checklist for Maintenance, Test and Calibration Procedures Used in Nuclear Power Plants," NUREG/CR-1369, Revision 1 (September 1982). This activity also has a progressive character as a result of plant configuration limitations, but it will be performed only with approved instruction. The staff has determined that the program should clearly indicate the necessary qualifications of personnel who will be used to perform this independent sample review of SIs. Since such personnel are not involved in the preparation, review for approval, or performance of these surveillance instructions, the program should explicitly define the persons allowable relationship to the surveillance instruction and the required level of training and expertise for these ancillary reviewers.

4.6.3 Conclusions

On the basis of its review and the NRC inspections, the staff concludes that the Surveillance Instruction Review and Revision Program has produced adequate procedures to support Unit 2 startup. However, the staff believes that the program for long-term control of surveillance instruction upgrades, including resolution of the issues of temporary changes, qualification of reviewers, and schedule, needs to be provided to completely resolve this issue.

4.7 Operability "Look Back"

As a result of violations regarding the adequacy and timeliness of corrective actions for repetitive equipment failures and out-of-tolerance conditions, the licensee implemented a trending and tracking program at Sequoyah (see also Section 4.8, Maintenance). Because this program was geared toward identifying future deficiencies, the staff raised concerns regarding potential operability questions resulting from past, undetected, repetitive failures. TVA committed to conduct an operability "look back" review, as described in its submittal of December 12, 1986.

The operability look back program was designed to identify adverse conditions associated with equipment operability, to evaluate the safety significance of these conditions, to document the effectiveness of corrective actions, and to propose further corrective actions where necessary. Data was collected from maintenance-related potentially reportable occurrences (PRO) and from interviews with senior plant engineers. The review process identified 44 conditions with corrective action recommendations requiring resolution before restart. An additional 163 issues were identified for corrective action after restart.

NRC inspection and assessment of the Sequoyah operability look back review program was performed the week of April 27, 1987 and is documented in Inspection Report No. 50-327/328 87-24. The inspection staff performed a detailed review of the issue summary packages for two systems, selected reviews in other areas, as well as a review of 44 restart items. Additionally, interviews were conducted with the reviewers and other plant personnel.

The staff concluded that the scope, guidelines, and implementation of the Sequoyah operability look back review program satisfactorily accomplished its intended purpose.

4.8 Maintenance

4.8.1 Introduction

Previous NRC inspections at TVA nuclear units indicated programmatic deficiencies in the site maintenance programs. These findings are documented in systematic assessment of licensee performance (SALP) reports for the TVA nuclear plants and in numerous inspection reports. TVA conducted a detailed review and reassessment of maintenance performance beginning in March 1987. The review, including findings contained in the most recent SALP report, NRC notices of violation, licensee event reports (LERs), the latest INPO evaluation, internally identified findings, and applicable Davis-Besse issues. These programmatic deficiencies have been attributed to (1) management problems in the development and administration of appropriate controls for maintenance of nuclear safety-related equipment, and (2) the failure to implement effective and timely corrective action when problems have been identified.

In Revision 1 to the SNPP, TVA discusses specific problems identified by the NRC and TVA that have existed at Sequoyah. These deficiencies include failure to implement appropriate preventive maintenance programs, failure to provide adequate planning of maintenance activities, and inadequacies in the training programs for the corporate and site personnel involved in maintenance activities.

To further assess the maintenance programs at Sequoyah, TVA's Manager of Nuclear Power directed TVA's Nuclear Manager's Review Group (NMRG) to conduct a comprehensive assessment of corrective and preventive maintenance practices.

As a central part of their corrective actions, in part to address the results of this report, TVA will increase maintenance management involvement by stressing personnel accountability. This will be accomplished through

- better review and improvement of maintenance procedures
- placing emphasis on trending equipment failures and preventive maintenance
- requiring improved training of craft personnel
- monitoring and use of established performance indicators

4.8.2 Evaluation

The NRC staff evaluated the scope, organization, and methodology of TVA's maintenance program and found it to be adequate.

NMRG Study Findings

The NRC staff has reviewed the scope and findings of the NMRG study of Sequoyah maintenance and finds that it was a comprehensive evaluation of the maintenance programs at the TVA sites and corporate offices. The performance areas reviewed were based on those identified in the INPO guidelines for the content of maintenance at nuclear power stations and included competent programmatic reviews and field observations of maintenance activities. The staff noted that the findings of the NMRG study closely parallel those findings identified by NRC inspections.

The NMRG study states that the most significant improvement areas needed, included the aggressive correction and prevention of hardware problems, corporate involvement in nuclear maintenance, and implementation of challenging goals and objectives for maintenance. The discussion on correction and prevention of hardware problem cites the diffusion of responsibility for maintenance control and checks, the lack of aggressive and coordinated efforts to solve problems and a lack of clear accountability for solving specific problems.

Specifically, the NMRG found deficiencies in corporate involvement in the maintenance program, inadequate training and qualifications of planners, preventive maintenance program deficiencies, inadequacies in maintenance instructions and the performance of instructions and work requests, deficiencies in the planning and scheduling of maintenance, inadequate control of maintenance activities, failure to provide adequate post-maintenance testing, problems with materials suitability, inadequate control of maintenance tools and equipment, lack of management involvement in ongoing maintenance activities, incomplete maintenance history programs, a failure to use trending techniques to guide maintenance, ineffective quality assurance reviews of maintenance, and a lack of follow-through on corrective action for maintenance deficiencies.

TVA's maintenance plan addresses the findings of the NMRG report and also addresses the role of SQN plant management in emphasizing adherence to SQN procedures. TVA's actions to address the NMRG findings are discussed below.

- Agressive correction and prevention of hardware problems

  SQN has reviewed the technical specifications and the FSAR for maintenance requirements; corrective action for deficiencies noted in the maintenance program are being implemented (SNPP Section II.4.3.2). This will be completed before restart.

SQN has established a Maintenance Planning Section under the Maintenance Superintendent to plan, coordinate, and prioritize work (SNPP Section II.1.2.2). Administrative controls have been strengthened to ensure that preventive maintenance is performed as planned.

Training will be provided to maintenance planners on post-maintenance testing that will enable the planners to specify adequate post-maintenance testing requirements to ensure equipment operability. This training also will provide instruction on determining the required level of detail needed in maintenance plans and instructions. Training will be completed before restart.

SQN's long-term approach to correct deficiencies includes the following: (1) SQN will hire outside specialists to assist in a complete update of the preventive maintenance program, which is expected to extend over at least 2 years. (2) A master plan will be developed to address space and equipment needs for the maintenance groups by March 1988. (3) A structured training program will be developed and implemented for maintenance planners that will develop the requirements and skills for planners. This will ensure that new and existing planners can capably develop and issue work instructions. (4) Finally, SQN has hired a Preventive Maintenance Manager who reports to the Maintenance Superintendent and is responsible for implementing and improving the preventive maintenance program.

The NRC staff agrees that completion of these actions will help to correct and prevent hardware problems through increasing resources dedicated to maintenance and better equipping the maintenance organization to handle day-to-day maintenance activities.

° Corporate involvement in nuclear maintenance

TVA corporate management is dedicated to providing more corporate direction for nuclear maintenance and establishing a viable preventive maintenance program.

A position has been established for a corporate Nuclear Maintenance Manager. This manager will be responsible for developing and implementing improved maintenance programs and policies at all TVA nuclear plants. Knowledgeable maintenance personnel from all nuclear sites will contribute to these maintenance improvement efforts under the guidance and direction of the corporate Nuclear Maintenance Manager. Although each nuclear site will remain responsible for planning, scheduling, and executing its own maintenance, the corporate Nuclear Maintenance Manager will be responsible for regular assessment of the effectiveness of site maintenance and for assisting site maintenance personnel with needed improvements.

Significant corporate-initiated improvements have been planned. These improvements will emphasize reducing recurring corrective maintenance, improving use of preventive maintenance, and adherence to established preventive maintenance routines.

The full scope of significant maintenance activities will be defined before performing the activity; will be coordinated with the appropriate organizations, including Operations and Quality Assurance; and will be completed and documented before closeout of the activity.

Enhanced training for planning and scheduling personnel will be developed and implemented. This training will include training on the selection of proper safety classifications for maintenance work and identification of proper post-maintenance testing.

The staff agrees that a centralized corporate Nuclear Maintenance Manager can contribute to an effective maintenance organization. The staff is particularly interested in how the site maintenance personnel will interface with this corporate nuclear maintenance organization. Results of the corporate Nuclear Maintenance Manager's efforts will be reported regularly to the Manager of Nuclear Power.

- <u>Implementation of challenging goals and objectives for maintenance</u>

  Corporate standards and goals for maintenance are being established to measure the effectiveness of each plant's maintenance program. Action plans are being developed to achieve corporate maintenance goals, to assist in the prioritization of maintenance activities, and to accomplish corporate objectives.

  In addition to addressing the concerns identified by the NMRG, SQN plant management is stressing management dedication to procedure adherence. Plant directives and procedures will be issued by the Site Director that require management involvement in the work place.

  The NRC staff agrees that the implementation of these goals and objectives should result in improved equipment performance and reliability. These actions should contribute to the safe operation of Sequoyah.

<u>Sequoyah Management Involvement</u>

The Sequoyah site management has determined that a common root cause for many of the issues is inadequate management involvement and the resulting failure to establish consistent accountability for work performed. Actions to correct this problem include

- increasing management attention and oversight of craft work

- providing increased training to craft personnel on QA requirements, the maintenance work control system, clearance procedures, temporary alterations, and procedural adherence

- increasing accountability by having had the Maintenance Department implement and continue to use an improved program for employee performance reporting

- implementing a new Maintenance Request System that includes establishing a Maintenance Planning Section and providing additional detail for the work request tags/cards

These actions should programmatically help to focus management's attention on factors that have in the past contributed to maintenance program weaknesses. However, management must aggressively pursue its attention to and oversight of the maintenance program.

Maintenance Instruction Enhancement

A writers guide has been included in SQN Plant Procedures (SQM-1) and all maintenance procedures submitted to Sequoyah's work processing group after June 30, 1986, are in accordance with the writer's guide and SQM-1. Cumbersome maintenance instructions will be replaced with stand-alone instructions and procedures with a series of steps will be minimized. Generic maintenance instructions will be incorporated into specific procedures. Experience and improved procedural quality also will be incorporated into procedures as they are updated. Craftsmen will be instructed to review maintenance instructions with their foremen, to list any suggestions to improve the instructions for future use and to prepare new maintenance instructions for major maintenance work related to critical safety system components (CSSC) equipment.

Maintenance instruction clarity, consistency, and accuracy are of paramount importance in a successful maintenance program; implementing these enhancements should improve the maintenance procedures.

Long-Term Preventive Maintenance

TVA has embarked upon a systematic effort toward shifting maintenance emphasis and resources from corrective maintenance and short-term operations support to proactive, long-term preventive maintenance for Sequoyah. This effort will be focused through:

° Efforts to improve preventive maintenance, which include increasing supervisory personnel within the Mechanical Maintenance Engineering Section, continuing to use the Plant Vibration and Diagnostic Unit, establishing a Maintenance Trending and Environmental Qualification (EQ) Section, increasing electrical maintenance participation in the development of preventive maintenance instructions, performing detailed review of the technical specifications and FSAR to ensure that maintenance requirements for preventive maintenance are identified, and establishing a Reliability and Performance Branch within Design Nuclear Engineering.

° Establishing significant enhancements in the area of motor-operated valves (MOVs), which includes developing a comprehensive safety-related MOV program for visual inspection, lubricating and testing Unit 2 MOVs during the Cycle 3 outage, forming a composite crew with cross disciplinary experience to perform maintenance on MOVs, and developing a history data base for each valve. The MOV testing and maintenance program is based on the motor-operated valve automated test system (MOVATS) and uses equipment and training of personnel provided by MOVATS, Inc.

° Providing better control of measuring and test equipment (M&TE) by assigning primary responsibility for control of out-of-calibration M&TE to the site services organization that maintains a computerized data base for M&TE and providing each maintenance group with a qualified individual to perform the M&TE out-of-tolerance evaluations.

A proactive, long-term preventive maintenance program is essential for an effective maintenance effort at a nuclear facility. The NRC staff views positively TVA's efforts to shift maintenance emphasis and resources from corrective maintenance and short-term operations to a proactive, long-term maintenance program.

Maintenance Training

Sequoyah management is fully committed to upgrading its maintenance training programs by seeking INPO accreditation. The instrument maintenance training program was accredited in January 1987. Mechanical and electrical maintenance training programs were accredited May 7, 1987.

Mechanical craft personnel have completed training on Limitorque actuator maintenance, emergency diesel generators, systems familiarization, air compressors, bearings, rigging and various pumps and valves. Electrical craft personnel have completed training on Limitorque actuator maintenance, emergency diesel generators, ac and dc motors, control circuits, generators, and M&TE.

INPO accreditation of TVA's entire safety-related maintenance training programs provides an adequate basis for NRC staff acceptance of these programs.

Additional Maintenance Restart Activities

The Sequoyah Operations staff will review the pre-start checklist of surveillance instructions, system status files, configuration logs, and TACF logs to determine the status of plant systems as required by general operating instructions (GOI), GOI-1 and GOI-2. SI-604, "Essential Instrumentation Operability Verification," also will be performed by the instrument maintenance group to ensure that the essential surveillance instrumentation needed to monitor plant processes during normal operating conditions is available and operable. The Maintenance Department will also review outstanding maintenance requests on safety-related equipment to ensure that unworked items will not degrade equipment or impede operator action necessary for safe operation of the plant.

To assess the reliability of technical specification equipment, potential reportable occurrences (PROs) initiated for equipment failures that occurred between January 1984 and December 1985 were reviewed to determine if the corrective maintenance performed was adequate to prevent recurrences. Ten items required additional action; all will be completed before restart.

In addition to these initial efforts, Sequoyah has performed an evaluation of plant equipment operability. This effort included evaluating PROs associated with the plant maintenance sections and interviewing plant managers, senior engineers, and senior reactor operators. The evaluation of the PRO history files provided assurance that equipment deficiencies identified therein, from the beginning of the PRO program until the start of this evaluation, had been properly dispositioned. The interview process provided input from senior plant personnel with years of experience in operation, testing, and maintaining plant equipment. These two processes together provide a high level of confidence that any deficiency with safety equipment was identified and properly dispositioned. This review of plant equipment operability has been completed and items identified as required for restart will be scheduled and completed before restart.

4.8.3 Conclusions

The NRC staff has conducted a series of maintenance inspections at Sequoyah to ensure that TVA has identified the programmatic problems and is taking adequate corrective action to correct the deficiencies. The staff has inspected the actions TVA has taken to correct the deficiencies related to the restart of Sequoyah. The inspections included an evaluation of the program as outlined in this SER and an assessment of the current status of the Sequoyah maintenance program as well as a review of corrective actions for NRC open items and a review of status of SNPP commitments and NMRG findings.

The staff concludes that significant progress has been made in improving the maintenance area. The structure of the maintenance organization has been evaluated and numerous constructive changes in the maintenance organization have been accomplished.

TVA engineering and management staffs have devoted many staff hours to identifying the problems in the maintenance areas and finding solutions to these problems. Management interest in improvements has been shown by the dedication of management resources to this area, including additional staff, additional time spent in plant staff engineering reviews, and additional management effort dedicated to reviews such as the NMRG study and equipment operability study (OES). Support of management iniatives is indicated in the dedication of the plant and corporate staff to achieve improvements.

During recent inspections the staff determined that TVA had spent significant resources in resolving the issues that have been identified by the NRC, NMRG, employee concerns program, and other review groups. The staff confirmed during recent inspections that the plant has issued a comprehensive action plan for resolution of the NMRG findings and has established tracking systems for restart and long-term issues.

In addition, progress has been made in establishing effective programs for preventive maintenance and corrective maintenance and in establishing clear assignment of responsibility and accountability.

Through interviews and reviews of resumes, the staff observed that managers in the maintenance area are well qualified and are aware of their responsibilities in the implementation of the maintenance program. The staff also observed that upper management, both plant and corporate, supports the implementation of corrective and enhancement efforts.

The staff noted that managers do not adequately address long-term program development and that improvements are needed in time management, interface with support groups, and stabilization of the corporate organization.

Interviews indicate that TVA has taken the first steps in resolving these problems as evidenced by:

(1) TVA has conducted a time study of managers at the plant and has identified problem areas. It is the staff's understanding that this study involved evaluations of management skills, work processes, climate and stress factors, facilities and tools and that a report with recommendations on

improving the utilization of management talent will be provided to TVA in the near future and evaluated by TVA for corrective actions.

(2) The staff noted that the maintenance management appears to be working with support groups to establish effective interfaces as evidenced by management planning meetings with QA and utilization of SROs in the work planning process.

(3) The staff noted that the permanent corporate organization is beginning to take shape with the hiring of several very capable managers. The staff feels that the corporate organizations can have a significant impact on the establishment of an effective program, but believe that the stabilization of the corporate staff is essential to making this a positive impact and not a negative impact.

Therefore, the NRC concludes that TVA's Maintenance Program is acceptable.

4.9 Restart Test Program

4.9.1 Introduction

In response to employee concerns, TVA conducted a reassessment of its plants' operational safety. A major re-review of the Sequoyah Nuclear Plant, Unit 2, initial design, construction and operating practices has been conducted and a Restart Test Program (RTP) was also instituted to ascertain the functional integrity of the accident mitigation and safe shutdown systems. The program is described in TVA letters of May 26 and July 6, 1987.

The NRC has conducted several inspections of the restart test program as documented in Inspection Reports 50-327/328 87-30, 87-43, 87-54.

The principal objective of the RTP is to instill confidence that certain preoperational tests conducted during initial plant licensing and surveillance inspections routinely conducted following plant licensing and during the long plant shutdown are valid tests that can ensure the current functional integrity of safety systems and components. This assurance is required because the functional integrity might have been jeopardized by plant modifications, maintenance practices, or the like.

This assurance is obtained by reviewing post-modification and maintenance tests and any other tests, or programs that might have a potential impact on the validity of the subject tests.

The scope of the RTP includes testing of integrated safety system functions, beyond periodic surveillance requirements at the component or subsystem level. Such testing is being considered for systems where major modifications could have potentially altered system performance. TVA is presently reviewing all major plant reassessment programs (e.g., Design Baseline and Verification Program, Calculations, and Post-Modification Tests) and has determined that a form of integrated testing is required for (1) portions of the onsite power supply system (diesel generators), (2) the auxiliary feedwater system, and (3) the heating ventilation and air conditioning (HVAC) system.

The main systems identified by the RTP that will require testing to ensure their functional integrity are those systems reviewed by the Design Baseline and Verification Program (DBVP). The DBVP was instituted to assess the adequacy of the plant design and the as-built plant configuration and reconcile potential differences between the design basis and plant modifications. The systems reviewed by the DBVP are the accident mitigation systems that are included in Chapter 15 of the plant FSAR, and the safe shutdown systems. The RTP included verification of the normal functions of these systems.

The accident mitigation and safe shutdown systems that were identified by the RTP for testing, were further subdivided into component or subsystem level functions for which individual functions tests are being conducted. In this program, the integrated performance of the main system function is largely ascertained from valid individual component or subsystem level tests.

The restart test organization was established to implement the RTP and consists of the restart test group (RTG) and the joint test group (JTG).

The RTG consists of test personnel who report to the Restart Test Manager. This group is responsible for developing the function review matrix, function analysis reports, test outlines (all contained in a function analysis package), detailed test instructions, as well as detailed test plans and schedules. The group also is responsible for performing special testing and preparing test analysis packages, test analysis reports, special test instructions, functional test matrices, and reviewing completed test results.

The JTG is responsible for review and approval of various aspects of the RTP.

The function review matrix (FRM) is developed by the RTG to list the identified functions, the tests that acceptably prove these functions, the programs that were reviewed for potential impact on these functions, the results of this review, and any applicable remarks. This matrix is primarily used for internal control and tracking by the RTG. This matrix is presented to the JTG as part of a function analysis package developed by the Restart Test Engineer.

A function test matrix is developed by the RTG to list by system the identified functions, the results of the function test reviews (which include test results), and any remarks. This matrix is completed after the final JTG review of the test analysis package. The JTG reviews and approves the test matrix before it is transmitted to the Site Director.

A restart test program punch list is generated by the RTG to provide an internal method for identifying and tracking open items generated during a review. Open items on this list have unique identifiers to facilitate tracking.

Design functions of systems covered by the RTP are developed by the Division of Nuclear Engineering (DNE) and additional functions may be identified by the RTG as a result of the function review process. As identified previously, these functions include systems required to mitigate FSAR Chapter 15 events as well as systems required for safe shutdown of the plant. Normal functions of these systems also are included. A function under review that affects one or more additional system (interface function) is tracked on the FRM to ensure adequate review in the function review process. These functions are cross-referenced to a previously completed test or a test planned to be implemented during restart, such as a surveillance test or post-modification test. The identified test documents are reviewed to verify that they contain test results that prove the adequacy of the function in the as-constructed condition. This process is documented in the functional analysis report (FAR). If it is determined, however, that a particular function is not adequately tested, new test instructions are generated (special test instructions) and scheduled for implementation at the appropriate time during restart to demonstrate acceptable operation of the identified function.

The above decisions, as well as any applicable test results documented in a test analysis package, that are required to prove the functions, are reviewed by the JTG which, in turn, presents its recommendations to the plant operations review committee and the Plant Manager for review and approval.

Several procedures were written to address the various aspects of the RTP, including the restart test organization, qualification of restart test directors, and the RTP methodology.

4.9.2 Evaluation

Although the RTP did not repeat the pre-operational tests, it did take the as-built plant configuration and assess the effects of subsequent modifications on these test results. Credit was taken for any testing performed as a result of these modifications, for regularly performed surveillance instructions, and for other program outputs.

The staff determined that individual component or subsystem level testing, though not completely equivalent to a fully integrated system test, is equivalent to testing required at other licensed facilities, following initial pre-operational testing, where major modifications have not altered plant configuration and system response requirements. Moreover, the performance of larger tests for systems where major modifications could have potentially altered system performance provides assurance that some tests equivalent to pre-operational tests have been or are scheduled to be conducted. Therefore, the staff has determined this approach to be acceptable.

The staff identified major functions that are omitted from the program, including plant natural circulation and core performance tests. TVA's justification for omitting these functions from the RTP is based on the following:

(1) Natural circulation tests conducted for Unit 1 at Sequoyah continue to be applicable to Unit 2.

Plant configuration has not been altered to affect the heat sink relationship to the heat source and core geometry has not been changed.

Tube plugging for the steam generators has been maintained within allowable margins and no modifications have been made to the reactor coolant flow path since the issuance of the operating license.

(2) Core performance analyses for each reload have been reviewed and approved by the staff, and no modifications have been made to the core geometry since the operating license was issued.

Core physics tests also are performed following each refueling outage to verify that core performance parameters are within the reload analysis envelope. Other tests required by the Technical Specifications will be performed during power ascension to verify present core performance characteristics.

TVA's line slope program (see Section 3.4.1) resulted in some hardware modifications. The RTP has verified that, for all affected cases, instrument functionality and test integrity were preserved.

The staff reviewed TVA's bases for use of the DBVP for identifying systems whose functional integrity must be ascertained before restart of Sequoyah Unit 2. The staff has determined that the DBVP has provided a comprehensive evaluation of the accident mitigation of safe shutdown systems and that the modifications proposed from this evaluation have served to re-establish system functional integrity for the affected systems.

The staff review of the RTP systems resulted in the inclusion of the flood mode boration makeup system and the control rod drive system. The inclusion of the normal functions of these systems, in addition to functions required for accident mitigation and safe shutdown, enhances the completeness of the TVA review.

The staff reviewed the RTP organization and determined that it contains the essential elements required for the proper execution of the program objectives.

Staff audits and field inspections have determined that

o The input provide to the RTP by the DNE is comprehensive.

o RTG's review of this input is thorough and has, in some instances, resulted in additional functions not previously identified by the DNE.

o The function review process is thorough, taking into consideration the results of some 18 programs, processes, and related material including post-modification tests, as-constructed drawings, post-maintenance test surveys, surveillance instructions, design criteria, technical specifications.

o The generated documentation that includes the function analysis reports and test analysis reports is thorough.

The staff's audit reviews and inspections of the implementation of various aspects of the program have provided assurance that the administrative controls and implementing procedures applied in the development of function and test review documentation and test results reviews are properly executed.

The staff performed the safety injection audit during plant recirculation to ensure that the programmatic aspects of the RTP, which include the RTP methodology, have been properly implemented and demonstrate that the chosen mode of operation has been adequately tested. This particular mode of safety injection was chosen for review because Sequoyah probabilistic risk assessment studies have determined that a small-break LOCA event with loss of plant recirculation results in the highest probability for core meltdown. The staff has determined that

- The FARs are thorough in scope and contain adequate documentation for addressing component or subsystem level functional testing. They include related tests performed on a component or subsystem level during pre-operational tests, surveillance instructions, etc. and include the effects of other program outputs on system functions.

- Test Analysis Report (TARs) were assembled for tests completed after the inception of the RTP, including regularly performed surveillance inspection.

- Punch list items were closed, in most instances, soon after the TARs were approved and remaining punch list items will be closed before restart.

- The RTP relies principally on pre-operational tests conducted during initial plant licensing, and surveillance tests, for ensuring functional integrity.

4.9.3 Conclusion

On the basis of its review of the RTP, the staff concludes that continued implementation of the program, as presently constructed, will ensure the functional integrity of safety systems at Sequoyah Unit 2.

4.10 Training

4.10.1 Introduction

Because of the programmatic concerns arising from licensed operator requalification deficiencies identified at Browns Ferry and deficiencies identified in operator and shift technical advisor (STA) knowledge of the safety parameter display system (SPDS), the staff determined that the Sequoyah training program would have to be reviewed for adequacy prior to startup.

Section II.2.3 of the SNPP documents TVA's review and evaluation of training and staffing. In addition to review of this information, the staff conducted an inspection at the Sequoyah site and at the TVA Power Operations Training Center (POTC) the week of February 17, 1986. The results of this inspection are documented in NRC Inspection Report 50-327/328 86-17. The areas inspected are all INPO accredited and included licensed operator and non-licensed operator training and licensed operator requalification training.

Operator requalification examinations were administered by the NRC to licensed holders at TVA December 15 through 18, 1986. Additional inspections of the requalification program were conducted December 14 through 18, 1987 (Inspection Report 50-327/328 87-75).

4.10.2 Evaluation

The overall pass rate of 74 percent for the past 3 years at Sequoyah was cause for staff concern. Contributing causes appeared to be the short length (12 weeks) of the licensee's training course and a shortage of instructors to support the training.

In the SNPP, the licensee committed to increase the reactor operator certification program to 16 weeks. In addition, the licensee has developed observation training qualification cards for reactor operator (RO) and senior reactor operator (SRO) candidates to establish specific study and job assignments during their 13-week observation training phase, to help accomplish the goals of this phase of training. The staff concludes that these measures will enhance the training program and address the concerns previously raised.

The requalification period for licensed operators was 4 weeks, and the staff considered this period brief considering the amount of material to be covered. This conclusion was supported by discussions with the operations and training staff.

The requalification examination administered in December 1986 found the Sequoyah program to be marginal. Three of four reactor operators and one of eight senior reactor operators failed the written examinations, all passed the simulator examination. The reactor operators who failed have received additional training, were re-examined (successfully) and have returned to licensed duty. The weaknesses identified during the NRC requalification examinations were addressed in the requalification training program.

In the SNPP, TVA committed to increase the requalification period to 6 weeks. In 1987, the licensee implemented a six-shift rotation to provide one week in 6 for training, as discussed in Inspection Report 50-327/328 87-37.

Concerns also were raised concerning the amount of requalification training for non-licensed operators. In the SNPP, TVA noted that training for assistant unit operators was increased from 1 week to 2 weeks in 1986 and will be 6 weeks in 1987 and thereafter. The staff finds this commitment acceptable.

In Section II.2.3.6 of the SNPP, TVA describes the training that will be given to project managers. The duties of the project managers involve ensuring that proper planning and controls are in place for projects requiring the approval of the Manager of Nuclear Power. Training of the project managers is intended to provide them with the understanding needed to function quickly and effectively. Also, the program will help to develop the skills necessary to achieve proper planning and control over the projects.

The staff has reviewed the information provided in the SNPP and has determined that the training program for project managers is acceptable to permit restart of the Sequoyah facility. However, the staff will continue to monitor this program to ensure proper implementation.

As described in the SNPP, a training program for new technical staff has been developed. The training consist of 4 weeks that are devoted to plant reference material and procedures along with the appropriate codes and regulations. This training is in addition to the INPO-accredited Engineers and Managers Certification Training Program.

The staff concludes that the training program for nuclear site personnel is acceptable for restart.

TVA has attained INPO accreditation for non-licensed operator training, licensed RO training, licensed SRO training, STA training, technical staff and managers training, instrument and control technician training, chemistry technician training, radiation protection technician training, electrical maintenance and mechanical maintenance training. Thus, their program is accredited in all areas.

The SPDS was installed and implemented on Sequoyah Units 1 and 2 in September and October 1985, respectively. Inspections in November 1985 determined that adequate training had not been conducted for operators and STA on the SPDS. As a result of this finding, TVA conducted retraining, which included a comprehensive operational performance test. TVA also developed an SPDS user's manual that will be a controlled plant document available in the control room. These corrective actions were inspected as documented in Inspection Report 50-327/328 86-28.

Technical support managers have completed either STA training or the engineers and managers certification training. This exceeds industry norms and the staff finds this level of training acceptable.

Maintenance training is discussed in Section 4.8 of this safety evaluation.

4.10.3 Conclusion

The staff concludes that the training plans set forth by TVA are acceptable.

4.11 Security

In the 10 CFR 50.54(f) letter (September 17, 1985), the staff noted that there were several areas in which TVA had not been performing adequately. These areas were identified from their low ratings within their respective SALP categories. As a result of these concerns, TVA has initiated several actions intended to upgrade performance. In the most recent SALP, the staff found an improving trend in the area of security, compared to the degradations previously noted. However, to ensure that this improvement would continue, TVA undertook several actions. These actions, which are discussed in Item 4 of Appendix 2 to the SNPP, are evaluated below.

TVA identified in the SNPP those measures it will take to enhance the knowledge of supervisors and employees in their responsibilities for complying with security requirements. Public Safety Service, a division of the Office of Corporate Services, will trend all security degradations to identify areas for improvement and revise the training program for public safety to include experience from prior security incidents. To ensure the planned improvements were being properly implemented, the staff conducted physical security inspections at the Sequoyah plant as documented in Inspection Report Nos. 50-327/328 86-30, and 50-327/328 86-47.

The staff has reviewed the information provided in the SNPP and has performed several physical security inspections as part of its evaluation of the improvements to the Sequoyah plant security. Based on the results of its evaluation, the staff concludes that the action taken by TVA to improve security addresses the staff's concerns. In addition, the staff finds that with the implementation of these actions, TVA will have an acceptable security program for restart of either Sequoyah unit.

4.12 Emergency Preparedness

4.12.1 Introduction

SNPP Appendix 2, Section 6, Revision 1, documents TVA's actions taken in the Sequoyah emergency preparedness (EP) program to resolve problems identified in NRC SALP evaluations. The corporate Emergency Preparedness Branch has been reorganized and additional staff identified to provide additional resources in the areas of emergency planning and procedures, state and local government interfaces, development and conduct of exercises and drills, and onsite and offsite facilities. Additional staff has been identified at the sites for program implementation.

Problem areas which have been addressed by TVA include (1) inadequate coordination between the Central Emergency Control Center (CECC) and the Radiological Dose Assessment (RDA) staff, (2) inaudible inplant alarms, and (3) vaguely written implementing instructions for protective action recommendations. Improvements have been made in emergency organization, emergency facilities and equipment, emergency classification system, accident assessment, training and drills, and procedures to enhance the licensee's emergency capabilities.

4.12.2 Evaluation

Improvements to TVA's Radiological Emergency Plan (REP) have been made in the areas addressed below.

TVA has changed the emergency organization so that the RDA staff operates as an integral function of the CECC. This change involved the consolidation of the RDA staff from Muscle Shoals, Alabama, to the CECC offices in Chattanooga, Tennessee. The effectiveness of this change was demonstrated by the successful performance of the CECC staff during the Sequoyah emergency preparedness exercise November 19, 1986.

Another organizational change included providing engineering support from the onsite Division of Nuclear Engineering (DNE) as well as DNE engineers located in Knoxville, Tennessee, to the onsite Technical Support Center (TSC) by onsite DNE staff. This support was previously provided indirectly to the site through the CECC or by DNE staff in Knoxville.

TVA has completed installation of sirens and strobe lights in accordance with approved engineering change notices issued to meet the requirements of IE Bulletin 79-18, Audibility of Alarms in High-noise Areas. Tests to verify the system's effectiveness with the added sirens and strobe lights will be completed after restart of both units, when the equipment operating noise levels are normal.

The SPDS has been installed at Sequoyah to meet the requirements of NUREG-0737, Supplement 1, Item I.D.2. The SPDS and the onsite TSC functions of the TSC computer are functional for both units and are accessible in the CECC. The installation and validation program for the SPDS is considered adequate and the systems were declared operable by TVA within commitment dates.

TVA has evaluated and revised the emergency classification system criteria, which was identified as being vague in the 1985 emergency preparedness exercise. Additionally, TVA is continuing evaluation of the criteria for possible further enhancement. As revisions are made, TVA will enhance operator training on emergency action levels and emergency classification.

TVA also has revised the protective action recommendation (PAR) chart used by the Site Emergency Director/Shift Engineer for a licensee declaration of a General Emergency to enable them to make consistent offsite protective action recommendations, including utilization of specific plant status indicators. The use of the revised PAR was satisfactorily demonstrated during the Sequoyah emergency preparedness exercise November 1986.

Previous problems with coordination of offsite monitoring teams has been addressed by TVA by assigning the CECC RDA staff the responsibility for directing offsite TVA radiological environmental monitoring efforts in support of site government operations in an emergency once the CECC is staffed. Emergency preparedness procedures have been revised to reflect this change in responsibility and the 1986 SQN exercise demonstrated satisfactory coordination of environmental monitoring efforts.

TVA has included a training module on offsite PARs in the licensed operator requalification training program. Simulator and classroom training on the use of the SPDS as well as training on the onsite TSC functions of the TSC computer have been included in requalification training.

TVA has designated a full-time staff position at Sequoyah; the site EP Program Manager is responsible for implementation of the EP program on site. To assist the Manager, a full-time technical position also has been identified. The Manager's duties include coordinating the development of the site-specific portions of the emergency plan and the site-specific implementing procedures; implementation of onsite drills; onsite EP training program; providing support to the annual exercises scenario development efforts; maintaining site emergency facilities, equipment, and supplies; and providing timely resolution of internally and NRC identified weaknesses for Sequoyah.

TVA has established the EP Exercises and Facilities Section within EPB, with EP exercise scenario development and implementation being one of its major functions. The Site EP Program Manager provides input to EPB on development of the annual exercise. The site manager assists, as necessary, in the exercise scenario implementation including training, supervision of exercise controllers and designated observers, and the critique of the onsite exercise performance.

Over the past 2 years, TVA has put considerable effort into revising and enhancing onsite and corporate EP procedures. The REP has been revised to reflect organizational changes that have taken place and redefined responsibilities and interfaces needed because of the changes. Additionally, a proposed "generic" REP for the Office of Nuclear Power has been developed and is currently under internal TVA review. This "generic" REP would consolidate the individual site REPs into a single ONP Emergency Plan with site-specific appendices.

The NRC inspection of the exercise conducted on August 6, 1987 (Inspection Report 50-327/328 87-49), identified no violations or deviations. An additional inspection of the REP was conducted in September 1987 (Inspection Report 50-327/328 87-58).

4.12.3 Conclusion

On the basis of its review, the staff finds that, with proper implementation, past EP problem areas should be satisfactorily resolved.

4.13 Radiological Controls

In Section II.1.2.3 of the SNPP, TVA discusses its improvements to the radiological controls (RC) organization. These include the following:

° A site Radiological Assessor position has been established on the Site Director's staff to provide programmatic overview of the Sequoyah RC program.

° The Superintendent of Site RC now reports directly to the Plant Manager.

° The contamination area control program has been implemented.

° A new decontamination facility has been placed in operation.

° An inventory and centralized storage area has been designated for radiation shielding materials.

° The Health Physics Shift Supervisor participates in maintenance planning.

° A training position has been established in support of RC.

° Additional staff positions on site have been established for professional health physicists.

The staff concludes that these measures will strengthen the RC program at Sequoyah. Several inspections have been conducted of the Sequoyah radiation protection program, as discussed in Inspection Reports 50-327/328 86-54, 87-03, and 87-56. The staff concludes that the actions taken by the licensee, including correction of previous weaknesses in its program for maintaining exposures as-low-as-reasonably-achievable (ALARA), are sufficient to support plant restart.

5 EMPLOYEE CONCERNS

During the spring of 1985, a number of TVA employees informed the NRC and selected members of Congress of safety concerns, primarily related to the Watts Bar Nuclear Plant. In addition, TVA learned of many employee concerns through its own organization. The concerns expressed indicated that many TVA employees had lost confidence in TVA's nuclear management and its ability to properly conducted nuclear activities. In addition, some of these employees expressed fear of reprisal from TVA management if they raised their concerns directly. Two programs relating to employee concerns have resulted; they are referred to as the new program and the special program. These two programs are discussed in detail in the staff's Safety Evaluation Report on the Tennessee Valley Authority Revised Corporate Nuclear Performance Plan, NUREG-1232, Volume 1, dated July 1987.

The new employee concern program (ECP) was implemented at Sequoyah on February 1, 1986 as described in a TVA submittal of February 3, 1986. The key element of the program is the ECP Site Representative at Sequoyah. The ECP staff receive and investigate concerns from employees who feel that normal channels of resolution have failed. The program is further described in other TVA submittals including the SNPP. The staff issued its safety evaluation accepting the TVA new ECP on September 30, 1987.

In May 1985, TVA awarded the Quality Technology Company (QTC) a contract to develop and implement a program for conducting confidential interviews with TVA employees performing assignments for the Watts Bar Nuclear Plant. Concerns also were collected from TVA employees at the Sequoyah and Browns Ferry plants. This program, which emphasized the identification of employee concerns dealing with nuclear safety at all TVA facilities, identified more than 5,000 employee concerns. In February 1986, TVA initiated a program to evaluate and resolve these employee concerns. The employee concern special program (ECSP) was developed to review the concerns received through the QTC or from the Nuclear Safety Review Staff (NSRS) for applicability to Sequoyah. This work was performed by the Watts Bar employee concern task group (ECTG). The staff evaluation of the ECSP was issued to TVA by letter dated October 6, 1987.

The employee concerns were grouped into nine categories for evaluation and resolution. The categories are construction; engineering; industrial material control; operations; quality assurance/quality control; welding; management and personnel; industrial safety; and intimidation, harassment, wrongdoing, or misconduct.

Because the Sequoyah plant is presently scheduled to be the first TVA plant restarted, the concerns applicable to Sequoyah only, within each employee concern subcategory, were divided into individual element reports that addressed related concerns. For Sequoyah, over 300 element reports were prepared covering six of the categories. TVA has submitted element reports to address the resolution of employee concerns for Sequoyah. Safety evaluations on the individual element reports will be provided in Part 2 of this safety evaluation.

Subcategory and category reports will address the resolution of employee concerns for the other TVA nuclear plants. TVA will not submit any element report for the management and personnel and industrial safety categories because TVA has concluded these do not contain safety-related concerns. The staff has concluded that employee concerns in these two categories have been adequately addressed as discussed in letters to TVA (December 14, 1987(c), and August 24, 1987, respectively). Concerns in the ninth category, relating to intimidation, harassment, wrongdoing, or misconduct, will be investigated and the results reported separately by the TVA Office of General Counsel or the TVA Inspector General. The staff's review of TVA's handling of these concerns is discussed in an October 8, 1987 letter to TVA.

On the basis of its review of the TVA employee concerns program, the NRC staff concluded in Volume 1 of NUREG-1232 that TVA now has a policy that promotes quality and safety and TVA has taken steps to ensure that this policy is understood by TVA employees and that the policy is strictly enforced. The actions taken by TVA to improve employee confidence define an acceptable program for dealing with employee concerns. In combination with the other improvements in the nuclear program that TVA is implementing, these steps should improve the confidence of employees in TVA's management. The staff considers effective implementation of the new employee concerns program necessary if TVA is to significantly change its prior performance record.

The staff will continue to monitor program implementation and the effectiveness of actions taken to deter intimidation and harassment. The NRC staff has issued, by letter dated March 11, 1988(b), its "Preliminary Safety Evalautions on the Tennessee Valley Authority Employee Concern Element Reports."

6 ALLEGATIONS

A number of allegations of safety problems at TVA have been made directly to the NRC staff in lieu of being provided to TVA under the employees concerns program. In a number of instances, the technical content of the allegation has been provided to TVA for its review and response to the NRC. For these cases, TVA has entered the allegation into its employee concerns program and the technical resolution of the issue is discussed in the safety evaluation for the specific element report. The remaining allegations will be handled by the staff in accordance with established NRC policies for allegations. The NRC has reviewed all allegations to identify potentially safety significant Sequoyah-related allegations, which will be resolved before restart of Sequoyah.

APPENDIX A

LIST OF CONTRIBUTORS

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APPENDIX B

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---, February 3, 1987, letter from J. A. Domer to Document Control Desk (NRC), Subject: Exemption from 10 CFR 50, Appendix A, General Design Criteria 55 and 56."

---, February 4, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Supplemental Items Interim Acceptance Criteria for Cable Tray Supports."

---, February 6, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Concrete Evaluation Report."

---, February 17, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Superheat in Main Steam Valve Vaults."

---, February 27, 1987(a), letter from R. Gridley to Document Control Desk (NRC), forwarding comments on draft SER on DBVP.

---, February 27, 1987(b), letter from R. Gridley to Document Control Desk (NRC), Subject: "Electrical Design Calculations."

---, February 27, 1987(c), letter from R. Gridley to Document Control Desk (NRC), Subject: "Summary of Cable Ampacity Sampling Program."

---, March 12, 1987, letter from R. Gridley to Document Control Desk (NRC), transmitting list of commitments on electrical calculations.

---, March 23, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Status of Post-Modification Test Review."

---, March 24, 1987, letter from R. Gridley to S. Ebneter (NRC), Subject: "Surveillance Instruction Review Program."

---, March 26, 1987, letter from S. White to S. Ebneter (NRC), transmitting Revision 4 to Revised Corporate Nuclear Performance Plan.

---, April 1, 1987, letter from S. White to Document Control Desk (NRC), transmitting Revision 1 to the Sequoyah Nuclear Performance Plan.

---, April 1, 1987(a), letter from R. Gridley to Document Control Desk (NRC), Subject: "Sequoyah Drawings to be Maintained as Configured."

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---, April 2, 1987, letter from R. Gridley, Document Control Desk (NRC), Subject: "Instrument Sensing Line Slope questions."

---, April 8, 1987(a), letter from R. Gridley to Document Control Desk (NRC), Subject: "Sequoyah - Alternate Analysis Program Phase II."

---, April 8, 1987(b), letter from R. Gridley to Document Control Desk (NRC), Subject: "Concrete Evaluation Supplemental Information on Concrete Sampling."

---, April 8, 1987(c), letter from R. Gridley to Document Control Desk (NRC), Subject: "Exemption from 10 CFR 50, Appendix A, General Design Criteria 55 and 56."

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---, May 13, 1987, letter from M. Martin to Document Control Desk (NRC), transmitting element reports on material control.

---, May 15, 1987, letter from R. Gridley to Document Control Desk (NRC), forwarding "Engineering Assurance Oversight Review Report - DBVP."

---, May 26, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: Restart Test Program."

---, June 12, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Electrical Calculations - Revised Final Status Report.

---, July 2, 1987(a), letter from S. A. White to Document Control Desk (NRC), , transmitting Revision 2 to the Sequoyah Nuclear Nuclear Performance Plan.

---, July 2, 1987(b), letter from R. Gridley to Document Control Desk (NRC), Subject: "Moderate Energy Line Break Evaluation."

---, July 6, 1987, letter from J. A. Domer to Document Control Desk (NRC), Subject: "Restart Test Program."

---, July 8, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Leakage of Spray Behind the Crane Wall Following a Postulated Design Basis Accident."

---, July 20, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Instrument Sensing Line Slope Questions Norris Report No. WR 28-1-85-124.RL."

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---, September 18, 1987, letter from R. Gridley to Document Control Desk (NRC), transmitting response to IE Bulletin 87-01.

---, October 6, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Sequoyah Unit 2 - Pipe Support Modification Restart Criteria Meeting Summary."

---, October 9, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Moderate Energy Line Break Flooding Evaluation."

---, October 23, 1987, letter from R. Gridley to Document Control Desk (NRC), transmitting Supplemental EA Oversight Review Report for DBVP.

---, October 29, 1987, letter from S. White to J. Keppler (NRC), regarding findings from Integrated Design Inspection.

---, November 17, 1987(a), letter from R. Gridley to Documental Control Desk (NRC), Subject: "Unit 2 Support Modification Restart Criteria Supplemental Revision."

---, November 17, 1987(b), letter from R. Gridley to Document Control Desk (NRC), Subject: "Additional Information Requested by NRC on Fuse Replacements."

---, November 20, 1987, letter from R. Gridley to Document Control Desk (NRC), transmitting results of cable testing.

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---, December 4, 1987, letter from R. Gridley to Document Control Desk (NRC), transmitting response to Inspection Report 50-327/328 87-44.

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---, December 8, 1987(b), letter from J. Russell to Document Control Desk (NRC), Subject: "Employee Concerns Special Program - Sequoyah Element Report C017301."

---, December 10, 1987, letter from S. White to Document Control Desk (NRC), transmitting Revision 5 to Revised Nuclear Performance Plan.

---, December 15, 1987, letter from R. Gridley to Document Control Desk (NRC), Subject: "Design Control Program."

---, December 28, 1987, letter from S. White to S. Ebneter (NRC), Subject: "Silcone Rubber Insulated Cable Issue Resolution."

---, December 29, 1987, letter from S. White to S. Ebneter (NRC), Subject: "Integrated Design Inspection (IDI): Response to NRC Inspection Report 50-327/328 87-48."

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---, August 1, 1986, B. Youngblood letter to R. Gridley (TVA), Subject: "Verification Testing for Auxiliary Power System Voltage Study for Sequoyah."

---, August 7, 1986, B. Youngblood letter to R. Gridley (TVA), transmitting questions on verification testing.

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---, WCAP-10988, "COBRA-NC, Analysis for Main Steamline Break in Catawba Unit 1 Ice Condenser Containment," (Proprietary), November 1985.

---, WCAP-8822-P-S1/WCAP-8822-P-S2, "Mass and Energy Release Following a Steam Line Rupture," January 1985 and September 1985, respectively.

APPENDIX C

TECHNICAL EVALUATION REPORT (TER)
RELATED TO ELECTRICAL EQUIPMENT
ENVIRONMENTAL QUALIFICATION

TECHNICAL EVALUATION REPORT

NRC DOCKET NO. 50-327, 50-328

NRC TAC NO. --

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 41

FRC TASK 658

REVIEW OF THERMAL ANALYSIS
OF ELECTRICAL EQUIPMENT FOR MAIN STEAM LINE BREAK
ENVIRONMENTAL QUALIFICATION

SEQUOYAH UNITS 1 AND 2

TER-C5506-658

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

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May 8, 1987

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Department Director
Date: 5-8-87

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CONTENTS

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

1. INTRODUCTION AND SCOPE OF REVIEW

1.1 INTRODUCTION

Equipment that is used to perform a necessary safety function in a nuclear power plant must be shown to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria (GDC) 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10CFR50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability in electrical equipment have been set forth in 10CFR50.49 and Regulatory Guide 1.89, Rev. 1.

During and following postulated accidents in nuclear power plants, safety-related electrical equipment may be subjected to harsh environments. As part of the effort to demonstrate that equipment is capable of maintaining qualified functional operability under all service conditions, the U.S. Nuclear Regulatory Commission (NRC) regulations state that testing with supporting analysis may be used to show that equipment is acceptable.

The Tennessee Valley Authority (TVA) opted to use thermal analysis in conjunction with qualification test results after new temperature profiles for the main steam line break (MSLB) for the Sequoyah Nuclear Power Plant main steam valve vaults (MSVVs) were found not to be bounded by the qualification test temperature profiles. The analyses were submitted to NRC to demonstrate qualification for the MSLB. The NRC staff requested Franklin Research Center (FRC) to review the analyses and to verify the validity, completeness, and acceptability of the heat transfer calculations provided to the NRC staff in the Licensee's submittals. The TVA analyses represent time-dependent thermal responses of safety-related electrical equipment located in the main steam isolation valve vaults of Sequoyah Nuclear Plant Units 1 and 2.

This report provides an evaluation of the TVA submittals for the heat transfer analysis of components in the main steam isolation valve vaults.

1.2 SCOPE OF REVIEW

FRC was contracted by the NRC to provide technical assistance in determining the acceptability of TVA's analyses for fulfilling the requirements of 10CFR50.49. The following tasks were to be performed:

1. Review the list of equipment provided in the submittal to determine if the most critical subassembly was being used to conduct the analysis.

2. Review the failure modes identified in the submittal for completeness.

3. Review the heat transfer calculations for acceptability to determine if:

   a. the methodology was reasonable and was an acceptable means to analyze the conditions of interest

   b. the results were sufficiently accurate to reasonably represent reality.

2. BACKGROUND

After having been informed by the NRC that certain mass and energy releases had not been taken into account in calculating the response of the primary containment atmospheres to an MSLB, TVA became aware that the issue would also affect the MSVVs located outside the primary containment. TVA reevaluated its Sequoyah Nuclear Plant's MSVV temperature profiles considering the additional energy from the MSLB and calculated a peak atmospheric temperature of 535°F using a standard subcompartment code. This temperature was substantially higher than the 325°F design temperature used in the equipment qualification (EQ) program. From a list of options for resolving this problem, TVA chose Westinghouse's suggested approach of reanalyzing an MSLB in the valve vault by taking into account the circulation of the cool outside air in the vaults that would occur after such a break. This effect was not modeled in the subcompartment code used in the previous analysis. The Westinghouse analysis indicated a peak MSLB atmosphere temperature of 435°F, which was still higher than the EQ program peak temperature of 325°F.

To demonstrate that electrical equipment located in the MSVV will be able to operate as necessary during an MSLB, TVA opted to analyze the thermal response of Categories A and B components to the MSLB profile and to compare the results to existing results from qualification tests. (A Category A device is required to operate to mitigate an event; a Category B device is not needed to mitigate a design basis event, but must not fail in a manner detrimental to safety. Category C devices are not needed to mitigate design basis events and have no failure modes that affect safety functions or could mislead the operator.) TVA believed that although high surface temperatures were possible, it was unlikely that the internal components of equipment would rise above the temperatures they experienced during qualification testing. This premise formed the basis for their approach.

TVA calculated the surface and internal temperatures for equipment that would be exposed to the expected MSLB temperature profile and to the EQ test temperature profiles. The TVA proposed that, by comparing the results from these calculations, the qualification of the equipment for MSLB service can be determined from existing qualification test documentation if the surface or internal temperatures during the MSLB event are bounded by the surface or internal temperatures from the existing qualification test.

3. DISCUSSION OF TVA SUBMITTALS

The Licensee submitted three documents relating to MSLB equipment qualification. The first document, which was submitted to the NRC on August 13, 1986, contained extensive information on the east and west valve vaults, and identified thermal response of a limited number of pieces of electrical equipment located in the vaults to an MSLB. The information included a physical description of the valve vaults with the size of vent areas. The results of the Westinghouse COMPACT analyses of the vaults' atmospheric temperature profile were also presented as were assumptions and detailed descriptions of the vault models supported by data on the compartments and heat sinks.

The COMPACT results were presented for three postulated MSLB events, namely, a double-ended rupture of the steam line (1.4 ft^2), a 0.9 ft^2 break assumed to occur upstream of the main steam line, and a 0.9 ft^2 break downstream of a main steam isolation valve (MSIV). The thermal response of a solenoid for the MSIV, an ASCO solenoid valve, and a cable in a conduit were presented for all of the break cases analyzed. These components were analyzed by use of the COMPACT code. The justification for choosing these components for analysis was that if it could be shown that the components' response to the MSLB temperature profile did not exceed the chamber temperature profile from EQ testing, no component in the valve vaults would exceed its EQ temperature during a MSLB. This methodology relied on the assumption that these components had the least thermal mass of all of the components and would respond most rapidly to the MSLB temperature profile. The devices were modeled as one-dimensional slab heat sinks except for the cable in the conduit which was modeled in two dimensions.

Analyses of the components done independently by the Licensee using the HEATING5 heat conduction code were included in the August 13, 1986 submittal. The purpose of these analyses was to evaluate the effect of different modeling. This submittal also contained the results of RELAP5 modeling performed for verification of the COMPACT results for the atmospheric profiles.

In response to a request for additional information (RAI) dated November 14, 1986, TVA forwarded a second submittal dated December 23, 1986. In response to the request to provide a logical basis for the selection of the

ASCO solenoid coil and cable as the critical devices for evaluation. TVA replied that the selection was based on the concept that the components with the least thermal inertia would heat up most during the MSLB. The valve vault equipment lists had been reviewed, and components with low mass and thin housings were selected for analysis. TVA believed that the thermal response of these components would bound the response of all other larger and heavier components. However, upon receipt of the RAI, TVA performed thermal analyses for all of the equipment in the vaults that required qualification. These analyses were submitted to support the basis of the original selection of equipment for evaluation.

TVA also reported that no equipment was removed from consideration in the MSLB EQ study on any basis other than function. This response was prompted by the request to demonstrate that failure of a device that was removed from the list on a basis other than function will not degrade any safety system or provide misleading information to the operators. Revised lists of the Categories A and B equipment expected to be in the valve vaults at the time of restart of Sequoyah Unit 2 and the cable types located in the vaults were provided in the December 23, 1986 submittal. The information in the tables included device number, manufacturer, model number, and a description of the function of the component as requested. Table 1 identifies the equipment requiring qualification. The critical internal components of Categories A and B electrical equipment were identified by TVA to be the cable insulation, other elastomers, and solenoid coils. A discussion to support TVA's conclusion that these components would not fail was presented. TVA's type PJJ* cable was chosen for evaluation because it is a small multiconductor cable and thus would have a relatively rapid heatup. In addition, its thermoplastic jacket and insulation materials are more heat sensitive than the thermoset materials in other cables in the valves.

Analyses were provided in the December 23, 1986 submittal for the Limitorque valve operators, junction boxes, terminal connectors, and Namco limit switches in addition to the MSIV solenoid valves and ASCO solenoid

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*PJJ is a TVA code referring to multiconductor cable with polyethylene insulation and polyvinyl chloride jacket.

Table 1. List of Equipment Requiring Qualification for Main Steam Line Break Temperature Profile in the East and West Main Steam Valve Vaults

Equipment	ID No.	Manufacturer
MSIV Solenoid	FSV-001-004A Through FSV-001-004J FSV-001-029A Through FSV-001-029J FSV-001-011A Through FSV-001-011J FSV-001-022A Through FSV 001-022J	Gould Allied
Valve Operator-Auxiliary Feedwater Pump - Turbine Steam Supply	FCV-001-015 FCV-001-016 FCV-001-017 FCV-001-018	Limitorque
Valve Operator - Main Feedwater Isolation	FCV-003-033 FCV-003-100 FCV-003-047 FCV-003-087	Limitorque
Main Steam Line Warming Solenoid Valve	FSV-001-147 FSV-001-148 FSV-001-149 FSV-001-150	Asco
Steam Generator PORV Limit Switches	PVC-001-012 (LS) PVC-001-023 (LS) PVC-001-005 (LS) PVC-001-030 (LS)	NAMCO
Steam Generator Blowdown Isolation Solenoid Valves	FSV-001-007 FSV-001-014 FSV-001-025 FSV-001-032	Asco
Steam Generator Blowdown Isolation Limit Switches	FSV-001-007 (LS) FSV-001-014 (LS) FSV-001-025 (LS) FSV-001-032 (LS)	NAMCO
Level Control Solenoid Valves	LSV-003-174 LSV-003-175	Asco
Junction Boxes	1-JBox-991-1987-B 1-JBox-991-1988-A 2-JBox-991-1998-A	N/A

Table 1. List of Equipment Requiring Qualification for Main Steam Line Break Temperature Profile in the East and West Main Steam Valve Vaults (Cont.)

Equipment	ID No.	Manufacturer
Junction Boxes (Cont.)	1-JBox-991-1985-A	N/A
	1-JBox-991-3067-B	
	1-JBox-991-3114-A	
	1-JBox-991-3116-B	
	2-JBox-991-1986-A	
	2-JBox-991-3070-B	
	2-JBox-991-3115-A	
	2-JBox-991-3117-B	
	1-JBox-991-2041-B	
	1-JBox-991-2042-A	
	1-JBox-991-2857-B	
	1-JBox-991-2858-A	
	2-JBox-991-2890-B	
	2-JBox-991-2891-A	
	2-JBox-991-2892-B	
	2-JBox-991-2893-A	
	1-JBox-991-3041-A	
	1-JBox-991-3042-A	
	1-JBox-991-3061-A	
	1-JBox-991-3065-B	
	1-JBox-991-3066-B	
	2-JBox-991-3062-A	
	2-JBox-991-3063-A	
	2-JBox-991-3064-A	
	2-JBox-991-3068-B	
	2-JBox-991-3069-B	
	2-JBox-991-1997-B	
Terminal Connector	SQN-XXX-TB-991	GE
Raychem Splices	WCSF-N Series	Raychem Corp.

valves which had been submitted in the August 13, 1986 document as one-dimensional analyses. All of the new analyses were two-dimensional. An explanation of the heat transfer methodology was also included. Assumptions and derivations of heat transfer coefficients were discussed. The December 23 submittal also documented the assumptions made in each individual analysis.

A justification for modeling equipment as multilayer slab-type heat sinks was provided in a December 23 submittal. TVA stated that a one-dimensional model, in conjunction with proper selection of heat transfer paths, can be used to conservatively maximize the equipment's external and internal temperatures. The results of the analyses were presented in the form of temperature-time profiles of the responses.

A third submittal, dated February 17, 1987, was forwarded to the NRC in response to an RAI dated January 20, 1987. This document addressed specific questions in the RAI. Responding to a request to identify the pieces of equipment which have been tested to determine internal temperatures, it was stated that qualification of all MSVV equipment types to the superheat profile is based on thermal analysis. Telephone discussions with TVA had indicated that some testing was done that would supersede the analysis. A list of equipment to be relocated prior to restart was also provided.

TVA's response to a request for information concerning acceptability of the terminal blocks for use in a steam environment referred to testing in which terminal blocks were exposed to the worst-case accident profile postulated for Sequoyah's containment. It was stated that as a result of these tests, terminal blocks had been removed from transmitter circuits that required qualification in accordance with 10CFR50.49, but had been determined to be acceptable for other 10CFR50.49 applications.

In response to a request to provide a description of the justification for the change in reclassification of the Masoneilan valve positioners to Category C from Category A, it was stated that failure of the positioners in conjunction with a single active failure does not place the plant in a configuration that would prevent the availability of one intact steam generator and one motor-driven auxiliary feedwater pump and that the devices no longer required qualification.

TVA also provided details of the guidelines used in constructing the heat transfer models. TVA's assumptions were made to conservatively increase the temperature predicted during a MSLB, while lowering the surface and internal temperatures resulting from analysis of EQ test chamber profiles. In this way, higher than expected surface and internal temperatures resulting from MSLBs are compared to lower than expected surface and internal temperatures derived from the qualification tests, thereby adding conservatism. The values of heat transfer coefficients used for the MSLB analyses were provided along with schematics of the models used. The Licensee concluded that qualification of the MSIV devices was established for each component by comparison of the calculated thermal response during a MSLB to the calculated response during qualification testing. The qualification for steam and moisture exposure is based wholly upon the existing qualification test results. The review of the existing qualification test results was not within the scope of this evaluation. The results of the existing qualification tests have been assumed to be acceptable during this evaluation. A sample of Licensee-supplied temperature profiles is included as Figures 1 through 3. Figure 1 represents the EQ temperature profile of a MSIV solenoid versus the MSLB temperature profile. Figure 2 is a plot of the calculated MSIV solenoid surface temperature during the MSLB. Figure 3 represents the calculated internal coil temperature during a MSLB as compared to the EQ test temperature.

FIGURE SUPPLIED
BY THE LICENSEE

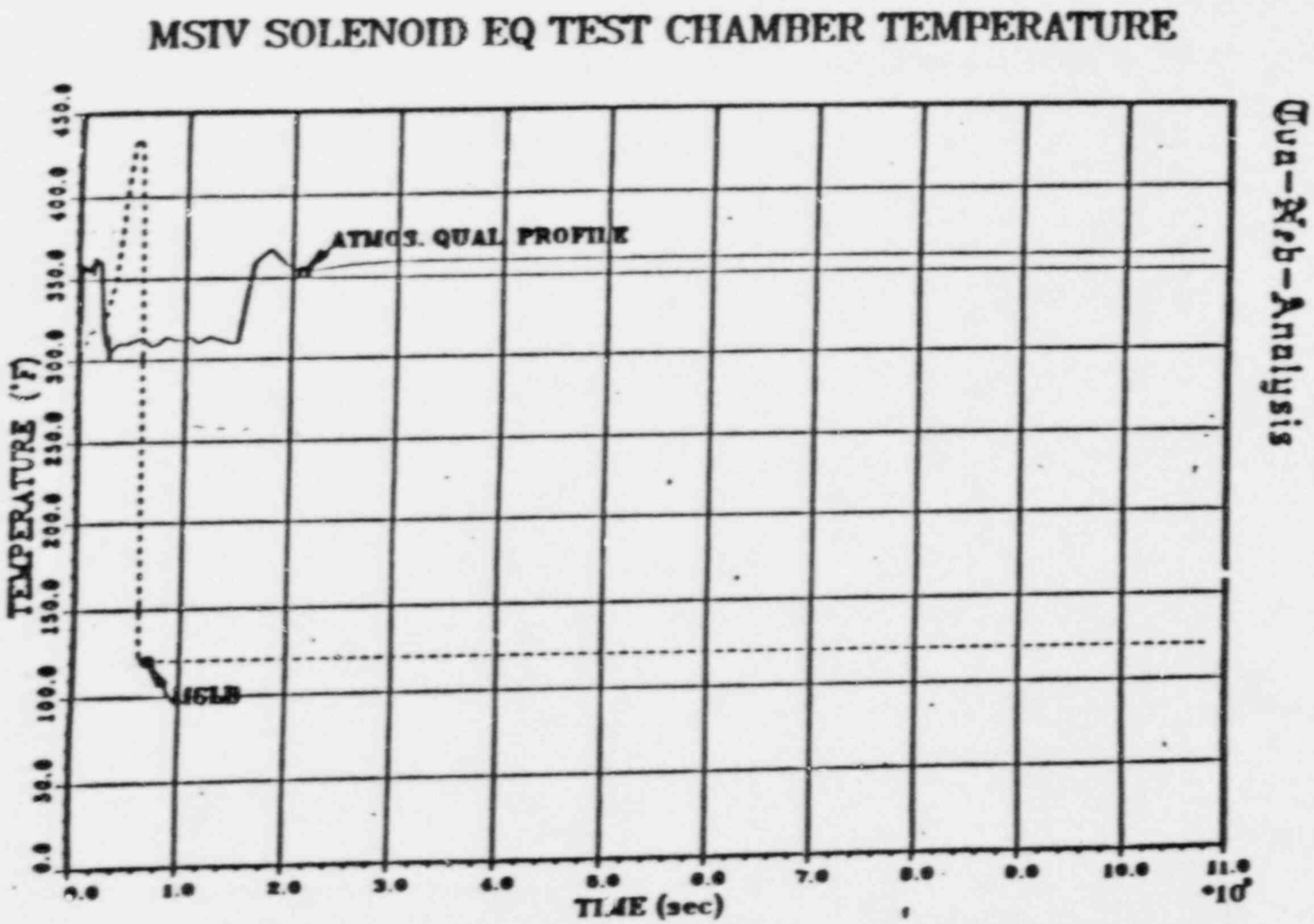

Figure 1. MSIV Solenoid EQ Test Chamber Temperature

TER-C5506-658

FIGURE SUPPLIED
BY THE LICENSEE

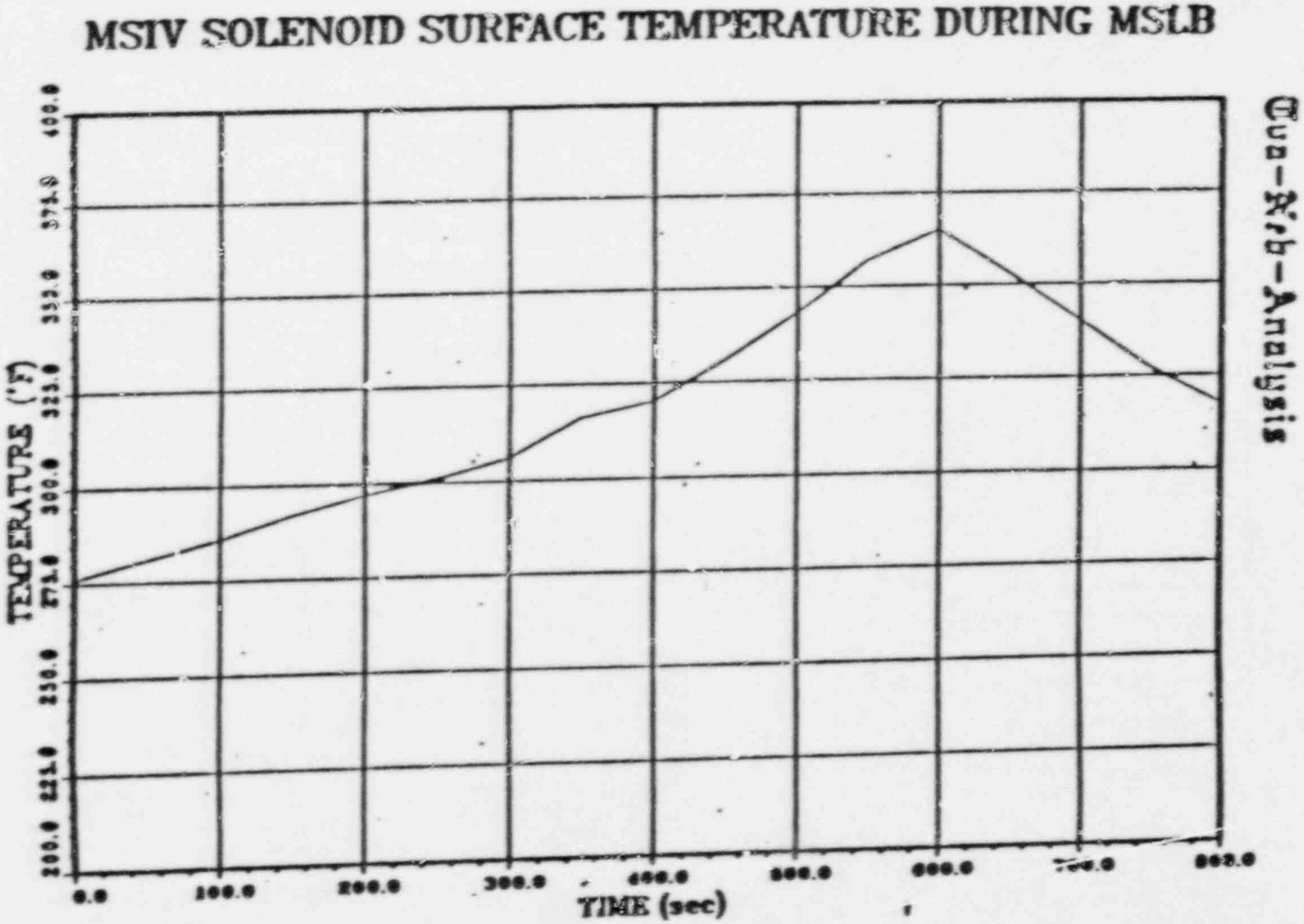

Figure 2. MSIV Solenoid Surface Temperature During MSLB

TER-C5506-658

FIGURE SUPPLIED
BY THE LICENSEE

MSIV SOLENOID COIL TEMPERATURE DURING EQ TEST AND MSLB

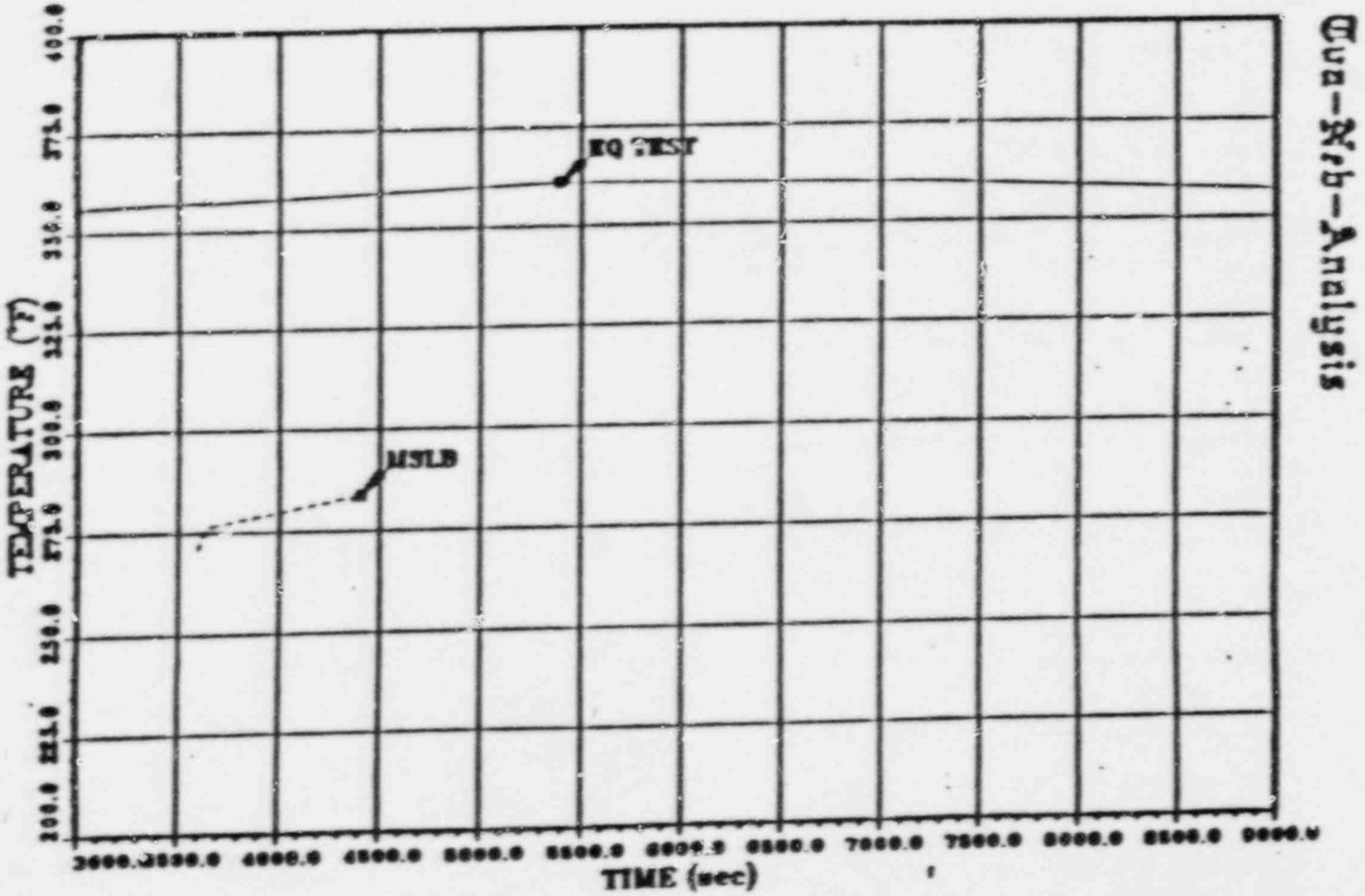

Figure 3. MSIV Solenoid Coil Temperature During EQ Test and MSLB

4. EVALUATION

4.1 TVA METHODOLOGY

TVA used two-dimensional sections in developing the heat transfer models. Although the two-dimensional models may produce temperature profiles that are lower than actual because of the reduction in heat transfer surfaces, the relative position of the MSLB and EQ profiles would not change and, therefore, a qualification determination can still be made. Except for the cable-in-conduit model, it was stated that no gap resistances were modeled so as to maximize the heatup of internal components in response to a MSLB.

TVA methods used to derive the heat transfer coefficients in the analyses were consistent with guidelines in NUREG-0588, Appendix B where applicable and were conservatively obtained in other cases. An important concept in this evaluation is the understanding that qualification testing was performed for several hours under saturated conditions. This would allow for nearly ideal heat transfer conditions. The MSLB event being analyzed would be of much shorter duration and at lower saturation temperatures. While this would allow for high surface temperatures, the internal temperature of vital components should actually be significantly lower than that shown in the EQ profiles.

The specific approaches used to analyze the MSV. equipment are based on generally acceptable analytical practices. The demonstration of qualification of solenoid valves, the junction boxes and terminal blocks, the Limitorque valve operator, and the Namco limit switch were based on direct comparisons of the MSLB and EQ thermal responses of "worst-case" equipment configurations. Where several models of equipment from the same manufacturer required qualification, the model having the smallest thermal inertia was used, adding conservatism to the overall conclusions.

4.2 COMPONENT EVALUATION

4.2.1 MSIV Solenoid

The qualification of the MSIV solenoid valves for the thermal effects of an MSLB was demonstrated by comparing the HEATING5 computer model results of the solenoid valve thermal response during an MSLB to the results of the computer model for the EQ testing. Heat flow between the environment and the MSIV solenoid was defined by heat transfer coefficients specified at the boundaries of the two-dimensional model of the valve.

The heat transfer coefficients used in the MSLB model were obtained from the Westinghouse COMPACT code analysis of the value vault environment. The coefficients accounted for convective and radiant heat transfer, which are expected to be the dominant modes. No condensing heat transfer is expected to take place because the housing temperature of the energized solenoid exceeds the satura ion temperature. The heat transfer coefficients used to determine the heat flow between the EQ test chamber and the MSIV solenoid conservatively represent the physical thermal dynamics of the EQ testing. For condensing heat transfer, four times the maximum coefficient from the Uchida correlation was used. Stagnant natural convection was modeled during the remainder of the EQ test when the surface temperaure exceeded the saturation temperature of the test chamber. During the spray periods in the qualification test profile, a laminar convection heat transfer correlation for film flow was used.

The major assumptions made in the MSIV solenoid analyses are acceptable because they can be expected to produce results which reasonably represent the actual thermal responses. The assumption of stagnant natural convection neglects any velocities in the test chamber that might be induced by the periodic addition of steam to maintain test conditions. However, the velocities are small and of short duration and consequently will not have an appreciable effect on the heat transfer rate to the MSIV solenoid. This assumption also conservatively accounts for heat flow during those periods when the surface temperature exceeds the chamber temperature since it reduces the rate at which the solenoid cools down.

The model used for the HEATING5 analysis of the MSIV solenoid was a two-dimensional cut through the coil (critical component), which minimized the thermal shielding between the atmosphere and the coil and thus maximizes the response of the coil temperature to the atmospheric transient. Analysis was performed in the rectangular coordinate system, which is acceptable since the thermal mass of the cylindrical coil is essentially conserved. No heat was assumed to be transfered through the valve body to the coil. This is reasonable in view of the relatively large thermal mass of the body.

The results of the MSIV solenoid valve analyses show that the thermal response of the coil to an MSLB is bounded by the response to the EQ test profile. On that basis, the coil may be considered qualified for the thermal profile for the MSLB event.

4.2.2 ASCO Solenoid

The thermal qualification of the MSVV solenoid valves for an MSLB was demonstrated by comparing the HEATING5 results of the MSLB analysis to the results of the model for the EQ test profile. Heat transfer between the MSLB environment and the ASCO housing surface and internals was accounted for in coefficients which conservatively describe the heat transfer process that occurs during the event. For an energized solenoid, heat transfer was assumed to occur by convection and radiation. Since the surface temperature of the energized solenoid is higher than the MSLB saturation temperature, no condensation is expected to occur. For the unenergized solenoid, a coefficient of four times the maximum Uchida coefficient was used to model condensing heat transfer. Heat transfer between the ASCO and the test chamber environment was defined as stagnant natural convection correlation since the ASCO was energized during the EQ test and its surface temperature was higher than the EQ saturation temperature at any given time.

The major assumption made in the HEATING5 modeling of the ASCO solenoid's responses to the MSLB and EQ environments was that no condensing heat transfer to an energized ASCO solenoid existed. This is reasonable since the surface temperature of an energized ASCO solenoid exceeded the saturation temperature.

The model of the ASCO valve was constructed in the polar system of coordinates. This method realistically represents the basically cylindrical assembly in which the components are arranged in a concentric manner. To maximize heat transfer to the coil and minimize thermal lag, a vertical section was taken through the valve so as to include the opening in the steel yoke of the coil's magnetic circuits.

The results of the HEATING5 analyses of the ASCO solenoid response to a MSLB and the EQ test profile show that both the housing surface temperature and the coil temperature responses to the MSLB are bounded by the respective responses to the EQ test profile. This is sufficient to demonstrate qualification to the thermal effects of a MSLB.

4.2.3 Junction Boxes

4.2.3.1 Terminal Blocks

Qualification of the MSVV terminal blocks inside junction boxes was demonstrated by comparing the HEATING5 results of the thermal response to the MSLB profile of a junction box containing the smallest terminal block to the results of the computer code for an identical configuration in the EQ test profile. The heat transfer methodology employed was similar to that used for other components already described.

The model that was constructed for the HEATING5 analysis of the terminal block inside a junction box was a two-dimensional section of the box taken through the terminal blocks and the concrete structure to which it is attached. A section thus chosen can be expected to minimize the thermal shielding between the environment and the terminal blocks. Conduction of heat away from the junction box to the concrete heat sink would be negligible for two reasons:

1. The concrete is a poor conductor of heat.

2. The attachment point between the junction box and the concrete wall would be expected to heat up faster than the junction box because of its relatively low thermal mass. Heat would thus tend to flow from the attachment to the junction box housing.

The results of the HEATING5 modeling of the thermal response of the terminal block inside a junction box indicate that the response to the EQ profile bounds the response to the MSLB profile. This evaluation represents adequate demonstration of MSLB temperature profile qualification.

4.2.3.2 Cables and Splices

Qualification of the cables and splices inside junction boxes was demonstrated by calculating the transient thermal response of the inside surface and the air contained within an empty junction box. The basis for this approach was that if the temperature of the inside surface and the air contained within an empty junction box during a MSLB is bounded by the EQ test profile for the most critical cable and the splice then these items are qualified for an MSLB event. This approach is reasonable and conservative since the peak temperature of any piece of equipment within the junction box cannot exceed the peak temperature of the inside surface of box. The heat transfer methodology used in this model was the same as for the terminal block with the junction box. The results indicate that the response of the empty junction box is bounded by the EQ profile for the most limiting cable and the splices.

4.2.4 Limitorque Valve Operators

Qualification of the MSVV Limitorque valve operators was established using the same approach as described for the equipmented discussed earlier. Heat transfer between the environment and the operator was defined by coefficients that were determined using the general approach described. No special assumption was made for this model.

The two-dimensional model of the Limitorque valve operator was the input for the HEATING5 code is an acceptable representation of the physical device for use in the thermal analysis. The analysis was conducted using a cross-sectional model made perpendicular to the motor through the electrical components. The motor was not included in the model due to its large mass, which would allow for it to act as a heat sink. The results of the HEATING5 modeling of the Limitorque valve operator indicate that the surface temperature during an MSLB is bounded by the surface temperature during EQ testing. This is to be expected because of the rather large mass and hence thermal inertia of the device. Consequently, the thermal responses of the operator internals would be similarly related. On this basis, thermal qualification of the Limitorque valve operator is demonstrated.

4.2.5 NAMCO Limit Switch

The approach to establishing qualification of the NAMCO limit switch to the thermal transient of a MSLB is the same as was used for devices previously evaluated. The application heat transfer methodology was consistent with a reasonably accurate prediction of the heat transfer mode which predominates at any given time. No device specific assumptions were applied to the NAMCO switch model.

The model constructed as the input for the limit switch is two-dimensional. A cut was taken so that heat entered the device from one side. An insulated boundary was defined on the other side. This boundary condition accounted for the fact that thermal diffusion across the metal components on the other side of the boundary was much slower than on the side where the critical plastic components were located due to a thermal inertia of the parts. Accordingly, for practical purposes, it can be assumed that no heat crosses that boundary. Consequently, the location of the cut is justified. The results of the analysis indicate that although the surface temperature of the limit switch housing during a MSLB exceeded the temperature during EQ testing, the response of the terminal block (critical component) to the EQ test profile bounds the response of the block to a MSLB.

4.2.6 Cable in Conduit

Qualification of the cable in conduit was demonstrated by a method similar to that used for the other devices. A direct comparison of the cable-in-conduit MSLB profiles with EQ test profiles could not be performed because only the cable was subjected to EQ testing. The modeling was therefore performed using the contact temperature inside the conduit. This approach provided conservatism since the cable is subjected to more severe condition during EQ testing than it would be during an MSLB.

The specific assumptions made for the HEATING5 analyses can be expected to give results with reasonable accuracy. The models neglected the thermal resistance between the cable material layers. This approach is conservative in that it would give a faster heatup rate for the cable. A gap resistance of 10 Btu/ft^2 hr/°F was used to model the contact between the cable and the conduit. Experimental results have been reported to indicate contact resistance between 2 to 5 Btu/ft^2 hr/°F; therefore, a value of 10 Btu/ft^2 hr/°F is conservative, permitting faster heat transfer from conduit to cable. The model used to determine response of the cable in conduit was constructed in the rectangular coordinate system. This approach simplied the modeling since the circular cross sections of the conductors were not concentrically arranged. The conservatism of the modeling was maintained by keeping the area bounded by the rectangular representations the same as that of the physical component since this increases the area and hence the heat flux to the internal components. The effect of the thickness of hollow sections such as the conduit and wrap is negligible because the thickness is small compared to the other dimensions.* Representation of solid internal sections, such as conductors, in rectangular coordinates is justified on the basis that the temperature of the component is directly proportional to the atmospheric temperature in both coordinate systems. Consequently, if the response to an MSLB is bounded by the response to the EQ test profile in the rectangular coordinate system, the same relationship can be expected to hold for a model constructed in the polar coordinate system.

The results of the analysis of the response of the cable in conduit to an MSLB event and the response to the EQ testing indicate that although the peak temperature of the conduit during the MSLB event exceeds the peak temperature cable temperature during EQ testing, the cable surface thermal response from the MSLB is bounded by the cable surface temperature from EQ testing. The cable can thus be considered qualified for the MSLB temperature profile on this basis.

5. CONCLUSION

Based on the above evaluation, there is reasonable assurance that the heat transfer modeling accurately reflects component temperatures during an MSLB. Where assumptions were required during the modeling, the Licensee maintained a conservative approach, providing additional assurance that the predicted component temperatures during an MSLB approach a worst-case scenario. Therefore, the Licensee has effectively demonstrated that the components located in the main steam valve vaults identified in Table 1 would not exceed their qualified temperature profile during an MSLB and may be considered qualified for this condition.

6. REFERENCES

1. General Design Criteria 1 and 3 of Appendix A to 10CFR50

2. Sections III, XI, and XVII of Appendix B to 10CFR50

3. 10CFR50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"

4. Regulatory Guide 1.89, Rev. 1, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"

5. "Sequoyah Nuclear Plant - Electrical Equipment Qualification for a Main Steam Line Break in the Main Steam Valve Vaults," Tennessee Valley Authority Submittal with Letter Dated August 13, 1986

6. "Sequoyah Nuclear Plant - Additional Information on Sequoyah's Equipment Qualification Under Superheat Conditions," Tennessee Valley Authority Submittal with Letter Dated December 23, 1986

7. "Sequoyah Nuclear Plant - Additional Information on Sequoyah's Equipment Qualification Under Superheat Conditions," Tennessee Valley Authority Submittal with Letter Dated February 17, 1987

8. Final Work Assignment No. 14, Transmitted by S. Bajwa (NRC) to Dr. S. Pandey (FRC) on November 6, 1986

APPENDIX D

TECHNICAL EVALUATION REPORT (TER)
RELATED TO THE WELDING CONCERN PROGRAM
AT TVA's SEQUOYAH UNITS 1 AND 2

TECHNICAL EVALUATION REPORT (TER)
RELATED TO THE WELDING CONCERN PROGRAM
AT TVA'S SEQUOYAH UNITS 1 AND 2

AUGUST 1986

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EXECUTIVE SUMMARY

Specific concerns brought up by TVA employees indicated several areas of the TVA welding program at the Sequoyah Nuclear Units 1 and 2 (construction and operation) which, if accurate, question the adequacy of the program. This, coupled with the utility's review of various quality indicators (NRC inspections, audit findings, etc.), suggested that conditions existed in the TVA welding program which did not meet industry/regulatory codes or standards.

The utility's approach to resolution of the Employee Concern Program was to evaluate the concerns with a three-way investigation. The first evaluation consisted of a review of a sample of documents from the plant which were compared to TVA's commitments to the USNRC. In the utility's Phase I report, they believed that these commitments had been met with the exception of preweld inspections.

The second approach to the resolution of the Employee Concern Program by TVA was composed of two independent audits of the Sequoyah welding program. The first audit performed by Aptech Engineering consisted of an in-depth review of the two units' PSI/ISI programs. This audit, in general, concluded that the welding program contains the necessary controls to ensure a high quality of welds. An additional independent audit of the welding program at Sequoyah was performed by Bechtel Engineering. The Bechtel team expended thirty auditor weeks (five-member team) and audited all aspects of the welding program (both construction and operations). This audit disclosed no findings relative to any employee concerns, but did observe that many TVA documents were ".... confusing, overlapping, repetitive and unclear".

The third segment of the TVA investigation consisted of a sample reinspection of 333 Class 3 piping welds, 15 spiral duct welds, and 403 structural joints by TVA inspectors. As a result of this reinspection, the utility concluded that all of the reinspected weld joints meet design requirements and that additional reinspections (by the utility) are not required.

The USNRC's evaluation of TVA's response to the Employee Concern Program consisted of reinspections at the plant (both the Region 1 NDE Van and a combined NRC and BNL Welding Team audit), and the formation of an expert welding team, under BNL contract, to review TVA's resolution of welding issues and to make recommendations on the adequacy of TVA's corrective action. This team consisted of five independent experts in the fields of welding/materials/structural engineering. The team evaluated the TVA investigation and responses to 117 concerns (either specific or generic) relative to the Sequoyah units. These evaluations found areas of the Sequoyah welding program which suffered programmatic breakdown. Various questions on these areas have been transmitted to the USNRC for forwarding to the utility. Since there were these areas of "programmatic breakdown", it becomes necessary to address the question of the adequacy of the Sequoyah welding utilizing a hardware inspection approach.

Three NRC inspections involving the Sequoyah units were performed. The first NRC inspection was performed during February 18-28, 1986. This inspection included 417 inspector hours on site to evaluate TVA's reinspection

The second inspection took place in February 1986, by the NDE Van. This involved the inspection of 40 pipe weldments (Class 3, either dye penetrant or magnetic particle inspection), 361 structural weldments visually inspected, and 46 piping welds (ANSI B31.7) visually inspected. This report concluded that "..... the NRC findings were representative of the types found by the licensee."

The third NRC reinspection took place in June-July 1986 and involved 30 pipe welds, 502 pipe support welds, 31 instrument tubing welds, 120 instrument support welds, 130 structural welds (electrical), 280 HVAC support welds, and 100 structural welds and generally concluded that the licensee complied with the governing codes and specifications for the welds examined.

A review of the evaluations and inspections performed to date have shown that the Sequoyah units have suffered some areas of "programmatic breakdown," but the hardware itself does not have any defects of great detriment or magnitude. This being the case, if questions posed to TVA are answered to the NRC's satisfaction, then the welds at the Sequoyah units are deemed "suitable for service."

The expert welding team has also sent separate summaries of their technical opinions of the exployee concern program for the Sequoyah units, which is also part of the TER.

1.0 INTRODUCTION

1.1 Background

Various quality indicators (e.g., NRC inspections, audit findings, nonconformance reports, etc.), manifesting themselves during the construction of the TVA nuclear units, directed the utility toward possible existing conditions in their welding program which did not meet industry/regulatory codes or standards.

Specific concerns brought up by TVA employees also indicated several areas of the TVA welding program (both construction and operations) which, if accurate, additionally question specific practices at the various TVA units.

The NRC requested a meeting with TVA in order to discuss these welding program concerns and provide a listing of various comments and questions by the regulatory body on the adequacy of the TVA welding program. The utility evaluated these comments and presented a two-phase plan to the NRC at a public meeting on January 7, 1986. These two phases would be applicable to each of TVA's nuclear plants and would involve:

- Ensuring that the TVA welding program which is currently in effect adequately reflects the regulatory requirements and TVA's commitment to same.
- Evaluation of the implementation of the TVA welding program and verification that field weldments are adequate for service.

The first phase of the Action Plan is stated in Volume 1 of the Project Review Plan:

- Review TVA commitments to NRC
- Verify that written program reflects commitments:
  - Determine that welding related commitments are reflected in design output.
  - Determine that construction and nuclear operations programs, as applicable, reflect design output and quality requirements.
- Assemble welding program quality indicators (including employee welding concerns) by type and plant.
- Analyze and evaluate effect of quality indicators on programs.
- Issue adequacy statement regrading written programs to implement/control welding.

The Phase II program is broken into two parts:

- Independent Audits
- Hardware Inspections and Corrective Actions

1.2 Independent Audits

This part of the program is to encompass an in-depth auditing of the utility's welding program. It is to be approximately one month in duration accomplished by a five man audit team. The audit is to cover ASME and AWS, as well as non-ASME safety-related applications at the site.

1.3 Hardware Inspections and Corrective Actions

A reinspection program was devised by TVA, with NRC concurrence, for selective structures of systems from six different groups. This program consists of inspections of a minimum of 100 welds from each group. The six groups include:

1. ASME Class 3 and ANSI B31.1 welds and attachment welds
2. Supports associated with Group 1 (above)
3. Cable tray/conduit supports
4. Miscellaneous structures
5. HVAC support welds
6. Butt welds on spiral welded ducting

The pipe welds were to be reinspected to ANSI B31.1 or B31.7 using both visual and nondestructive (surface only) examination and the structural welds examined in accordance with NGIG-01 [4].

The results of these inspections and audits are described later in the TER and were documented in the "Tennessee Valley Authority Welding Project, Sequoyah Nuclear Plant, Phase 1 and Phase 2 Review and Program Results".

2.0 FORMATION OF THE EXPERT WELDING TEAM

The excessive number of employee concerns expressed by TVA employee's regarding the utility nuclear units generated sufficient concern in the USNRC to form a triumvirate NRC team (NRR, I&E and Region II) manage the overall NRC staff activities including the TVA resolution to their welding concern program.

Part of the NRR responsibilities was to: "Contract with Brookhaven National Laboratory (BNL) to constitute an expert team to review TVA's resolution of welding issues and to make recommendations on the adequacy of TVA's corrective actions, as appropriate."

The implemention of these responsibilites was realized in the form of two contracts to BNL entitled "Evaluation of Welding Concerns at TVA Operating Reactors," FIN A-3839, and "Evaluation of Concerns at TVA Near-Term Operating Licenses," FIN A-3836.

The contract under FIN A-3839 is specific to this TER and has as its objective: The formation of a panel of independent experts in the field of welding/materials/structural engineering to evaluate the utility response and action plan for addressing the employee welding concerns. The work requirements for the contract are:

Task 1: Sequoyah Nuclear Power Station

1. BNL will contact, issue and administer subcontracts to various welding/structural engineering experts in order to form the welding team.

2. BNL will coordinate the receipt and appropriate distribution of TVA's resolution of the welding concerns and various supporting documents to the team necessary to develop a comprehensive and informed evaluation of the TVA welding program.

3. BNL will convene, coordinate and schedule team meetings as necessary to meet the program objectives.

4. BNL will review and evaluate the TVA welding program as a full participant of the welding team.

5. BNL will evaluate and categorize welding concerns received from TVA and distribute as necessary to the team members.

6. BNL will coordinate and schedule field inspections if necessary for team members to assess the program implementation and the structural integrity of affected components. The team is composed of experts in the various fields involved with welding. The welding team secretary is Carl J. Czajkowski, a BNL Staff Research Engineer specialized in failure analysis, welding and metallurgical investigations. Every effort was made to verify that this team did not have a preconceived bias relative to the utility and the NRC. Based on the above, the following list was proposed as the team of consultants:

William D. Doty	An independent consultant, formerly a Technical Director of U.S. Steel's Research Center; author of several books and numerous papers; a Member of Welding Research Council and Pressure Vessel Research Committee.
Carl E. Hartbower	An independent consultant, formerly Chief Welding Engineer at FHA, NRL; AWS D1.1 member.
Paul E. Masters	An independent consultant, formerly Chief Welding Engineer at American Bridges Co.; advisory member to AWS D1.1 Committee.
William H. Munse	Professor Emeritus of Civil Engineering, University of Illinois; member of AWS & AISC Code Committees.

Robert Stout	Dean Emeritus Lehigh University, specialized in welded steel structures; author of several books and numerous technical papers on the subject.

More complete copies of the resumes of the welding team are listed as Attachments (1-6) to this TER.

Additionally, in April 1986, the BNL effort was enhanced by the addition of Mr. Milford H. Schuster (resume - Attachment 7), formerly of Long Island Lighting Company.

As of the writing of this TER, the team has had two group meetings (totaling 3-1/2 days of effort) and each member has been to the Sequoyah site for discussions and weld inspections (Attachments 8 and 9). Additionally, all information relative to the concerns has been sent and reviewed by the team. A three-day meeting was also held between Messrs. C. Czajkowski and M. Schuster (BNL), D. Smith (NRC/NRR) and G. Georgiev (NRC/I&E) on May 13-15, 1986. At this meeting, the concerns relating to Sequoyah Units 1 and 2 were categorized. The categorization was made in six groups:

1. Welding Procedures
2. Welder Qualification/Training
3. Welding Inspection
4. Weld Design and Configuration
5. Filler Metal Control
6. Miscellaneous/One of a Kind

The first five groups were considered to be essential elements in any welding program which would be necessary to assure that a welding program was adequate to produce a sound weld as an end product. Into the six categories, all of the concerns (both generic and Sequoyah-specific) were divided. The total input for the concern listing came from three sources:

1. A Franklin Institute listing dated 3/21/86 (F).
2. A list supplied by TVA as the "Concerns" applicable to the Sequoyah units (T1).
3. The contents of Appendices 5.1 and 5.2 of the TVA Welding Project Phase II Report-Volume 3 (T2).

These three sources were cross-checked one against another and a total listing of concerns generated. The concerns were placed in the categories with these results:

1. Welding Procedures - 0 concerns
2. Welder Qualification/Training - 27 concerns
3. Welding Inspection - 48 concerns
4. Weld Design and Configuration - 7 concerns
5. Filler Metal Control - 26 concerns
6. Miscellaneous/One of a kind - 9 concerns

This totals 117 concerns (either generic or specific) for the Sequoyah Nuclear Units. These concerns and the utility response to these concerns were evaluated in detail by the welding team and are contained in Section 3.0 of this TER. There were 26 specific concerns relative to Sequoyah (T2) with the balance being Watts Bar concerns with possible generic implication for the Sequoyah units.

3.0 EVALUATION OF EMPLOYEE CONCERNS

It had been previously determined (Section 2) that there were 117 employee concerns considered applicable to the Sequoyah Units and that these had been divided into one of six categories. This section of the report will list the employee concerns for each of these categories, the reference for how the concern was determined (Franklin Institute Report (F), TVA original submittal (T1), Appendices 5.1 or 5.2 of final report (T2)), the TVA Report which answered the concerns, and a brief description of the concern. A welding team evaluation of the concern is also included in this section.

It should be noted that the exact number of concerns may differ from various reports and lists due to the continuous updating and overlapping of concerns (generic or specific). This TER has therefore listed all of the concerns that the BNL team consider to be Sequoyah applicable. Even though some differences may occur, it is believed that all major categories (fit up inspection, bad electrode, etc.) of the concern program have been evaluated.

3.1 Category 1 - Welding Procedures

There were no concerns found to be specific for this category.

3.2 Category 2 - Welding Qualifications/Training

This category had 27 concerns associated with it, as listed on the following page.

Concern Number	Reference	Report Number Responding to Concern	Brief Description of Concerns
EX-85-042-003	F,T2	WP-03	All positions based on WQ 2Q position
EX-85-021-002	F,T1,T2	WP-03	No objectives evidence WQ
IN-85-426-002	T1,T2	WP-03	Updating welder certs inadequate based on bead on plate
IN-85-346-003	T1,T2	WP-03	Updating welder certs
IN-85-480-004	F,T2	WP-03	Updating certs irregularity
PH-85-052-002	F,T1,T2	WP-03	Updating and backdating of welder certs
PH-85-052-X03	F,T1,T2	WP-03	Welder certs classified
IN-85-352-001	T2	WP-03	Welder cert updating-just burned rod
IN-85-424-011	T2	WP-03	Welder cert updating-card stamped
IN-85-493-004	T2	WP-03	Welder cert inadequate
IN-85-532-005	T2	WP-03	Welder cert recertified without having used process
IN-85-835-002	T2	WP-03	Welder recerts by stamping
IN-85-778-001	T2	WP-03	Welder recerts updating
IN-85-940-X04	T2	WP-03	Welder recerts updating
IN-85-113-003	T2	WP-03	Welder recerts stamped every 90 days; no rod burning
IN-85-770-002	T2	WP-03	Update on welder certs
IN-85-627-036	T2	WP-03	Welder certified/backdating
IN-85-706-001	F,T1,T2	WP-07	Welder insufficient welder training and exp.
XX-85-045-001	F,T2	WP-07	Insufficient welder training
XX-85-049-001	F,T2	I-85-135-SQN	Updating and backdating of welder certs
XX-85-049-X03	F,T1,T2	I-85-135-SQN	Welder certs classified
XX-85-101-006	F,T1,T2	ERT XX-101-006	Welder performed welds without proper certs
SQM-6-005-001	F,T2	I-86-115-SQN	Welder passed though qualification fa[illegible]ified
SQM-6-005-X02	F,T2	I-86-115-SQN	Welder certs records falsified
XX-85-088-003	T2	ERT XX-088-003	Alteration of welder certs by correction fluid
XX-85-088-X04	F	ERT XX-85-088-X04	Correction fluid on welder certs
XX-85-088-001	F,T2	ERT XX-088-X04	Welder certs altered (Knoxville) correction fluid

3.2.1 Seventeen of the 27 concerns were answered by Welding Project Generic Employee Concern Evaluation Report WP-03. This report addressed the following five issues:

Issue #1: Welder Performance Qualification (WPQ) continuity records have been backdated.

TVA Evaluation: Welder Performance Qualification Continuity Records have not been backdated. This issue is not substantiated. This issue is closed by this report.

Issue #2: WPQ continuity records have been falsified.

TVA Evaluation: Welder Performance Qualification Continuity Records have not been falsified. A detailed investigation of these issues was performed by NSRS and documented in NSRS Report I-85-135-SQN. Both these issues were not substantiated. The investigation did, however, discover that program implementation had been deficient and that NO had already taken steps to correct identified deficiencies. The Bechtel SQN Implementation Audit conducted in January 1986 determined that both OC and NO programs for these activities had been effectively implemented prior to the NO audit. Based upon this analysis, these issues are closed pending the completion of the corrective actions regarding review of Welder Performance Qualification Records as outlined in NSRS Report I-85-135-SQN.

Issue #3: The WPQ continuity program is inadequate because there is no objective evidence to confirm actual process usage when WPQ continuity records are stamped by QC.

TVA Evaluation: This issue was not substantiated because it related to WBN practice. All welders who have transferred to SQN from other sites have successfully passed a requalification test administered at SQN. Implementation deficiencies discovered by SQN, NO, QA have had corrective actions initiated. This issue to be closed based on the above actions.

Issue #4: The WPQ continuity program is inadequate because continuity can be maintained by running one weld bead.

Issue #5: A one-position test plate is not sufficient to reinstate all WPQ tests.

TVA Evaluation: Issues 4 and 5 are acceptable practices and are to be closed on that basis.

Expert Welding Team Evaluation (All 5 Issues)

In general, the investigation (NSRS-I-85-135-SQN) appeared to adequately cover the essential bases for the TVA evaluations. There is a need for more information, however, on the status of corrective action implementation of item I-85-135-SQN-02 from the NSRS report. The welding project report does not mention this item in its evaluations of the problem.

I-85-135-SQN-02 - Corrective Action Backfit Evaluation

"TVA formal corrective action processes such as corrective action reports, nonconformance reports, etc., should be evaluated to include a backfit evaluation provision to determine if the identified deficiency requires such action to provide substantial, additional protection for the public health and safety or the common defense and security."

3.2.2 WP-07 was used to address two employee concerns, IN-85-706-001 and XX-85-045-001. The issues involved in these concerns were:

1. The TVA Welder Training Program is inadequate for nuclear construction.

2. Welder performance qualification tests do not test a welder's overall ability.

TVA Evaluation

The utility felt that these concerns were unsubstantiated for the following reasons:

1. There is no requirement relative to welder training programs.

2. The base requirement for welder skill is the Welder Performance Qualification Test Program.

3. The Welder Performance Qualification Test Program is outlined in both the OC and NO, QA programs.

4. The Bechtel SQN Implementation Audit has established that these programs have been and are being effectively implementated by OC and NO.

5. No indication of a generic welder skill problem was discovered by the SQN Reinspection Program.

Expert Welding Team

The expert welding team agreed that sufficient investigation and followup had been perfomed by the utility. It was additionally agreed that there is no requirement for a welder training program by current codes or standards, since the "proof test" of a welder making a sound weld has always been his/her performance qualification test. Additionally, the welder performance test was never intended as a gauge of a welder's overall ability; it is merely a method of determining the particular welder's ability to produce a "sound weld" with a specified procedure.

3.2.3 NSRS Investigation Report No. I-85-135-SQN was used to evaluate concerns XX-85-049-001, X03. The issues involved were:

Issue #1 - Sequoyah: Welder certifications have been updated for welders who did not meet update requirements or backdated to give appearance of requirement compliance.

Issue #2 - Sequoyah: Welder certification card falsified. Construction Department concern. CI has no more information.

TVA Evaluation: The utility feels that although 1 above was substantiated, the two concerns can be closed out for the following reasons:

1. The concern that the welder update (continuity) requirements were not being met was substantiated and had been identified recently in a QA audit finding. All active welder records have been properly updated by supporting documentation or the welder retested.

2. The concern that records may have been backdated in order to give an appearance that the welder was qualified could not be substantiated. There were some clerical-type errors where incorrect dates were entered on welders' records, but these were corrected when a review identified discrepancies between welder continuity record sheets and supportive documentation (i.e., welder performance qualification record). In addition, the toolroom clerk may have missed entering weld filler material draws on a welder's record and correctly updated the continuity records later, but this is not considered backdating. No evidence was found that indicated falsification of records had occurred.

3. There appears to be no safety concern since all active welder records were either corrected or readily restored to requirements. Also, all safety-related welding is independently inspected per an approved QA program.

4. Corrective Action Report SQN-CAR-85-09-14 (Ref. 13) did not address the consequences of the previous (nonactive) welder continuity program.

Expert Welding Team

The expert welding team agreed with the corrective actions and investigations associated with the welder falsification concern. It did not, however, feel that sufficient information was presented on the eight welders identified in I-85-135-SQN (e.g.):

1. Did they pass their retest the first time?

2. Did TVA inspect any welds made by these welders while they were "out of certification?"

3. How long were they out of certification?

4. The eight welders found out of certification were out of all welders reviewed or just the twenty-five?

3.2.4 Concern XX-85-101-006 was investigated by ERT Report XX-101-006. This report had as its issue that a welder performed welds without having the proper certification. The report substantiated the concern and had four recommendations.

TVA Evalation

Appendix 5.2 of the TVA Report states "WP has determined that this analysis missed the point of the concern. WP recommends this concern not be substantiated.....".

Expert Welding Team

Before any evaluation can be made on this concern or report, more information is required from TVA on why the report and recommendations are dismissed.

3.2.5 NSRS Report No. I-86-115-SQN was written in response to concerns SQM-6-005-001,X02. The issues involved were:

1. Whether a known welder was capable of making proper welds.

2. Whether there was collusion to certify this welder resulting in falsified records.

TVA Evaluation:

1. The concern that the welder in question was incapable of making proper welds was partially substantiated by virtue of the poor performance evaluation of work performed in the turbine building. The welder does make adequate welds in the shop.

2. The concern that the welder was passed by collusion between engineering and the general foreman resulting in falsified records could not be substantiated.

Expert Welding Team

The concern appeared to have had adequate investigation and corrective action by TVA. This concern appears to be a management problem and not a hardware or a safety issue.

3.2.6 The last three concerns in this category XX-85-088-001, 003 and X04 all involved the use of correction fluid in altering welding certifications.

TVA Evaluation:

XX-85-088-X04 and 001 were substantiated by a QTC report (same number as concern). The investigation showed that no substantive information was obliterated.

Concern XX-85-008-003 was considered unsubstantiated by the investigation report.

Expert Welding Team

The welding team agreed that -003 was unsubstantiated from the available data reviewed. It should be noted that in the two cases of substantiation, no corrective action was considered necessary. In the unsubstantiated case, the review was limited to only those "hard copies" available, a limited scope.3.3

3.3 Category 3 - Welding Inspection

There were a total of 48 concerns in this category.

Concern Number	Reference	Report Number Responding to Concern	Brief Description of Concerns
IN-85-282-002	T2	WP-11	Surface grinding
PH-85-040-001	T2	WP-02	Inspection thru paint
WI-85-013-003	T2	WP-02	Inspection thru paint
WI-85-041-006	T2	WP-02	Inspection thru paint
WI-85-041-008	T2	WP-02	Inspection thru paint
IN-85-458-001	T2	WP-02	Inspection thru paint
IN-85-767-003	T2	WP-02	Painted welds
WI-85-030-008		WP-02	Inspection thru paint
IN-85-406-003	T2	WP-04	No inspection tools
IN-85-134-002	T2	WP-04	No tools
IN-85-007-001	T2	WP-04	No tools
IN-85-007-003	T2	WP-17	Vendor welds
IN-85-657-001	T2	WP-17	Vendor welds
IN-85-127-001	T2	WP-17	Bergen Patterson/Vendor weld appearance
SQM-5-001-001	T2	WP-16	Undersized socket welds
SQM-5-001-002	T2	WP-16	Preweld inspection by foreman
IN-85-212-001	T2	WP-16	Weld inspection
IN-85-406-002	T2	WP-09	No inspection criteria
PH-85-012-X03	F,T2	WP-05	Deleted HVAC
XX-85-069-001	T1,T2	ERT Report	NDE certs
XX-85-069-002		ERT Report	NDE certs
XX-85-069-003	T1,T2	ERT Report	NDE certs
XX-85-069-006		ERT Report	NDE certs
XX-85-069-007	T2	ERT Report	NDE certs
XX-85-069-X13	T1	ERT Report	NDE certs
XX-85-069-003-R1	T2	I-85-738-SQN	NDE certs
XX-85-069-X05	T1,T2	I-85-738-SQN	NDE certs
XX-85-069-X07	T2	I-85-738-SQN	NDE certs
XX-85-108-001	T2	I-85-776-SQN	No inspections performed
XX-85-108-002	T2	I-85-776-SQN	No inspections socket welds
IN-85-001-005	F	I-85-753-WBN	Vendor welds
XX-85-054-001	T2	I-85-346-SQN	QC inspector sign off
XX-85-065-001	T2	I-85-750-SQN	Remote inspection
XX-85-083-001	T2	I-85-652-SQN	Poor welding inspection
XX-85-102-011	T2	I-85-735-SQN	Different programs
IN-85-981-001	F,T1,T2	WP-06	Poor training of inspector
WI-85-041-002	F,T1,T2	WP-06	Inspector quals.
IN-85-476-004	F,T2	WP-06	Inspector quals.
SQM-6-008-001	F		Undersized socket welds
WBM-5-001-002	T2	WP-16	Preweld inspection by foreman
WI-85-081-007	T2	WP-06	Inspector not qualified

3.3 (Cont'd)

Concern Number	Reference	Report Number Responding to Concern	Brief Description of Concern
XX-85-098-001	T2	WP-18	Laminated piping in Unit 2
NS-85-001-001	T2	WP-02	Inspection of welds thru paint
IN-85-271-001	T2	WP-02	Surface grinding of welds
WBM-5-001-001	T2	WP-16	Preweld inspections
BEM-5-001-001	T2	WP-16	Preweld inspections
BEM-5-001-002	T2	WP-16	Preweld inspections
BFM-5-001-002	T2	WP-16	Preweld inspections

3.3.1 Nine of the concerns were responded to by WP-02. The issues involved in these nine concerns were:

1. Specifications allow inspection of welds after painting or coating with inorganic zinc primer in violation of FSAR and AWS requirements after tests demonstrated that adequate inspections could not be made.
2. There may have been/were welds inspected through primer.
3. Inspectors did not understand thickness provisions for primer and could not have performed an adequate inspection.
4. NRC involvement in approving procedure for inspecting welds through paint.

TVA Evaluation

The four issues were considered not substantiated for the following reasons:

1. Procedures were and are in effect for OC and NO, respectively, which provided for initial inspection of welds prior to painting.
2. The Bechtel audit established that those procedures were effectively implemented for both OC and NO.
3. NRC does not formally approve or disapprove specific construction practices.

Expert Welding Team

The team believed that sufficient investigation had been performed on these concerns due to the fact that inspection through paint was not allowed at SQN. The team agreed with the utility's findings.

3.3.2 Three concerns were addressed by WP-04. The major issue of these three concerns was:

1. Welding inspection tools were not issued to welding inspectors.

TVA Evaluation

The utility considered these concerns unsubstantiated for the following reasons:

1. Weld inspection tools were and are furnished to welding inspectors.

2. More sophisticated inspection tools were furnished to welding inspectors as they became commercially available and as the need for more percise verifications of weld attributes was identified through program improvements.

3. Records were and are available which document the purchase and distribution of these tools.

Expert Welding Team

It was felt that sufficient investigation was performed by the utility to close the item. It was also deemed prudent that a definite number of available records should be reviewed by the I&E Audit Team (July 1986) to verify issuance of inspection tools. This could not be accomplished and should be reviewed at some later date by the NRC.

3.3.3 Three concerns were aimed at vendor welds and were addressed by WP-17-SQN. The issues involved were:

1. Vendor welds are not of the same quality as TVA field welds.

2. Vendor welds are not inspected in the field.

TVA Evaluation

The utility investigated and substantiated these vendor welds and drew the following conclusions:

1. The employee concerns are substantiated as they relate to the observed general condition of vendor welds.

2. A similar problem had been identified, reported, documented, and dispositioned in accordance with applicable QA program requirements at WBN.

Expert Welding Team

The team was in agreement that the investigation and follow up was adequate but felt that some additional information was required:

1. Were B-P hangers rejected at receipt inspection or post-facto as part of the concern investigation?

2. What was the inspection criteria for the hangers at the plant? At the vendor's shop?

The above two questions are not in the scope of the EWT but would more adequately answer the issues raised.

3.3.4 Nine concerns were evaluated by WP-16-SQN. The issues involved with these nine concerns are:

1. Do uncertified welder foremen perform perform preweld inspections?
2. Is this a violation of the TVA Quality Assurance Program?
3. Is this a violation of ANSI N45.2.5 requirements?

TVA Evaluation

The issues considered in these concerns are not substantiated for the construction era at SQN but are substantiated for the Nuclear Operations era due to the following reasons:

1. SQN construction had a program in place which contained procedures which adequately addressed the elements of ANSI N45.2.5.
2. Nuclear Operations has identified this issue as an area of noncompliance and has documented this noncompliance in accordance with QA program requirements. Corrective actions have been implemented which completely address this issue and confirm no effect on hardware.

Expert Welding Team

The team believed that since the utility has now committed to fit up inspections (NO) by certified QC inspectors that adequate corrective action has been implemented. The utility has not adequately answered the question for construction since they did commit to N45.2.5. This standard does state "This inspection shall include visual examinations of preparations..." in section 5.5 entitled "Welding." Additionally, Section 2.4, "Personnel Qualifications" requires that "personnel performing tests and inspections required by this standard shall be qualified in accordance with ANSI N45.2.6. Personnel performing field inspection and testing activities shall be certified for Level I capability...". More information is required of the utility on this apparent violation of the ANSI standard for the construction phase of SQN. This instance might also be a possible violation of Criteria X and I of 10CFR50, Appendix B.

3.3.5 Concern IN-85-406-002 was answered by WP-09. The concern expressed was:

1. Prior to 1979, there was no specific weld inspection criteria for use by inspection personnel.

TVA Evaluation

This issue was not considered substantial for the following reasons:

1. Inspection procedures which delineated code and standard requirements were in effect at SQN for OC.

2. The Bechtel SQN Implementation Audit provides an independent verification of the adequacy of these procedures.

Expert Welding Team

The investigation and explanation by the utility adequately answered the concern.

3.3.6 Two concerns were related to spiral-welded pipe and had the following issues answered by WP-05-SQN.

1. EGT piping is too close to wall for adequate access for welding.

2. Welds should be welded and inspected from the inside of the pipe to assure adequacy.

3. Welding and brazing inspection may have been/was deleted from the QA program without adequate justification.

TVA Evaluation

Issues 1 and 2 were substantiated due to:

1. It has been determined by direct inspection that there are areas of spiral weld duct which are not welded on the outside diameter because of the close proximity to walls and other barriers in similar systems.

2. It has been determined by direct inspection that welds have been made and subsequently inspected on the inside diameter of the spiral weld pipe where there are corresponding areas which are not welded on the outside diameter. Issue three was not substantiated because there was a program in place for welding inspection on duct work and duct supports doing construction at SQN.

Expert Welding Team

The follow-through and investigation by the utility was adequate to close out these concerns.

3.3.7 Six concerns related to NDE inspectors were answered on an ERT investigation report which had the following issue:

1. Employees OJT (on the job training) records have been falsified.

TVA Evaluation

Appendix 5.2 of the TVA final report, page 1 of 2, lists concerns XX-85-069-001, 001-R1, X05 and X07 as being closed by NSRS Report I-85-373-NPS with "No falsification of records was substantiated. WP concurs with report recommendations." The ERT report addressed six concerns on OJT and determined that the concern was substantiated and had the following recommendations.

The results of this investigation clearly indicated both a programmatic breakdown and falsification of records within the TVA NDE training/certification program. Based on these findings, the following is recommended:

1. The turnover of this report to the Office of General Counsel (OGC) for investigation of legal wrong doing, and

2. TVA issue an immediate stop work order against the certification of NDE inspectors until such time as the situation can be evaluated and corrective action taken.

Expert Welding Team

No additional evaluation can be done until more information is received regarding these recommendations.

The information needed is:

1. Were any MT/PT/Visual reinspections done on any of the "uncertified/unqualified" inspectors?

2. Volumetric examination was not really addressed. What is the impact on ISI/PSI previously inspected welds? Did any "qualified" individuals reinspect any "unqualified/uncertified" inspectors'previously accepted work?

3.3.8 Three additional concerns on NDE certification were addressed by NSRS Report I-85-738-SQN. This report dealt with the following issue:

1. Sequoyah: Very often, rejected items are accepted by someone other than a supervisor or a higher level (grade). To illustrate the point, CI stated that the supervisor will send another examiner/inspector with less qualification and experience to reexamine the once rejected items and will get acceptance.

TVA Evaluation

The utility found that the concerns were not substantiated based on the following:

1. Previously rejected items have been accepted by a second examiner who was a certified Level III examiner. On each occurrence the examiner would note on the NOI and the corresponding data sheet the basis for acceptance of the item which, in effect, voided the NOI. This process does not appear to violate any specific regulatory requirement or ASME rule applicable to the ISI program.

2. Previously rejected items have been accepted by Level II NDE examiners who were designated as Acting NDE Unit Supervisor. The acceptances occurred when Part III of the associated NOI was completed by the acting supervisor. This process does not appear to violate any specific regulatory requirements or ASME rule applicable to the ISI program.

3. One NOI was found to have part III closed without documenting that all of the corrective action had been completed for the affected item. This occurrence is a failure to meet the intent of existing program requirements (reference 6c). This NOI does not clearly fit the CI's description; however, no other examples could be found which support the stated concern.

Expert Welding Team

Response appears adequate if the follow up and corrective action is completed (TVA Evaluation 3).

It should be noted that the procedures for handling NOI's had misleading and insufficient information to make the handling of NOIs consistent. This appears to be a symptomatic condition to many TVA procedures. The confusing and misleading procedures were also discussed by the TVA sponsored Bechtel Audit performed at SQN.

3.3.7 Two concerns regarding lack of inspections of socket welds were evaluated in Report I-85-776-SQN which dealt with the following issues:

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

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

TVA Evaluation

The utility determined that the concerns were unsubstantiated for the following reasons:

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

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

Expert Welding Team

It was felt that the concerns were unsubstantiated and that the utility's programmatic close out of the items satisfactory.

3.3.10 Concern IN-85-001-005 was addressed by NSRS Report I-85-753-WBN which had as its issue:

1. Vendor welds were bought off even though they exhibited "shoddy workmanship."

TVA Evaulation

The concerns were substantiated and an engineering disposition of the affected parts was "use as is."

Expert Welding Team

Utility response was adequate. Will be reviewed for WBN Project.

3.3.11 Employee concern XX-85-054-001 was addressed by NSRS Report I-85-346-SQN which had as its issue:

1. Sequoyah: QC holdpoints are signed off by craftsmen (craft known) performing the work. Personal friendship between inspectors and craft allow this to occur without being reported. Time frame is between 1979 to 1984. No specific provided."

TVA Evaluation

The utility felt that the concern was unsubstantiated for the following:

1. The individual identified by the concerned individual as having knowledge concerning this problem did not acknowledge seeing any craft personnel signing any QA documentation or know of any instances where it occurred.

2. The weld documentation system with all its crosschecks and reviews would have a high probability of not allowing the signoff of a QC holdpoint by an unqualified individual.

3. None of the people interviewed knew of any instance where a craftsman signed off on a QC holdpoint.

4. Since inspections were performed by the next available inspector, assurance of getting a particular inspector (personal friend) could not happen with any degree of certainty.

Expert Welding Team

The team believed that the investigation and closeout by utility was satisfactory.

3.3.12 Concern XX-85-065-001 was handled by NSRS Report I-85-750-SQN. This concern had as its major issues:

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

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

TVA Evaluation

The utility felt that this concern was unsubstantiated for the following reasons:

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

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

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

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

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

Expert Welding Team

The team agreed with the utility's findings but also observed that the number of reportable defects found on the reinspection showed an overall "sloppiness" in the original inspection sequence.

3.3.13 Concern XX-85-083-001 was answered with Report I-85-652-SQN. This had as its major issues:

1. Were Sequoyah welds properly inspected?

2. Were Watts Bar welds excessively inspected resulting in unjustified welding cost?

TVA Evaluation

The utility felt that the concern was unsubstantiated for the following reasons:

1a. The allegation that Sequoyah welds may not have been properly inspected could not be substantiated because these welds were inspected under an inspection of QC program which met the QA and Code requirements applicable to construction activities at Sequoyah.

1b. The allegation that Watts Bar welds were excessively inspected could not be substantiated because these welds were inspected under an inspection and QC program which met the ASME Code requirements applicable to Watts Bar. Since Watts Bar is an ASME Code stamped plant, the independent third party (ANI) verification of inspections performed by TVA personnel could be construed as a more strict inspection program. In addition, Watts Bar has been subjected to many reinspections to resolve possible safety concerns and to satisfy NRC inquiries. These, also, could be construed as a more strict inspection program.

2. A comparision of the overall welding inspection and documentation requirements between two nuclear plants of different ages, different codes of record, and code plant versus non-code plant cannot be described succinctly and, if done, differences will be observed. These differences would not necessarily indicate that one inspection program is better than the other or that the weld integrity of one plant is better than the other.

Expert Welding Team

The team believed that:

1. Programmatically the NSRS report does answer the question that the quantity of inspections between the two plants was similar. Previous reports, however, give rise to the speculation that the quality of these initial inspections may have left something to be desired at SQN.

2. WBN welding cannot be evaluated at this time. The team will evaluate this issue at a later date for the WBN program.

3.3.14 Concern XX-85-102-011 was evaluated by report I-85-735-SQN which had as its two specific issues:

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

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

TVA Evaluation

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

Expert Welding Team

The team agreed with the programmatic closeout of the concern by the utility.

3.3.1.5 Four concerns were addressed by WP-06-SQN, involving the following issues:

1. Prior to 1981, an inadequate Welding Inspection Training and Certification Program allowed welding inspectors to complete their training in two weeks.

2. The Training/Qualification Program for AWS welding inspectors is questionable becuuse the inspectors only have two months OJT which is not documented.

3. The Topical Report has been "bastardized" regarding TVA compliance with ANSI N45.2.6.

4. Welding inspectors are not qualified. They should be welders before becoming welding inspectors.

TVA Evaluation

The utility felt that the concerns were unsubstantiated for SQN (WP-06-SQN) for the following reasons:

1. A program was in place during the construction era which adequately addressed the applicable requirements for training, qualification, and certification of both visual welding and nondestructive testing personnel.

2. The Bechtel-SQN Implementation Audit established that this program adequately addressed the code and standard requirements of the construction era and confirmed that the program was effectively implemented for that era. The part of Issue 4 which states "they (welding inspectors) should be welders prior to becoming welding inpectors" should be dismissed. This is not an essential element of any training, qualification, and certification program. It is simply a statement of personal opinion.

Expert Welding Team

The team essentially agreed on the evaluation for SQN only. It should be noted that in a previous ERT report for these concerns that it was substantiated for the WBN units.

3.3.16 Concern XX-85-098-001 was addressed by WP-18-SQN. This concern has as its major issue:

1. Laminations in pipe prevented making a good butt weld in Unit 2 condenser.

TVA Evaluation

The utility wrote that the issue voiced in this concern is valid but not substantiated. It has been determined not to be detrimental for the following reasons:

1. ASME Class 1 rules state that weld prep laminations one inch and less in length are acceptable material conditions which do not require weld repair. Those greater than one inch are allowed to be weld repaired after grinding to a specified depth.

2. Condensers are constructed to requirements less stringent than ASME Class 1 which do not address laminations as injurious defects.

3. Laminations are commonly occurring discontinuities in wrought steel products and are not prohibited by materials specifications.

4. The effect of a lamination in a pipe subjected to internal pressure is of no concern.

5. Laminations pose no problem to weld joint integrity.

Expert Welding Team

The team agreed that the resolution was adequate and satisfactory, even though the item (not safety-related) did not necessarily fall into the scope of its review.

3.3.17 Concerns IN-85-271-001 and IN-85-282-002 were answered by WP-11-SQN and had as its major issue:

1. Grinding of welds may mask surface defects.

TVA Evaluation

1. The issue considered in this concern is not substantiated due to the fact that grinding is an acceptable practice.

Expert Welding Team

Satisfactory closeout of this concern.

3.4 Category 4 - Welding Design and Configuration

There were seven concerns that fall into this category.

Concern Number	Reference	Report Number Responding to Concern	Brief Description of Concern
EX-85-039-003	T1,T2	WP-15	Box hanger poor weld design
IN-85-613-001	T1,T2	WP-15	Thermal stress pipe/hanger weld
XX-85-086-002	T1,T2	WP-15	BNL wrong design for box hanger
XX-85-086-003	T1,F	WP-15	Weld design for box hanger
IN-85-405-001	T2	WP-15	Hanger over-designed
XX-85-068-007	T1,T2	I-85-636-SQN	No stamped spool falsified piece
XX-85-100-001	T1,T2	ERT XX-85-100-001	Improper weld repair

3.4.1 Five of the concerns were responded to by WP-15-SQN which had as its issues:

1. Box anchor drawings have a typical detail which shows a weld configuration which limits pipe movement.

2. There is a possibility of fatigue in service in process piping to box anchor connections due to lack of provisions for expansion.

3. There is a possibility of fatigue in service and material degradation due to continuous welding using large diameter electrodes and excessive amperage.

4. There is a possibility of thermal stresses degrading piping where large (half-inch to one-inch) fillet welds on box anchors attach to process piping.

TVA Evaluation

The utility considers the concerns unsubstantiated for the following reasons:

1. Engineering evaluations and tests relative to expansion and large welds have determined that their effect is not detrimental to process piping.

2. Continuous welding with large diameter electrodes is the optimum method of welding of box anchors.

Expert Welding Team

The team felt that more information was required on issues 2-4 as follows:

2. More information is required on possibility of "fatigue in service" for hangers. This was not addressed.

3. The answer to this issue is somewhat misleading/erroneous due to the fact that the use of larger diameter electrodes generally results in a greater heat input to the weldment.

4. This appears to be a Bellefonte specific issue. More information is needed to determine if the mockups had any bearing on SQN work.

The one installed box anchor at SQN (Issue #1) did not have this problem due to special handling, while the other seven hangers had the drawing changed. This issue appears closed.

3.4.2 Concern XX-85-068-007 was answered by NSRS Report I-85-636-SQN and had the following issues associated with it:

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

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

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

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

TVA Evaluation

The utility felt that the concern was not substantiated for the following reasons:

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

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

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

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

Expert Welding Team

It was felt that this report adequately addressed the concern and closed it out

3.4.3 The last concern in this category XX-85-100-001 was addressed by an ERT report. This concern had as its major issue:

1. An undetermined number of welds may have been repaired improperly.

TVA Evaluation

The utility felt the concern was not substantiated because insufficient evidence was found to substantiate the occurrence.

Expert Welding Team

Utility response was sufficient to close out concern.

3.5 Category 5 - Filler Metal Control

This category has 26 concerns associated with it, as listed on the following page.

Concern Number	Reference	Report Number Responding to Concern	Brief Description of Concern
EX-85-039-001	T1,T2,F	WP-01-SQN	No portable rod ovens
IN-85-424-001	T1,T2,F	WP-01-SQN	No portable rod ovens
IN-85-234-001	T1,T2,F	WP-01-SQN	No portable rod ovens
IN-85-426-001	T1,T2,F	WP-01-SQN	No portable rod ovens
IN-85-441-003	T1,T2,F	WP-01-SQN	No portable rod ovens
IN-85-352-002	T1,T2,F	WP-01-SQN	No portable rod ovens
WI-85-053-004	T1,T2,F	WP-01-SQN	Weld rod does not meet code
XX-85-068-006	T1,T2,F	WP-01-SQN	Weld rod control not code complying
IN-85-337-002	T2	WP-01-SQN	Weld rod control, exchange among welders
IN-85-424-004	T2	WP-01-SQN	Improper issue weld material
IN-85-424-007	T2	WP-01-SQN	Lack of weld rod control
IN-85-424-006	T2	WP-01-SQN	Weld material accountability
IN-85-454-004	T2	WP-01-SQN	Weld material accountability
IN-85-453-009	T2	WP-01-SQN	Weld material accountability
WI-85-041-001	T2	WP-01-SQN	Weld material accountability
IN-86-150-001	T2	WP-01-SQN	Weld material accountability
EX-85-021-001	T2	WP-01-SQN	Weld material accountability
IN-85-167-001	T2	WP-01-SQN	Weld material accountability
IN-85-672-001	T2	WP-01-SQN	Weld material accountability
IN-86-158-006	T2	WP-14	Weld material accountability
IN-85-411-002	T1,T2	WP-12	Bad ARCO weld rod
IN-85-247-001	T1,T2	WP-12	Poor 7018 electrode
IN-85-600-001	T1,T2	WP-12	Hobart poor 7018 electrode
IN-86-047-001	T1,T2	WP-14	No weld rod stub control
XX-85-013-001	T1,T2	ERT XX-85-013-001	309 used for 316 SS pipe
XX-85-041-001	T1,T2	NSRS I-85-756-SQN	Wrong weld rod CS to SS

3.5.1 Nineteen concerns were addressed by WP-01-SQN which had as its major issues:

1. The Weld Material Control Program does not meet code requirements. The context of this issue gives the inference that the concerned individuals are referring to the overall traceability of welding materials from procurement until the materials are consumed in the final weld.

2. Returned welding material is possibly not accounted for adequately.

3. Possible lack of portable electrode holding ovens.

4. Possible collection of moisture in electrodes due to lack of portable electrode ovens.

TVA Evaluation

The utility considers these concerns unsubstantiated for SQN for the following reasons:

1. Procedures were and are in effect for OC and NO which delineate the requirements for traceability and control of filler metals.

2. The provisions of these procedures reflect ASME B&PV Code rules and have been endorsed by ASME through tne ASME survey process.

3. The Bechtel SQN Implementation Audit established that these procedures were effectively implemented for both OC and NO.

4. The effect of noncompliance with these procedures was not found in the WP Sample Reinspection Program.

Expert Welding Team

The team had some comments/questions regarding the answer to the concerns:

1. The statements regarding traceability of material were adequate. It did appear that they had some progam conflicts, but this would not effect the hardware.

2. The issue regarding returned welding material was not addressed in WP-01-SQN.

3. For SQN (specific) this appears acceptable, but, were holding ovens issued at SQN between 1969-1974 (beginning construction phase)? Contradictory information received after the SQN site audit questions the completeness of this response.

4. This issue is only partially answered by the issuance of portable rod ovens. The accountability of returned rod (possibly left in a gang box or a glider over night) has a great influence on the rods moisture content.

3.5.2 Two concerns were addressed by WP-14-SQN. The issues addressed by these concerns were:

1. A system is needed which will provide the welder a receipt which can prove welding material was correctly returned to the rod issue centers.

2. TVA does not allow apprentices to weld.

TVA Evaluation

The utility felt that the concerns be closed since the issues have no quality or technical basis.

Expert Welding Team

The team felt that:

1. This issue was raised in WP-01-SQN as being "non-anwsered" in the report. This report also "non-answered" the concern.

2. This answer appears valid, but the team needs a copy of the concern for close out.

3.5.3 Three concerns were evaluated by WP-12-SQN which had the following issues considered:

1. E7018 electrodes are of poor quality.

2. Poor quality contributes to pinholes and porosity.

3. E7018 3/32-inch electrodes are of poor quality.

4. Electrode core wire is not centered and flux flakes off.

TVA Evaluation

The utility felt that these concerns were not substantiated because:

1. Electrode operability is a subjective judgement and cannot be measured quantitatively.

2. The specific cases discussed in the concerns were WBN occurrences.

3. A search of historic data on this subject by WP for SQN revealed no objective evidence of problems with these or other coated electrodes.

4. The reinspection effort did not reveal any evidence of electrode quality problems.

Expert Welding Team

The team believed that the utility did an adequate job in answering these concerns and had the following comments:

1. E7018 (if purchased to the correct codes and standards) is acceptable even though some welders may have a period of adjustment to different manufacturers' electrodes.

2. If the concentricity problem had occurred at SQN, it would have been impossible to locate all the electrode due to the electrode control procedures for the construction phase of SQN.

3.5.4 Concern XX-85-013-001 was addressed by a ERT report which had as its major issue:

1. 309 weld rod was used to weld 316 stainless pipe at Sequoyah Unit 1.

TVA Evaluation

The utility determined:

Based on the findings in this investigation, a change from E308 to E309 (same A&F designation) is not a violation of the code or procedure. The concern as stated may be true. However, the change from 308 to 309 filler metal has no impact to weld quality. This concern is closed.

Expert Welding Team

Utility response is acceptable

3.5.5 The last concern in this category XX-85-041-001 was answered by report I-85-756-SQN and had as its primary issue:

1. At Sequoyah, a weld was made in '79 or '80 in diesel generator building, unit 1, using the wrong type rod to weld carbon steel pipe to stainless steel pipe. A cover pass using the correct rod was run over the existing weld. Construction Dept. concern. CI has no more information.

TVA Evaluation

The utility felt that the concern was unsubstantiated because:

1. The concern of record could not be substantiated because all thewelds examined were found to be free of any defects which could be noted on the surface. All the welds were approximately the same physical size; therefore, not allowing the detection of any extra filler metal which might have been added to conceal a defective weld.

2. If the first pass weld was made with E308, the weld would not have been pleasing in appearance, but would have bonded to both the carbon steel and the stainless. The second pass with the correct electrode (E309) would have remelted some of the first pass and provided a smoother region of bonding.

3. With the rigid support being located adjacent to the weld, there is no reason to expect the weld would experience stresses to cause a fatigue failure. Also, if the instrument tube weld should develop a crack, it would be restrained from separating and creating a significant leak.

4. All the welds appeared to be sound and were free of any detectable defects after several years of operation.

Expert Welding Team

The team believed that the investigation into the problem was adequate.

3.6 Category 6 - Miscellaneous/One of a Kind

There were a total of 9 concerns in this category.

Concern Number	References	Report Number Responding to Concern	Brief Description of Concern
IN-85-192-002	T2	WP-08	Unpainted welds
IN-85-273-001	T1,T2	WP-08	Hanger welds not painted
IN-85-451-001	T2	WP-08	Weld not painted
EX-85-059-001	T2	WP-08	
WI-85-030-001	F,T1	WP-10	Welding + NDE corrective action not implemented
WI-85-030-010	T2	WP-10	Weld program study
IN-85-303-001	F,T1,T2	WP-13	No remote switches on welding machines
IN-85-247-002	F,T1,T2	WP-13	Only 2 setting on welding machines
XX-85-010-001	F		SQN-nut to baseplate welding plus chipped concrete

3.6.1 WP-08-SQN addressed four of the concerns which had as their major issues:

1. Welds over six feet of the floor have not been painted in the Reactor and Auxiliary Buildings.
2. Unpainted welds are in evidence on conduit and piping supports in the Reactor Building.
3. Hanger welds should be painted as soon as they are finalized by QC.
4. Rust causes welds to be weakened.
5. Sandblasting removes metal from welds.

TVA Evaluation

The utility resolved these issues as follows:

Issues 1, 2, 3, and 4 closed pending completion of protective coating reinspection and resultant corrective action under SQN-CAR-86-01-001.

Issue 5 is closed because the practice to sandblasting is an accepted practice in preparation of metals for painting.

Expert Welding Team

The team felt these were effectively closed out by the utility even though the concerns were not really welding issues.

3.6.2 Two concerns were addressed by WP-10-SQN which were considered with:

The corrective actions specified in Report Number QAE-80-2, "Review and Evaluation of the OEDC Welding and NDE Program," dated September 8, 1980, may not have been implemented.

TVA Evaluation

The utility felt this concern is substantiated and has closed the item because QAE-80-2 was not intended to be implemented at SQN and no impact on SQN hardware could be identified.

Expert Welding Team

Although not SQN-applicable, many of the recommendations in QAE-80-2 encompass "concern areas" at SQN and will be evaluated in greater detail for WBN.

3.6.3 Two concerns were addressed by WP-13-SQN which had the following issues associated with them:

1. Welding machines (grid packs) do not have suitable control settings for welding with 3/32-E7018 electrodes.

2. This unsuitability leads to porosity and pinholes in completed welds.

3. All GTAW equipment should have remote (high frequency arc starting) switches so that tungsten inclusions can be avoided.

TVA Evaluation

The utility closed out these issues based on the following:

1. There is no industry standard which mandates the use of specific welding equipment for specific jobs.

2. Equipment in use has sufficient control features to produce welds within the required criteria.

3. Alternate techniques can be used to compensate for the lack of sophisticated features on multiple operator-type equipment and still produce acceptable quality welds.

4. The WP reinspection did not discover any indications of a generic problem with welding equipment.

5. There is no effect on hardware due to these issues.

Expert Welding Team

The team determined that:

1. The welding machines described in the report appear to have sufficient settings for current ranges of typical welding procedures.

2. If the machine was unsuitable, porosity or pinholing could be a problem and should have been picked up on the reinspection program.

3. The high frequency arc start is a beneficial addition to GTAW welding, but is not essential in making a good quality welded connection.

3.6.4 The last concern, XX-85-010-001, is not a welding issue but should be a hanger installation concern.

4.0 PROGRAMMATIC REVIEWS BY TVA CONTRACTORS

In addition to the NSRS, QTC and WP reviews, TVA placed a contract with Aptech Engineering Services to perform a review of the SQN, PSI/ISI programs, as well as with Bechtel Engineering to perform an in-depth audit of their QA/welding program for both construction and operations.

4.1 Aptech Report

The review by Aptech Engineering Services was made in addition to the other TVA activities relative to the SQN welding concerns. The review covered welds subject to the ASME Section XI program for Class 1 and 2 piping, as well as Class 1, 2 and 3 component supports.

The review was performed using a two-pronged approach; first was the evaluation of the preservice and inservice inspection results, and second was the review of the operating experience of the two plants.

There were 1101 welds (both Class 1 and 2) examined during the SQN-PSI program out of a total number of 2618 Class 1 and 2 piping welds on the two units. Therefore, 42.1% of the total number of Class 1 and 2 piping welds were examined by the PSI program. Additionally, a total of 61 integrally welded attachments were subjected to the SQN-PSI program out of a total of 146 (41.8%). Less than 10% of the welds in the PSI program were examined by penetrant examination with the more than 90% of the welds examined volumetrically (ultrasonic testing).

The Aptech review of the SQN-PSI program uncovered only one significant NOI (Notice of Indications) reported. (Significant NOI refers to an indication which is unacceptable to ASME Section XI and requires repair and reinspection.)

The SQN-ISI program (to date of Aptech Review) had inspected a total of 456 piping/socket welds (predominantly repeats of PSI welds), 90 hanger integral attachments, and 100% of all Class 1, 2 and 3 hangers for SQN 1 and 2 (over 3,100 ISI inspections).

The total number of 27 NOIs were reported for Units 1 and 2. Of this total, 12 were significant NOIs (Unit 1) and 5 significant NOIs (Unit 2).

The operational history of the two plants showed that for 24,445 critical reactor hours of service (Unit 1), only 5 LERs were written relating to welds. (Licensing Event Reports (LERs) are written to USNRC to report failures on operating nucler plants). No failures were attributed to poor quality field welds.

Unit 2 has 21,985 critical reactor hours of service and has had no LERs relating to welds.

Based upon their review, Aptech Engineering came to the following conclusions:

- The welding program contains the necessary controls to ensure high quality welds (after the 1974 AEC audit).
- SNP evaluated the quality of welds made prior to the 1974 audit through reinspection and repair where required. Those welds made prior to the 1974 audit can now be considered to be satisfactory despite a breakdown in the QA program.
- The rate of significant indications detected during the preservice and inservice inspections is less than 5% with greater than 95% confidence.
- No Licensee Event Reports have been generated which relate to poor quality field welds.

BNL Evaluation

Although the Aptech findings give an optimistic prognosis for the plant, it must be remembered that this review only encompassed a paperwork review of the PSI and ISI programs and did not attempt to answer any employee concerns. The review did not require any physical reinspections of hardware and relied on documentation provided by TVA.

4.2 Bechtel Audit

A Bechtel audit team (five-member team) spent 30 auditor weeks reviewing records to determine the prior and current effectiveness of the TVA Welding Program (both construction and operations).

The scope of the audit included 17 key elements for review:

1. Implementation of technical and welding program requirements
2. Adequacy of design output documents

3. Initial welder or welding operator qualifications
4. Maintenance of welder or welding operator qualifications
5. Renewal of welder or welding operator qualifications
6. Initial welding inspection personnel qualifications
7. Maintenance of welding inspection personnel qualifications
8. Renewal of welding inspection personnel qualifications
9. Use of appropriate welding procedures
10. Use of appropriate inspection procedures
11. Use of appropriately trained and qualified personnel
12. Use and control of welding filler materials
13. In-process control of welding
14. Documentation of the above activities
15. Nonconformances and corrective actions
16. Training programs adequacy
17. Additional areas of concern as determined by a review of employee concerns.

The audit revealed one audit finding and four observations, none of which indicated a need for weld reinspections. The audit report also had no findings or observations relating to any employee concerns.

An observation of the auditors was that many TVA documents were confusing, overlapping, repetitive and unclear.

The Bechtel audit team also had this general observation:

> It is significant to the audit team that procedures were in place beginning in 1972 to provide the craft supervisors with quality assurance documents (procedures). The audit verified that by procedure, craft supervisors signed and returned a transmittal letter to indicate receipt of procedures and that an effective quality program was in place and complied with.
>
> From the nature of the concerns analyzed, it appears that there was a lack of understanding by the craft personnel of how the Sequoyah Quality Assurance System functioned, and this lack of understanding is the initiator of many of the employee concerns.

BNL Evaluation

This audit found no discrepancies which would indicate the need for weld reinspections. It must be remembered that this audit occurred after TVA has updated some of their records as part of the Employee Concern Program.

Additionally, the reinspection of a sample of welds at the Sequoyah Units to a less conservative inspection criteria (NCIG-01) did show a significant enough amount of rejectable structural attributes to assume that the original construction left something to be desired in meeting original code specifications.

5.0 HARDWARE INSPECTIONS (TVA AND NRC)

With the concurrence of the NRC staff, the utility had committed to do a reinspection on both Class 3 piping welds and structural members. Additionally, the NRC NDE van inspection of many welds during a February 18-28, 1986, visit to the Sequoyah Units, and the combined NRC and BNL Welding Team has done a "cradle to grave" audit at SQN Units 1 and 2.

5.1 TVA Reinspection

The TVA reinspection program sample consisted of 333 Class 3 piping welds in 7 systems (4604 linear inches) and 15 welds in spiral weld duct, as well as 403 joints for 50 structures, totaling 7,369 linear inches of structural welds. All of these welds were examined visually. In addition to the above, 304 piping welds were examined by either MT or PT (from the 333 pipe welds total).

5.1.1 Results of the Reinspection

Of the 304 piping welds receiving MT or PT, 296 were accepted on the first inspection (97.4%). All of the eight initially rejected welds were finally accepted as follows:

1. One accepted to ASME III after cleaning

2. Two accepted to ASME XI

3. Three welds accepted to ASME III after filing or grinding + Re-NDE

4. Two welds accepted to ASME XI after filing or grinding + Re-NDE

No weld repair by rewelding was required on any of the eight welds.

The 33 piping weld sample (304 of which had the MT or PT done) was visually inspected for 14 attributes. The inspection disclosed 184 rejectable welds out of the original 333 population (55.3%). The attributes inspected were:

Attribute	Rejected Welds
Contour/Transition	16
Offset/Alignment	2
Undercut	2
Reinforcement	7
Weld Spatter/Arc Strike	104
Weld Location	0
Weld Size	13
Weld Metal/Base Metal	0
Weld Convexity	0
Incomplete Fusion	5
Weld Overlap	8
Underfilled	12
Surface Porosity	15
Surface Slag	0
	184

All of the visually reinspected and rejected piping welds were eventually accepted to code requirments either by initial evaluation of engineering or after surface conditioning and reinspection. No cracks were reported on any inspections.

The fifteen spiral duct welds were all accepted on initial reinspeccion.

The structural welds were examined for 7 attributes and had 1,194 inches rejected out of 7,369 inches inspected (16.2%). These rejects break down as follows:

Size	765
Incomplete Fusion	18
Overlap	3
Craters	7
Profile	370
Undercut	31
Correct Filler Metal Type	0
	1,194

Additionally, nine weld joints were identified during the reinspection as having missing welds.

In all cases (rejected welds/missing welds), the evaluation by TVA Office of Engineering accepted the welds "as is."

As a result of the TVA Reinspection, the utility concluded:

> THE RESULTS OF THE REINSPECTION AND ENGINEERING EVALUATION OF THE REPORTABLE IMPERFECTIONS CONFIRM THAT THE REINSPECTED WELD JOINTS MEET DESIGN REQUIREMENTS, AND ADDITIONAL REINSPECTIONS ARE NOT REQUIRED.

5.2 NRC NDE Van Inspection

During February 18-28, 1986, the NRC NDE Van conducted an independent measurement inspection at the Sequoyah Units. Four USNRC representatives expanded 308 on-site and 24 off-site hours on the inspection. The scope of the inspection included:

27 pipe weldments - Dye penetrant inspection

13 pipe weldments - Magnetic particle inspected

361 structural weldments and 9 structures (several welds each) were visually examined to NCIG-01 with paint intact

46 piping weldments usually inspected to ANSI B31.7 with paint removed

The report stated that:

"There were several instances during this inspection where the NRC results differed from the licensee. In some instances, welds were rejected by the licensee but accepted by the NRC inspector; these differences were attributed to very conservative calls made by the licensee and to limitations present when inspecting welds which are painted. Conversely, some welds were accepted by the licensee but rejected by the NRC inspector. The inspector concluded, however, that the differences identified were not indicative of inadequate licensee programs and the NRC findings were representative of the types found by the lincensee."

5.3 Combined NRC and BNL Welding Team Audit

A combined NRC and BNL Welding Team audit was conducted at the Sequoyah site during June 2-6, June 16-20 and July 7-11, 1986. There were eight auditors performing various "cradle to grave" hardware and paperwork investigations at the site.

This audit had as its main objectives:

1. verification of the effectiveness of the TVA program to review, address, and close out NRC inspection program issues,
2. verification of the effectiveness of the TVA program to investigate and close out employee concerns, and
3. confirmation that the reinspection program carried out by TVA was performed in accordance with their commitments.

This audit encompassed 30 pipe welds, 502 pipe support welds, 31 instrument tubing welds, 120 instrument support welds, 130 structural welds (electrical), 280 HVAC support welds, 100 structural welds and associated paperwork.

There were some irregularities found during the audit with most of the hardware discrepancies having been previously identified as a result of the TVA reinspection effort. The report concludes that, in general, the inspected welds were found to comply with the governing codes and specifications.

6.0 DISCUSSIONS AND CONCLUSIONS

During the review of the welding concern issues for Sequoyah Units 1 and 2, many items of conflicting evidence came to light. There were many disparities between the WP, NSRS, QTC reports and the results of the Bechtel report and Aptech survey. These disparities appeared to be primarily programmatic in content and could be either isolated instances of program transgressions or problems endemic to the entire welding program at the Sequoyah Units.

In order to determine the TVA welding program effectiveness, it is necessary to review the six categories of concerns from both a programmatic approach and then a hardware-oriented approach.

6.1 Category 1 - Welding Procedures

a. Programmatically - There were no concerns specific to this category and there is verification that procedures governing this work were in effect during construction and operations.

b. Hardware - The materials of construction for SQN 1 and 2 were not unique and had been welded by the normal methods of metal joining (SMAW, GTAW, GMAW, etc.). That these procedures were adequate to produce sound welds is evident from the operating history of the two units.

6.2 Category 2 - Welder Qualification/Testing

Programmatically, this category had a significant number of concerns associated with it. The methods of updating a welder's certifications was questioned and was the largest single area for all the concerns voiced. There is no doubt that there were instances where procedural violations probably occurred (as in the case of the eight welders who had to be retested), but what is the potential hardware effect?

If the problem of uncertified/unqualified welders welding on critical systems of the Sequoyah units were all pervasive, then the reinspections done by both the utility and the USNRC would have had a relatively high reject rate. If one reanalyzes the visual rejects on piping welds (TVA reinspection) 184/333 welds, it can be seen that if "welder attributable defects" are extracted, e.g., undercut, incomplete fusion, weld overlaps, underfill and surface porosity, the reject rate for poor welding becomes 42 out of 333 welds (13%). This number may be considered high, but if one takes into account that these "rejects" were all able to be accepted by engineering without rewelding, the amount of significance one can place in the 13% value becomes insignificant.

Note: Arc strikes and spatter have not been included since they may have been caused by adjacent welding operations (above or below the area of interest).

The reject rate of "welder attributable defects" or the structural welds, e.g., incomplete fusion, overlaps, craters, undercut becomes 59 inches out of 7369, or 0.8%; not excessively high at all.

Additionally, these reinspected welds (Class 3 piping and structural welds) are the safety-related welds least inspected on the nuclear plant and would exhibit the most defects if an "all pervasive" welder qualification problem were in existence.

The apparent good quality of the welds covered under ASME Section XI determined in the Aptech report also lends credence to the supposition that these concerns on welder certification are most probably isolated occurrences at SQN.

6.3 Category 3 - Welding Inspection

Many of the same arguments used in the previous category can apply to the inspection of welds and the qualification of personnel performing same. The Bechtel audit verified that inspection procedures and training procedures were in effect at SQN, which programmatically should satisfy the welding program requirements.

It can be successfully argued that there may have been an overall sloppiness in initial inspections done by TVA personnel, especially when one looks at the amount of rejectable welds for size and profile on both the piping and structural welds. Both of these items are inspection/inspector intensive.

This overall "sloppiness" in inspection (during construction) was emphasized by the TVA reinspection. This reinspection was performed using the less conservative standard NG1G-01 in lieu of the original construction standard AWS D1.1. The initially high reject rates recorded on this reinspection are a clear indication that TVA had not performed their original inspections to the original acceptance code (AWS D1.1).

Even though there was a reasonably large number of rejects, none were significant enough to warrant repair by welding and were all accepted by engineering. One must also take into account that on a well publicized reinspection, many "defects" may come to light that would normally be considered nonrelevant during a regular inspection.

6.4 Category 4 - Welding Design and Configuration

This category contained only seven concerns, five of which related to the same box hanger design. Although some information is still outstanding from TVA on the design issue, none of the requested information would indicate an extensive problem. The other two concerns from this category were found to be unsubstantiated by the utility and at best would only indicate a limited instance of programmatic breakdown.

6.5 Category 5 - Filler Metal Control

The question of no portable rod ovens at SQN was the single largest item of concern. Return of weld rod at the end of the shift would also be allied with this concern. The Bechtel audit verified that procedures were in effect to control the issuance and use of filler material, so programmatically a system was in effect.

Before one can analyze the extent of the problem, the question must be asked; "Why do we want to use portable ovens in the field, and what is the potential effect if we don't?"

The primary reason for use of weld rod ovens is to prevent moisture pickup on the weld rod, which could cause hydrogen delayed cracking. Notoriously, this type of cracking will make itself known visually from few minutes to a few days after the weld is made. The results of the reinspection and PSI/ISI programs showed no evidence of cracked welds being found, so this is probably not a problem at SQN.

6.6 Category 6 - Miscellaneous/One of a Kind

Of the nine concerns in this category, only four were directly related to welding. Two of these dealt with control adjustments on welding machines which, if substantiated, would have caused defects that would have been observed on the reinspection program. They were not. The other two concerns were of a programmatic and not a hardware-specific nature.

A response (Attachment 10) to the first set of questions sent to TVA from the NRC was received August 1, 1986. The responses from the utility confirm the fact that there were programs, procedures or inspection plans in effect which outlined the necessary steps to provide a "sound weld" as an end product to construction. This does not mean that programmatic transgressions did not occur, but that a system was in effect to localize these transgressions and prevent system-wide quality problems.

The supposition that the SQN units did not suffer from "all pervasive" quality/welding problems is substantiated by the utility's reinspection program which revealed:

- No piping welds (Attachment 10) rejected by Code

- No structural weld joints which did not meet design requirements

Since there has been presented evidence that some programmatic breakdowns probably occurred, it appears that an evaluation of the units' "suitability for service" must rely essentially on the large number of hardware reinspections that have been performed to date. This being the case, the following conclusions can be drawn:

1. There is evidence that many of the confusing/misleading TVA procedures may have led to programmatic errors in the SQN welding program. The expert welding team questions on these have been transmitted to the NRC for forwarding to the utility.

2. The reinspection of various welds by the utility and NDE Van have not discovered defects of any great detriment or magnitude.

3. Since the combined NRC and BNL Welding Team audit did not show additional problem areas or concerns; if and the questions addressed to TVA are answered to the satisfaction of the USNRC, then there is no evidence to assume that the welds at the Sequoyah Units are not "suitable for service."

These conclusions are those of the BNL members of the welding team. These conclusions have been drawn after discussions with the other members of the team, TVA meetings and site inspections and audits. Each of the other members of the team have been requested to submit their own summaries of their opinions regarding the TVA Employee Concern Program. These are part of this TER (Attachments 11-15).

REFERENCES

1. Tennessee Valley Authority, "Welding Project Review Plan," Volume ..

2. Proceedings, "NSRS Public Meeting on TVA Welding Issues," January 7, 1986, Bethesda, MD.

3. Memorandum, R.D. Liaw to B. J. Youngblood, dated March 3, 1986.

4. Visual Weld Acceptable Criteria for Structural Welding at Nuclear Power Plants, NCIG-01, Rev.2, 5/7/85.

5. Tennessee Valley Authority Welding Project Sequoyah Nuclear Plant Phase 1 and 2 Review and Program Results.

6. Memorandum, H. Denton, J. Taylor and J. Nelson Grace to W. J. Dircks dated 12/5/85. Subject: NRC Plan for Dealing with Welding Issues at TVA Watts Bar Site.

LIST OF ATTACHMENTS

Attachment 1

PROFESSIONAL QUALIFICATIONS
of
CARL J. CZAJKOWSKI

I am currently a Research Engineer at Brookhaven National Laboratory (BNL), where I have been employed since 1980. I am in the Materials Technology Division of the Department of Nuclear Energy. My current duties are providing technical assistance (both field and laboratory) to the United States Nuclear Regulatory Commission (USNRC) in the areas of metallurgy and failure analysis related to nuclear power plants. Failure analyses performed on both radioactive and non-radioactive components in my current position have included the following material systems: austenitic stainless steels, ferritic and martensitic low alloy steels, inconel, aluminum and martensitic stainless steel. I have performed vendor audits for the Inspection and Enforcement Division of the NRC in the capacity of Technical Specialist in the aforementioned areas of expertise. I have performed a third party investigation of allegations pertaining to potential welding and quality control improprieties at a nuclear construction site. Additionally, I have testified as the NRC Technical Specialist for welding at hearings for a second nuclear plant.

Prior to my employment at BNL, I was employed for five years by the Long Island Lighting Company (LILCO). My job title from September 1977 to February 1980 was Chief Welding Supervisor at the Shoreham Nuclear Power Station. My duties in this position included supervisory responsibility for all welding problems or major welding efforts for the utility, as well as ordering and maintenance of equipment/gases/electrodes to support a 400 welder work force at the site. Additional responsibilities included conducting training sessions for supervisory and manual personnel on industry codes, standards and welding inspection, as well as administering the weld test booth for qualification testing. Subsequent to my promotion to Chief Welding Supervisor, I was employed by LILCO as a Quality Assurance Engineer (both home office and Shoreham site). This position was held by me from February 1975 to September 1977. The duties of this position encompassed preparation and review of LILCO's QA manual and procedures, reviewing A/E and NSSS quality programs, evaluating and surveying vendor activities, and performing field audits and surveillance of mechanical contractors' (Shoreham site) welding and nondestructive testing practices.

I also held the job title Quality Assurance Engineer while employed by Ebasco Services, Inc. from September 1973 to February 1975. This position's duties included review of procurement specifications and drawings for inclusion of quality requirements, preparation of quality plans for surveillance of safety-related component fabrication in vendors' shops, conducting interdepartmental audits of engineering and design disciplines, in addition to QA evaluation of vendors, including review of documented quality programs and source evaluation.

Professional Qualifications of
Carl J. Crajkowski

Page Two

Prior to my employment at Ebasco Services, I held the job title QC Materials Engineer for United Nuclear Corporation from April 1972 to August 1973. This position's responsibilities included review of material purchase orders for compliance with contract requirements monitoring of the test-overcheck program for ferrous and non-ferrous material, establishing materials receiving inspections instructions and audit participation, as needed.

My academic qualifications include a B.S. in Metallurgical Engineering from the University of Missouri at Rolla in 1971, and an M.S. in Metallurgical Engineering from the Polytechnic Institute of New York in 1982. I am a member of the American Society for Metals and the American Welding Society. I am the author or co-author of approximately fifteen publications in the area of failure analysis on reactor components.

BIOGRAPHICAL SKETCH
of
WILLIAM D. DOTY

W. D. Doty received his B. Met. E., M. Met. E., and Ph.D. (Metallurgy) degrees from Rensselaer Polytechnic Institute, where he also served as a Research Fellow. His graduate research was recognized through a national award from the American Welding Society. Dr. Doty joined the United States Steel Corporation in 1947, and served in various research and supervisory positions at their Technical Center; from 1958 to 1966 as Chief of the Bar, Plate and Forged Products Division; from 1966 to 1973 as Research Consultant, Steel Products Development; from 1973 to 1983 as Senior Research Consultant, Product Engineering; and from 1983 to 1985 as Senior Metallurgical and Product Consultant.

Dr. Doty is widely known for his research and publications in welding and steel product development, and is co-author of an authoritative book on the "Weldability of Steels." In 1966, he received the Sparagen Award of the American Welding Society; in 1972 he was elected a Fellow by the American Society for Metals; in 1975 he was elected an Honorary Member by the American Welding Society; and in 1984 he was elected a Fellow by the American Society of Mechanical Engineers and was that Society's recipient of the J. Hall Taylor Medal. His technical committee activities have been many and varied. He is a member of the Main Committee of the ASME Boiler and Pressure Vessel Committee, and from 1967 to 1973 he served as Chairman of the Pressure Vessel Research Committee, and is currently a member of the PVRC Executive Committee. He is a member of ASM, AWS, British Welding Institute, AIME, ASME and Sigma Xi, and is a Registered Professional Engineer in the State of Pennsylvania.

BIOGRAPHICAL SKETCH
of
CARL E. HARTBOWER

Mr. Carl E. Hartbower, retired from the Federal Service, will be available as a consultant starting in June 1982. Having served the Bridge Division of the Federal Highway Administration as their Principal Welding Engineer for almost nine years, he has been in a unique position to observe the welding-related problems that exist in the Interstate Bridge System.

Mr. Hartbower is a Registered Professional Engineer in Massachusetts (mechanical engineering) and in California (metallurgical engineering).

- Fellow of the American Society for Metals (1979) "in recognition of contributions to the metallurgy and engineering of large-scale welded structures, to the use of fracture mechanics in modern bridge design and fabrication, and to the advance of nondestructive testing techniques and inspection."

- FHWA Administrators Award in 1979 "in recognition of his outstanding contribution in fostering the Fracture Control Plan and for leadership in the safety problems attendant to the fabrication of major bridges."

- Fellow of the Welding Institute (British).

- Life member of the American Welding Society.

- Member of the American Society for Testing and Materials; Chairman of Task Group E24.03.03 (precrack Charpy test methods).

- Pioneering research (1) in development of the Navy's explosion bulge test (1940s), (2) on the welding of titanium alloys (1950s), (3) in development of the precrack Charpy test (1950s), and (4) on acoustic emission (1960s).

- Navy Civil Engineer Corps Awards (1951 and 1952) for research papers on the explosion bulge test.

- Spraragen Award for the best research paper published by the American Welding Society in 1968 (paper on acoustic emission); Mr. Hartbower's research on acoustic emission published by NATO in AGARDograph 176 (January 1974) and in AGARDograph 201 (Ocober 1975).

- Exchange Scientist - in April 1961 the U.S. was visited by Professor N. N. Rykalin of the Baykov Institute, Academician B. Ye. Paton of the Paton Institute, and Professor N. O. Okerblom of the Leningrad Polytechnic Institute. In exchange, the National Academy of Sciences selected Mr. Hartbower as one of three welding authorities to visit industrial and educational centers in the Soviet Union (see WELDING ENGINEER, p64 July 1962 and WELDING RESEARCH ABROAD, Vol. VIII, No. 2, Feb. 1962).

Biographical Sketch of
Carl E. Hartbower

Page Two

- In 1971, Mr. Hartbower revisited the USSR by personal invitation from the Soviety Academy of Sciences.

- Commander, Navy Civil Engineer Corps, USNR, RETIRED. Life Member of the Naval Reserve Association and Member of the Naval Institute.

- Elder in the Presbyterian Church.

Mr. Hartbower has BS (194_) and ME (1958) degrees from Worcester Polytechnic Institute; was previously employed by the Naval Gun Factory (Welding Engineer, 1943-45), Naval Research Laboratory (Physical Metallurgist, 1945-52), Watertown Arsenal Laboratories (Chief of Metals Joining Branch (1952-61), and Aerojet General Corp. (Associate Scientist (1962-73). He has published over 50 papers.

BIOGRAPHICAL SKETCH
of
PAUL E. MASTERS

Paul E. Masters received his BS in Engineering from Iowa State College and did advanced work in metallurgy at Carnegie Institute of Technology and the University of Pittsburgh.

Prior to his association with the American Bridge Division of United States Steel Corporation in 1942, Mr. Masters was in the Engineering Department of Yates-American Machine Company, Beloit, Wisconsin. In 1943 he joined the Welding Engineering Department of American Bridge, assuming the position of Chief Welding Engineer in 1956 and retiring in 1977.

As American Bridge's Chief Welding Engineer, he was responsible for engineering in connection with development of work practices and applications of processes for welding, oxygen cutting, and allied subjects throughout the Division. This included application, development of procedures, training and certification of all nondestructive testing methods as required by the various codes for fabrication of structural steel and pressure vessels.

Mr. Masters is an Honorary Member of the American Welding Society, a Fellow of the American Society for Metals, and is a Registered Professional Engineer in the State of Pennsylvania.

He has been on the Board of Directors and Executive and Finance Committee of the American Welding Society, active in AWS National Technical Committees and Task Groups, having been Chairman of the Structural Welding Committee, the Committee on Qualification and Certification of Welding Personnel, the Committee on Qualification, and the Subcommittee on Submerged Arc Filler Metals. He has been a member of committees on filler metal, process requirements, handbook, mechanical testing of welds, terms and definitions, and a member of the ASNT Select Committee on Certification. Mr. Masters received the 1978 Samuel Wylie Memorial Metal Award for contributing conspicuously to the advancement of welding and cutting.

Mr. Masters was a qualified Welding Inspector, certified by the American Welding Society, and was certified by the American Society for Nondestructive Testing as NDT Level III in radiography, magnetic particle and penetrant testing methods.

Since retiring from American Bridge Division, Mr. Masters has done welding consulting work in the United States and abroad. This includes practical instructing, education and the technical aspects of welding.

BIOGRAPHICAL SKETCH
of
WILLIAM H. MUNSE

W. H. Munse is a Professor Emeritus of Civil Engineering at the University of Illinois, Urbana, Illinois, where he received a B.S. degree in Civil Engineering in 1942. While an undergraduate student, he served as an Engineer for the city of Champaign, Illinois, and as a student assistant in Civil Engineering. After spending nine months as a structural draftsman at the American Bridge Company, he returned to the University of Illinois in 1943 as an instructor and research assistant and received an M.S. degree in Civil Engineering the following year. Upon completion of a tour of duty as an officer in the U.S. Navy, he served as a Research Engineer at Lehigh University for one year and then returned to the University of Illinois, where he has been on the professional staff since 1947.

Professor Munse's area of specialization has been the basic engineering behavior of metals and metal structures. He has made numerous contributions through his research on the static, fatigue, and brittle behavior of riveted, bolted and welded construction, and in the engineering application of the results of this research into the classroom, and in the translation of the research results into materials and designs specifications. This latter achievement has been made possible through his membership on the design specification committees of the American Institute of Steel Construction, the American Welding Society, the Research Council on Riveted and Bolted Structural Joints, and the American Railway Engineering Asociation, and on materials committees of the American Society for Testing and Materials. In addition, he has contributed to many national and international committees concerned with the behavior of metals and metal structures, serving as U.S.A. delegate to the fatigue commission of the International Institute of Welding and the fatigue and fracture commission of the International Ship Structure Committee.

The results of Professor Munse's research have been presented in many national and international journals and reports. He is author or co-author of more than 140 publications, author of the Welding Research Council book on Fatigue of Welded Steel Structures, and author of chapters in several other books and handbooks.

In addition, Professor Munse has served as a consultant or advisor to many industrial and governmental agencies on problems involving the properties and behavior of metal structures. Included have been the development of design specifications for the Corps of Engineers, the evaluation of the fatigue resistance and the development of fatigue design provisions for such organizations as the Chicago Bridge and Iron Company, the U.S. Steel Company, the ACF Company, the Association of American Railroads, and the U.S. Navy; and assisted in the evaluation of failures of bridges, railroad cars, buildings, a water tank, and an off-shore drilling rig. He has served also in an advisory consulting capacity to a number of other industrial and governmental organizations on structural and materials problems.

Professor Munse is an Honorary Member of the American Society of Civil Engineers, as well as a member of a number of other technical and professional engineering societies, including ASTM, AREA, AWS and TRB. In addition, he has served on many of the technical and professional committees of these organizations. Has has been awarded the ASCE's Walter L. Huber Civil Engineering Research Prize and the Adam's Memorial Membership of the American Welding Society in recognition of his research on the behavior of metals and metal structures. In 1976 he was recognized by the Japan Welding Society on their 50th Anniversary with their Distinguished Service Award.

In May 1983 the Structural Division of the American Society of Civil Engineers honored Professor Munse with a symposium on the "Behavior of Metal Structures" and, in October 1984, he was elevated to Honorary Membership by the Society.

BIOGRAPHICAL SKETCH
of
ROBERT D. STOUT

Dr. Robert D. Stout is internationally known for his work in welding and has won several awards for his accomplishments in this field. He has been on the Department of Metallurgy and Materials Engineering faculty at Lehigh University since 1939. He was Chairman of the Department from 1956 until 1960, when he was named Dean of the Graduate School, and served until his retirement in 1980.

A native of Reading, Pennsylvania, he came to Lehigh from Carpenter Steel Co., where he was a metallurgist.

In recognition of his contributions to education and engineering, Dr. Stout has received several national honors. In 1945, he was awarded the Lincoln Gold Medal for "conspicuous advancement of the science of welding". In 1952, he was presented the Stoughton Award by the American Society for Metals (ASM) for "outstanding contributions to the teaching of metallurgical engineering", and in 1972 he received the ASM's A. E. White Award for distinguished teaching.

In 1960, he was selected to deliver the Adams Lecture to the American Welding Society (AWS). He received the Spraragen Award for the best research paper published in the Welding Journal in 1963; the National Meritorious Certificate of the AWS in 1965; the R. D. Thomas Award in 1973 for service to international cooperation in welding; and the Charles H. Jennings Award for an outstanding research paper in 1974.

In 1962, Dr. Stout received the R. R. and E. C. Hillman Award at Lehigh, presented annually to "the member of the Lehigh faculty who has done the most toward advancing the interests of the University".

The David Ford MacFarland Award, presented annually "in recognition of achievements in the field of metallurgy which reflect credit upon Alma Mater," was conferred upon him in 1959 by the Pennsylvania State University chapter of the ASM.

Since 1955, Dr. Stout has served as one of the official American representatives to the International Institute of Welding, participating in the commission to study the behavior of metals subjected to welding. He has attended international meetings of the Institute and delivered the 1970 Houdremont Lecture to that organization. He was a member of the materials advisory board of the National Academy of Sciences from 1964 to 1968, and a member of the Pipeline Safety Standards Advisory Commitee from 1968-71.

A graduate of Pennsylvania State University in 1935, he received his M.S. in 1941 and the Ph.D. in 1944, both from Lehigh. In 1967, he received the honorary degree, Doctor of Science, from Albright College.

He has authored more than 125 articles which have appeared in leading technical journals, and is author of the book "Weldability of Steels" published in 1953 and revised in 1971. He has served as a metallurgical consultant to over 50 companies.

Dr. Stout was National President of the AWS in 1972-73. He has served as chairman of several committees of that organization. He also has been Chairman of the University Research Committee of the Welding Research Council. He is a member of the Society of Sigma Xi, national research honorary society, and Tau Beta Pi, national engineering honorary society, and is a Fellow of the ASM.

RESUME

NAME: Milford H. Schuster TITLE: Research Engineer

FIELD OF EXPERTISE: Welding, metallurgy, nuclear power plant construction, nondestructive testing, failure analysis

EDUCATION: Specialized training in welding, welding metallurgy physical metallurgy, reactor materials, ASME Sections I, II, III, V, VIII, IX and XI, electron beam welding, personnel management, engineering assurance, IHSI, G.E. BWR Training Program, ASNT-TC-1A, Level II MT, PT, UT, radiography, ANSI B31.1, pipe welding, quality control, quality assurance, machining.

EXPERIENCE:

1986-Pres.	Brookhaven National Laboratory (Materials Technology Division), Technical Consultant to USNRC-NRR and I&E Divisions, Failure Analysis.
1980-1986	Long Island Lighting Co. - Welding/Materials Specialist Consultant, Chief Welding Supervisor, Shoreham Nuclear Power Station.
1979-1980	EBASCO Services Corporation - Welding Specialist, Commission DE Federal, DE Elecricidad, Laguna Verde Nuclear Power Station, Mexico.
1978-1979	Daniel International Corporation - Project Welding Superintendent, Enrico Fermi Nuclear Power Station.
1976-1978	Courter and Company - Piping Supervisor, Welding Supervisor, Shoreham Nuclear Power Station.
1971-1976	Self employed - Welding consultant, Brookhaven National Laboratory; Partner auto parts business.
1956-1971	Brookhaven National Laboratory - Materials and Welding Technical Specialist.
1952-1956	USAF - Aircraft Fabricators Inc., Metals Processing Specialist, Welding instructor.

Attachment b

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

Upton, Long Island, New York 11973

Department of Nuclear Energy

(516) 282-
FTS 666 4420

March 18, 1986

Dr. B.D. Liaw
Eng. Branch
Mail Stop P-1132
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Liaw:

This letter is being sent to document the visit of Messrs. P. Masters, W.H. Munse, R.D. Stout and myself to the Sequoyah Nuclear Units on February 26-28, 1986. This site visit was made in conjunction with our duties as Expert Welding Team members on FIN A-3839 entitled "Evaluation of Welding Concerns at TVA Operating Reactors."

The trip report is broken up into three parts; each part will cover one days activity of the visit.

February 26, 1986

A meeting was held at the Sheraton East Ridge, Chatanooga TN, at 15:00 hours on February 26, 1986. The meeting was attended by both USNRC personnel and welding consultants (Attachment #1). The purpose of the meeting was to discuss the TVA Welding Concern Program and establish the Welding Team Charter. Informal presentations were made by the following:

D. Smith/W. Long

1. Discussed an overview of the TVA welding concern program and established the charter of the Expert Welding Team.

2. The charter of the Team is not to address any vendor welds 't will only address site related problems.

3. There are approximately 60 Sequoyah concerns on welding out 500 total welding concerns.

4. Definitions of various acronyms were discussed, e.g., NSRS, QCT, LER, etc.

A. Berdt

1. Provided a discussion of TVA history of the plants including:

   a. potential "generic" concerns

   b. the fact that TVA does its own construction and design on its units

   c. a definition of "employee concern"

2. Discussed the public meeting between USNRC and TVA of January 7, 1986.

3. Described the TVA and USNRC telecon discussions of January 9, 1986, and TVA's commitment for a physical reinspection of their plants.

4. Detailed the January 29, 1986, inspection report of Messrs. Crowley, Smith and Cortland.

   a. Discussed report's Executive Summary

5. Described the work the USNRC NDE Van was performing at Sequoyah Units 1 and 2.

C. Czajkowski

After NRC personnel left the meeting, I addressed the Team and detailed the specifics of the contract, the type of reports required, and most importantly, the independence which must be maintained by the Team of Experts.

February 27, 1986

A meeting was held at 10:00 hours at the Sequoyah Nuclear Plant site between TVA personnel, USNRC personnel and the Welding Team (Attachment #2). A presentation was made by TVA personnel (Attachments 3-6) of the planned activities of TVA in evaluating the employee concerns expressed about the Sequoyah site.

After the presentation, the Welding Team and NRC personnel were escorted on a tour of both units 1 and 2 (and common areas to both) which included many of the welds which were inspected by either TVA or the USNRC NDE Van or both. Various structural welds and pipe welds were visually examined by the Team with no abnormalities noted.

After the tour, the Welding Team discussed their observations with USNRC NDE Van personnel.

February 28, 1986

The Exit Critique of findings by NRC NDE Van personnel was convened at 10:41 hours (Attachment 7). The report of findings were given the number 327/328 86-13. The Van personnel had reviewed documentation for fifty pipe welds as well as seventeen structural weld packages. NRC Van personnel had inspected approximately:

350 structural welds (already inspected by TVA)
190 new structural welds (not previously inspected)

35 pipe welds (already inspected by TVA)
12 new pipe welds (not previously inspected)

There were some follow-up NRC items from the inspection and some areas which were still under evaluation by NRC personnel, but one point was made on the reinspection:

> The differences observed on the quality of welds inspected were not considered unusual or worse than industry norms.

I would like to note that both the TVA and NRC personnel were most forthright in providing answers to the Team in a timely and professional manner. If there are any questions, please contact me.

Very truly yours,

Carl J. Czajkowski

Carl J. Czajkowski

Attachments
CJC/ad
cc: Welding Team
P. Cortland
W. Kato
W. Long
D. Smith
J. Weeks

Meeting at the Sheraton East-Ridge, Chattanooga, TN
Regarding Welding Concerns
2/26/86 - 1500 Hours

Attendees

NAME	AFFILIATION
Carl Czajkowski	Brookhaven National Laboratory
David E. Smith	NRC/PWR-A/EB
Alan R. Herdt	NRC - Chief Engineering Branch, Region II
Bill Long	NRC/NRR
Paul Cortland	NRC/OIE
Paul Masters	Consultant - Brookhaven National Laboratory
William H. Munse	Consultant - Brookhaven National Laboratory
Robert D. Stout	Consultant - Brookhaven National Laboratory

(3/18/86 Memo Attachment 2)

Attendance List - February 27, 1986

NAME	AFFILIATION
J.E. Rose	
John Fox	TVA
L.E. Martin	TVA-ONP
Alan R. Herdt	NRC - Chief Engineering Branch, Region II
Bill Long	NRC/NRR Project Manager for Welding
Carl Czajkowski	Brookhaven National Laboratory
David E. Smith	NRC/PWR-A/EB
George Georgiev	NRC/IE
Robert D. Stout	Consultant - Brookhaven National Laboratory
William H. Munse	Consultant - Brookhaven National Laboratory
Paul Masters	Consultant - Brookhaven National Laboratory
John D. White	TVA Engineering
D.J. Etzler	TVA Office of Engineering
C.W. Hatmaker	TVA Office of Engineering
Gary J. Pitzl	TVA Office of Nuclear Power
Robert A. Montgomery	TVA Office of Engineering
Larry D. Alexander	TVA SNP
J.W. Coan	TVA WP
Robert M. Jessee	TVA WP

(3/18/86 Memo Attachment 3)

WELDING PROJECT CHARTER

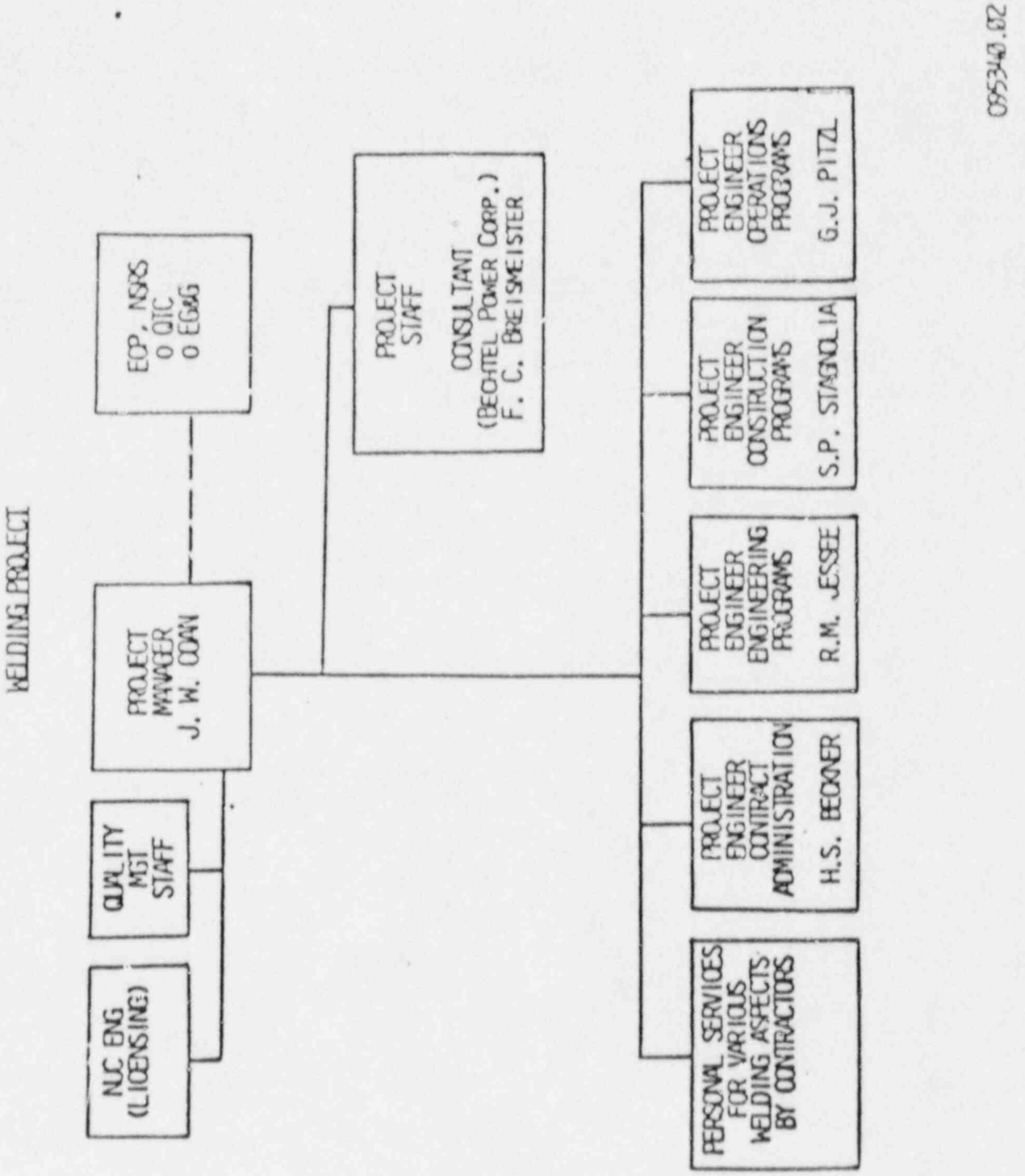
EXAMINE THE ORGANIZATIONAL WELDING PROGRAMS IN TVA, DETERMINE ANY REMEDIAL ACTIONS THAT MAY BE NEEDED, AND TAKE THOSE ACTIONS NECESSARY TO ASSURE THAT FUTURE TVA PERFORMED WELDING ACTIVITIES ARE IN ACCORD WITH TVA'S COMMITMENT TO EXCELLENCE IN ITS NUCLEAR PROGRAM.

VERIFY THAT THE TVA PERFORMED WELDING OF STRUCTURES, PIPING SYSTEMS, AND OTHER SAFETY-RELATED PLANT COMPONENTS, WHICH ARE CURRENTLY IN PLACE AT TVA'S NUCLEAR PLANTS ARE ADEQUATE TO MEET TVA, CODE, AND REGULATORY REQUIREMENTS.

THE PRIORITY WILL BE AS FOLLOWS:

1. SEQUOYAH
2. WATTS BAR
3. BROWNS FERRY
4. BELLEFONTE

095340.06


WELDING PROJECT
NUC ENG (LICENSING)
QUALITY MGT STAFF
PROJECT MANAGER J. W. COAN
ECP, NSRS
O QTC
O EG&G
PROJECT STAFF
CONSULTANT (BECHTEL POWER CORP.) F. C. BREISMEISTER
PERSONAL SERVICES FOR VARIOUS WELDING ASPECTS BY CONTRACTORS
PROJECT ENGINEER CONTRACT ADMINISTRATION H.S. BECKNER
PROJECT ENGINEER ENGINEERING PROGRAMS R.M. JESSEE
PROJECT ENGINEER CONSTRUCTION PROGRAMS S.P. STAGNOLIA
PROJECT ENGINEER OPERATIONS PROGRAMS G.J. PITZL
095340.02

PURPOSE

PHASE I

THE PRIMARY PURPOSES OF PHASE I ARE TO ENSURE THAT THE TVA PROGRAM, DESIGN DOCUMENTS, POLICIES AND PROCEDURES CORRECTLY REFLECT TVA COMMITMENTS AND REGULATORY REQUIREMENTS AND TO IDENTIFY AND CATEGORIZE CONCERNS/ DEFICIENCIES IN THE WELDING PROGRAM.

PHASE II

THE PRIMARY PURPOSES OF PHASE II ARE TO:

- EVALUATE THE IMPLEMENTATION OF PROCEDURES
- VERIFY THAT INSTALLED WELDMENTS MEET REQUIREMENTS OR ARE ADEQUATE FOR SERVICE
- CORRECT ANY PROBLEMS, IMPLEMENT CHANGES TO PREVENT RECURRENCE

096002.02

ACTION PLAN

PHASE I

1. REVIEW TVA COMMITMENTS TO NRC
2. VERIFY THAT WRITTEN PROGRAM REFLECTS COMMITMENTS
3. ASSEMBLE QUALITY INDICATORS OF "WELDING CONCERNS" BY TYPE AND PLANT
4. TREND AND EVALUATE EFFECT OF "QUALITY INDICATORS" ON PROGRAMS
5. ISSUE ADEQUACY STATEMENT REGARDING WRITTEN PROGRAMS TO IMPLEMENT/ CONTROL WELDING

PHASE II

1. PERFORM WELDING PROGRAM IMPLEMENTATION AUDIT
   - CONSTRUCTION PROGRAM IMPLEMENTATION
   - OPERATIONS PROGRAM IMPLEMENTATION
2. EVALUATE NEED FOR ADDITIONAL REINSPECTIONS
3. IMPLEMENT ANY ADDITIONAL REINSPECTIONS AND DEFICIENCY RESOLUTIONS (BOTH INDIVIDUAL AND GENERIC CASES)
4. WELDING PROJECT WILL ISSUE FINAL REPORTS, EACH PLANT

0%002.02

Attendance Roster
Exit Meeting

NAME	AFFILIATION
Robert Birchell	Compliance SQN
Larry S. Bryant	Mech. Maint. SQN
Gary S. Boces	Mech Maint. SQN
M.A. Skarzinski	SQN
Paul Herman	NRC
Glenn B. Kirk	SQN
William R. Ramsey	SQN
B. Patterson	SQN
P.R. Wallace	SQN
N. Choules	NRC- Region III
Dolan Falconer	NRC-Region II
Donald S. Brinkman	NRC-OIE
J.T. Taffanstedt	SNIP
J. Blankenship	Info. Office
D. Persinko	NRC/DHFT/MTB
R. Lloyd	NRC/IE
Arthur Howell, III	NRC/IE
Owen Gomley	NRC/IE
L. Watson	NRC Region II
Michael Purcell	Regulatory Engineering
L. McCormick	Regulatory Engineering
W.S. Wilburn	Site Services
M.R. Harding	SQN
H.D. Elkins, Jr.	SQN
R.V. Pierce	Mech. Mth./SNP
R.W. Olson	SQN-Modification
R.C. Denney	Design Services SNP
A.R. Meller	NSS
F.E. Denny	OE-QMS
R.N. Butler	QA Staff
Roger Landis	Mods/Mech
M. Sedlacik	Mod
L. Alexander	Mod
W. Liu	NRC Region II
R.W. Newsome	NRC Region II
H. Kerch	NRC Region I
A. Herdt	NRC Region II
B. Crowley	NRC Region II
D.E. Smith	NRC/NRR/PWR-A
George Georgiev	NRC/IE
J.H. Fox	TVA
Gary J. Pitzl	TVA Office of Nuclear Power
J. Brandy	MM/SNP
Gerald Minton	TVA
L. Mink	TVA
D Mickler	Construction
Robert D. Stout	Consultant - Brookhaven National Laboratory
Carl Czajkowski	Brookhaven National Laboratory
Paul Masters	Consultant - Brookhaven National Laboratory
William H. Munse	Consultant - Brookhaven National Laboratory
J.W. Coan	TVA
W.O. Long	NRC

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

Upton Long Island New York 11973

Department of Nuclear Energy

(516) 282
FTS 666- 3349

April 25, 1986

Dr. B.D. Liaw
Engineering Branch
Mail Stop P-1132
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Trip Report, Sequoyah Nuclear Power Plant, FIN A-3839, TVA Welding Concerns

Dear Dr. Liaw:

The SQN Expert Welding Consultant Team met with TVA and NRC personnel during the period of April 14 - 16, 1986, at the Sequoyah Nuclear Power Plant. The objectives of this visit were:

1. In plant hands-on evaluation of support and piping welds previously inspected by TVA, Bechtel and the NRC Inspection Van personnel.

2. Review as applicable welding concerns with TVA management, welding engineers, construction personnel, quality assurance representatives and inspection personnel.

3. Provide preliminary BNL Expert Welding Team input and recommendations regarding SQN/TVA action plan and inspection activities to date.

SQN Expert Welding Team participants were Messrs. C.J. Czajkowski, M.H. Schuster, W.D. Doty and C.E. Hartbower.

Coordination and NRC coverage was provided by D.E. Smith, Engineering Branch, DPL-1, Bethesda, MD.

The following trip activities are discussed in this trip report:

A. Entrance meeting
B. Field evaluation of support and piping welds
C. Interview of TVA personnel
D. Exit meeting
E. Summary conclusion

April 15, 1986

A. Entrance Meeting

The entrance meeting was convened at approximately 10 a.m. The meeti was attended by TVA personnel, NRC, and the Welding Team consultants (Attac ments #1.)

D. Smith, USNRC, Engineering Branch, Bethesda, MD., and C.J.Czajkowski provided an overview of the Welding Team objectives and purpose for the sit visit (as described in introduction).

L. Martin, TVA, discussed the status and current schedule of the Watts Bar employee concern program. He also stated that at this time the employe welding concern program at SQN 1 and 2 is not a critical path item and that the final report is in its final draft condition and will be submitted to th NRC in a short period of time.

April 15, 1986

B. Field Evaluation of Support/Piping Welds

In order to provide a more objective evaluation of SQN employee concern reinspection plan, and the SQN corrective actions and final report, the Welding Team was provided a tour of units I and II with access to reinspection sample lot structural component and piping welds which had been inspected by TVA and the USNRC NDE Van personnel. Personnel who participated in the tour were D. Smith, USNRC; Welding Team members, Messrs. C.J. Czajkowski, M.H. Schuster, W.D. Doty, C.E. Hartbower; and TVA members, R.M. Jesse, J.R. Fox, S.P. Stangnolia and C.W. Hartmaker.

Weld and component identification was provided by TVA personnel. Piping isometrics and design documents were neither reviewed nor verified to unique identification or "as-built" condition.

Welding Team members examined approximately 33 structural supports and piping welds. Structural support welds were predominantly of fillet weld configuration. Piping welds were Butt and fillet socket with emphasis on butt welds. Weld anomalies were noted and comparison with the TVA and NRC inspection reports will be accomplished. To date, the Welding Team is not in possession of the TVA inspection reports.

April 15, 1986

C. Interview of TVA Personnel

Upon completion of the field evaluation of piping and structural support welds, the Welding Team and USNRC representatives met with TVA personnel (Attachment II). The purpose of this meeting was to give Welding Team members an opportunity to discuss and review previously identified welding concerns with welding engineering, quality control, quality assurance, and TVA management. Specifics such as inspection personnel experience, training requirements, work assignments, weld rod issue, weld rod control, welder testing ASME Section IX/AWS D1.1, maintenance of welder qualifications, welder tracking requirements were discussed at length with TVA personnel.

Weld inspection requirements such as pre-weld, in-process and post-weld inspection, frequency of these inspections, verification of welder qualifications during these inspections, inspection tools, receipt inspection requirements, welding procedures, inspection procedures, weld documentation requirements (traveler), manpower requirements, weld repair requirements both documentation and in-process were also verified by discussion with the TVA attendees. Methodology utilized for TVA sample lot inspections and selection of welds and structural components was provided by TVA personnel. It should be noted that TVA participants at this meeting were cooperative and forthright in their responses to the Welding Team members.

April 15, 1986

D. Exit Meeting

The Exit critique of the visit was convened at approximately 1600 hours (Attachment III). D. Smith, USNRC, Materials Engineering, acted as the group spokesman.

The Welding Team requested:

1. Purchase order requirements, design specification and materials certifications for steam generator structural supports. Review of charpy V notch requirements will be performed upon receipt of the requested documents.

2. List of all employee concerns related to SQN. The Welding Team requests that these be provided on an ongoing basis until completion of this assignment.

3. SQN TVA quality control personnel certification records.

4. SQN TVA reinspection reports.

5. SQN corrective actions resolution and rework schedule.

6. SQN TVA welding concern closeout matrix (if available). (Concern vs. NSRS closeout report).

7. The Welding Team observed three structural supports with unwelded sections.

8. Provide independent reports/conclusions which have been or may be initiated in response to reinspection results.

-3-

Conclusions

In general, Welding Team members did not express specific concern as a result of this SQN visit. There was some discussion regarding NDE requirements for ASME Section III, Code class III piping weld. The differences between other industry standards and Nuclear NDE/inspection requirements was discussed at length.

If there are any questions, please contact the undersigned.

Milford H. Schuster

Milford H. Schuster

Carl J. Czajkowski

Carl J. Czajkowski

MHS:CJC/ad
Attachments

cc: Welding Team
P. Cortland
W. Kato
W. Long
D. Smith
J. Weeks

Attendance Roster
Entrance Meeting 4/15/86

NAME	AFFILIATION
Robert Birchell	SNP Compliance
Glenn Kirk	SQN
W.D. Doty	Consultant - Brookhaven National Laboratory
C.E. Hartbower	Consultant - Brookhaven National Laboratory
D.E. Smith	NRC/NRR/PWR-A
Carl Czajkowski	Brookhaven National Laboratory
Milford Schuster	Brookhaven National Laboratory
John Fox	DNE
I.E. Martin	DNP
S. P. Stagnolia	Nuclear Construction
R.M. Jessee	Nuclear Engineering
Carl Hartmaker	DNE
Roger Field, Jr.	SQN
L.M. Nobles	SQN
David Humble	Mech. Maint.
Richard Butler	SQN-QA
Robert W. Olson	SQN
Larry Alexander	SQN

Attendance Roster
Interview Meeting 4/15/86

NAME	AFFILIATION
M.H. Schuster	Brookhaven National Laboratory
John H. Fox	TVA
Lawrence Warner	TVA-ISI
Dennis Allen	TVA-ISI
Brett McCreary	TVA-QA
Carl Czajkowski	Brookhaven National Laboratory
David E. Smith	NRC
Carl W. Hartmaker	TVA-DNE
S Stagnolia	TVA-OC-WTG
Roger Field, Jr.	TVA-OE
Robert M. Jessee	TVA-DNE
W.D. Doty	Consultant - Brookhaven National Laboratory
C.E. Hartbower	Consultant - Brookhaven National Laboratory

TENNESSEE VALLEY AUTHORITY
CHATTANOOGA TENNESSEE 37401

5N 157B Lookout Place

AUG 0 1 1986

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Project Director
PWR Project Directorate No. 4
Division of Pressurized Water Reactor (PWR)
Licensing A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

In the Matter of the	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

Please refer to your letter to S. A. White dated June 10, 1986 which requested additional information on the Sequoyah Nuclear Plant Phase II Welding Project Reports. Enclosed is the response to your request.

If there are any questions, please get in touch with R. H. Shell at FTS 858-2688.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. Gridley

R. Gridley, Director
Nuclear Safety and Licensing

Enclosure
cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attention: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE

SEQUOYAH NUCLEAR PLANT WELDING PROJECT REPORTS

1. In 2.0 APTECH ENGINEERING REPORT (Supplemental Information), Page 3, 8th line, it is stated that "In the case of the feedwater lug, no engineering evaluation was requested by the plant." Why was installing the missing welds to drawing requirements chosen as the means of resolving a missing weld problem rather than performing an engineering evaluation as had been done with a very similar problem? Demonstrate that code requirements were met without installing the missing welds.

   Response: When missing welds are identified during inspections, it is usually much easier to add the welds as required by the drawing (provided there is sufficient access) than to request engineering disposition to leave "as is." This was the case for feedwater lug FDH-203. However, when sufficient access does not permit welding, engineering disposition and subsequent drawing changes are initiated. This was the case for the Safety Injection System stanchion-to-pipe weld, 1-SIH-17.

   In the case of 1-SIH-17, engineering gave preliminary approval to leave "as is" since cursory calculations showed that the actual weld provided was adequate for design loads. Therefore, addition of the weld was not required. Engineering will provide final calculations to demonstrate structural adequacy of the subject support when the drawing is revised and reissued.

2. The term "separated weld" is used in 2.0 APTECH ENGINEERING REPORT (Supplemental Information), Page 3, 12th line. Define the basis for your assessment of this weld failure as being due to operating transients and not having been due to poor weld quality or cracking during fabrication.

   Response: Since no cracked welds have been found during the reinspections, there is no reason to question the quality of the construction welds. Conversely, there is not a readily identifiable basis for attributing the occurence to an operating transient. TVA determined this to be an isolated case since no other cracked welds or damaged supports were found in the same area and the cause is indeterminate.

3. In the APTECH ENGINEERING REPORT, the Table titled, "NOI DESCRIPTIONS - SEQUOYAH NUCLEAR PLANT UNIT 1". NOI Number SQ0201, under Disposition and Additional Comments it is stated: ". . . clean weld area per SQM-17, paint and re-examine." Explain how code requirements were met with the examination following painting.

Response: The note under MOI SQ0201 in the APTECH ENGINEERING REPORT is an editorial error. The Maintenance Instruction required that the subject weld be added, cleaned, visual and PT examined, then painted. The inspection report shows that the weld passed final examination (visual and PT) on 12/9/85 and has not yet been painted.

4. In the APTECH ENGINEERING REPORT, Table 4-1 lists 5 Licensing Event Reports concerned with welds. Provide the number of LERs evaluated in this search. Were any failure analyses conducted of the welds covered by these LERs? If so, please provide them.

Response: There were 840 LERs evaluated in the search. A metallurgical failure analysis was done in conjunction with LER 80156. The failure analysis involved a vendor weld (seal water injection line to reactor coolant pump weld).

5. Were there ever other than E7018 carbon/low alloy steel shielded metal arc welding electrodes on the Sequoyah site, such as E8018C3? Demonstrate that incorrect electrodes were not used on any weldment.

Response:

A. Construction Phase

Yes, small quantities of E6010, E11018X and various other types of specialty maintenance electrodes were kept on site. These materials and their use were strictly controlled. Their uses were limited to such things as construction plant (temporary construction facility) maintenance and construction; maintenance of construction equipment; hard facing of construction equipment cutting edges; crane boom repair; build up for hard facing of worn construction equipment; and the fabrication of construction jigs and fixtures.

In addition to the previously described maintenance materials, small quantities of E8018C3 and E7010A1 materials were used on appropriate permanent plant features. The use of these materials was also strictly controlled in accordance with the construction Quality Assurance/Quality Control Program.

Checks and balances were reflected in construction procedures to insure the proper procurement, storage, and application of welding materials used for permanent plant construction. These included the recording and verification by QC Inspectors of filler materials by type of safety related pipewelds and a QC surveillance to spot check proper filler material application on all safety related welding. In addition QA reviews of safety related pipeweld records included electrode type as a check point.

B. Operation Phase

Yes, like construction, small quantities of various types of other electrodes are maintained for specialty welding and specialized maintenance applications. These include carbon steel coated electrodes other than E7018 which have not been used on safety related plant features. These applications include maintenance of shop and shop equipment, fabrication of temporary jigs and fixtures, and noncritical maintenance of non-safety related balance of plant items. These materials and their applications are strictly controlled in accordance with approved plant procedures.

Maintenance and modification procedures provide for the QC verification of proper filler material use for safety related applications. This verification provides indirect traceability to heat/lot numbers. In addition, a QA surveillance program provides additional spot checking of proper electrode usage.

6. For the Bechtel Audit, what were the total number of welders and inspectors in the populations from which the audit samples were taken? Provide separate totals for the Office of Construction and Nuclear Operations.

Response: Populations from which the Bechtel Audit Team selected are as follows:

Organization	Welders	Inspectors
Construction	approx. 3100	approx. 180
Nuclear Operations	approx. 205	approx. 120

7. The TVA Reinspections checked the relative magnetism for all welds, austenitic and ferritic. What was the procedure for this inspection method? Provide justification for different levels of magnetism and their acceptance criteria, particularly "weakly magnetic".

Response: The magnetic check for generic filler metal type (i.e., ferritic or austenitic) was performed by touching a small permanent magnet to the weld deposit and noting his judgement as to whether the deposit was strongly, weakly, or non-magnetic. The inspector also noted whether the base materials being joined were stainless or carbon steel.

Evaluation of correctness of filler metal was done by OE according to the following guidelines:

1. The correct weld metal for welds joining stainless steel to stainless steel should be weakly magnetic or non-magnetic.

2. The correct weld metal for welds joining stainless steel to carbon steel should be weakly magnetic or non-magnetic.

3. The correct weld metal for welds joining carbon steel to carbon steel should be strongly magnetic.

The above guidelines are as contained in P.S.3.C.11.1 (R1).

The "weakly magnetic" category as a permissible condition for items 1 and 2 above reflects that the correct stainless steel weld metal used in these welds should appear non-magnetic or weakly magnetic depending on delta ferrite content and/or degree of base metal dilution.

8. Cracks were not listed as one of the attributes in the tables of TVA Reinspection Report. Were any cracks found during the TVA Reinspection? Also, porosity was not an attribute listed in the structural welds table. What was the rejection rate for porosity in the structural welds in the TVA Reinspection?

Response: Both cracks and porosity were attributes that were checked in the reinspection effort. No cracks were found during the reinspection. Rejectable porosity was not found on any structural welds.

9. In 4.4.1, Page 8, line 21, of the five welds which were ground, were the manufacturer's minimum wall thickness requirements encroached upon? If so, to what extent?

Response: Only one weld (2CCF-68) of the five which were ground to reduce surface indications had its manufacturer's minimum wall thickness encroached upon. This weld is in a 4-inch schedule 40 carbon steel pipe. The measured thickness localized ground area is 0.198". This is 0.0094" less than the manufacturer's minimum wall requirement of 0.2074" but is more than twice the design minimum wall of 0.08".

10. In 4.4.1, Page 10, line 1, the rough condition of two welds found during the reinspection is discussed. Provide information that justifies the statement, "The indepth investigation of the welder and inspector qualification revealed no indications of inadequacy of the welder or inspector capabilities." What was done to demonstrate that this level of workmanship by this welder and/or judgement by this inspector were not repeated elsewhere at Sequoyah?

Response: After proper removal of paint, both welds were inspectable by the penetrant method. The inspectors' certification files were reviewed and both inspectors in question were found to have at least two years experience at penetrant testing when the inspections were made. The welder was initially certified in May 1975 and had welded in nuclear applications off and on since that time. TVA determined that no further investigation of the inspectors' or welders' work was necessary.

11. In 4.4.1, Page 11, in the table titled "PIPING WELDS", the rejection rate when expressed in terms of the percentage of welds rejected is 56% (184/333). Even allowing for some rejected welds counted more than once because of more than one rejectable attribute, the rejection rate is very high. a) What is the root cause of this high rejection rate of originally inspected and accepted welds? b) Is there any basis for concluding that there is a connection between the employee concerns expressing doubt about inspectors capabilities or that harassment and intimidation of inspectors occurred? c) With respect to question a), address in particular the attribute underfill, which has very specific code requirements. d) The arc strike/weld spatter rejection rate was 31%. What is the root cause for this high rejection rate? e) What were the original inspection criteria for these weld attributes? f) What were the reinspection criteria for these attributes? g) What is the justification for elimination of inspecting arc strikes for cracks in G-29C?

Response: The reinspection rejection rate on a per weld basis to inspection requirements is 24% (80/333). The 184 arc strikes and weld spatter indications were reportable but not rejectable. Base metal outside the weld area was not required to be examined by the construction code. The procedure used for the reinspection required base metal indications outside the weld to be reported.

Any reinspection effort will typically have a rejection rate of 5-10 percent. However, a reinspection such as this can have a rejection rate approaching 20-25 percent because of the circumstances under which the reinspection was made.

a. What is the root cause of this high rejection rate of originally inspected and accepted welds?

Response: The root cause of the high discrepancy rate involves both psychological factors and a changing inspection philosophy in recent years. Inspectors performing this reinspection anticipated "second-guessing" of their judgements by others. Because there is judgement involved in weld inspection close calls will inevitably become rejects under such conditions. It is unrealistic to expect the results of a reinspection performed under the degree of scrutiny involved here to yield results comparable to those performed in the 1970-80 era. This does not imply inadequate inspection during construction. It does reflect a change in weld inspection philosophy and methodology over the past 15 years and most particularly in the past 2-3 years. The significant change involves less reliance on the inspector's eyes and judgement of the weld as a whole, and more on quantitative measurement of every attribute on every increment of weld.

To a lesser degree, the current discrepancy rate is a result of changes in acceptance criteria (see "d" below).

b. "Is there any basis for concluding that there is a connection between the employee concerns expressing doubt about inspectors capabilities or that harassment and intimidation of inspectors occurred?

Response: The program was working properly and inspectors were performing properly. We have no evidence that would support the concerns about inspector capability and inspector harassment or intimidation.

c. With respect to question a), address in particular the attribute underfill, which has very specific code requirements.

Response: Seven of the 11 welds rejected for underfill involve sockolet branch connection fittings to pipe runs. These fittings are proprietary products designed to provide integral reinforcement of the branch opening. Because of the configuration of the fittings themselves and the geometry of the connection as a whole, the correct weld size and configuration is not obvious. This is particularly so in the cases where there is little difference in the size of the run pipe and branch connection.

The remaining four instances of underfill involved welds joining members of unequal thickness (pipe to valve or fitting). Here the reported underfill was with respect to the edge of the thicker member. However, the weld thickness was greater than the minimum pipe wall thickness. (Refer to Note 6 of Appendix 4.4.)

We agree that the code requirements are explicit with regard to underfill as applied to typical piping girth butt welds. Underfill in such welds has not historically been a problem and was not in this reinspection.

d. The arc strike/weld spatter rejection rate was 31%. What is the root cause for this high rejection rate?

Response: TVA procedures in use during the construction of Sequoyah Nuclear Plant prior to March 21, 1979 did not require the reporting of arc strikes unless a crack was present. The procedures used during the reinspection did require reporting of arc strikes. The data simply reflects the procedure requirements in the two different time frames.

Weld spatter has been prohibited by TVA inspection criteria since 1970. Neither the construction era nor current piping codes (ASME Section III and B31.1) address the condition. Although lumped with arc strikes as a discrepant condition, it was reported on only three piping welds.

e. What were the original inspection criteria for these weld attributes?

Response: Please refer to item "d" for response.

f. What were the reinspection criteria for these attributes?

Response: Both arc strikes and weld spatter were treated as discrepant conditions during the reinspection.

g. What is the justification for elimination of inspecting arc strikes for cracks in G-29C?

Response Cracks have been and are presently prohibited in welds and adjacent base material in TVA inspection procedures. This prohibition includes cracks in arc strikes or anywhere else within the zone of inspection.

12. In 4.4.1, Page 11 and 4.2.1, page 13, in the tables titled "PIPING WELDS" and "STRUCTURAL WELDS" respectively, expressing weld rejection rates based upon the attribute inches is misleading. There was only a finite number of welds inspected, and a qualified craftsman should be capable of making welds which meet all of the attributes in all of the inches submitted to inspection. For these tables, please rearrange the data as follows:

Response:

PIPE WELDS

TYPE OF WELD	NO. OF WELDS REPORTABLE INDICATIONS	NO. OF WELDS WITH REINSPECTED BY CODE	NO. OF WELDS REJECTED BY CODE
Socket Welds			
Office of Const.	204	78	0
Nuclear Ops.	34	6	0
Butt Welds			
Office of Const.	68	46	0
Nuclear Ops.	22	6	0
Attachment to Pipe Wall			
Office of Const.	5	[illegible]	0
Nuclear Ops.	0	0	0
Total Welds			
Office of Const.	277	127	0
Nuclear Ops.	56	12	0

STRUCTURAL WELDS

TYPE OF WELD	NO. OF WELDS REINSPECTED	NO. OF WELDS WITH REPORTABLE INDICATIONS	*NO. OF WELD JOINTS NOT MEETING DESIGN REQUIREMENTS
Fillet Welds			
Office of Const.	1080	160	0
Nuclear Ops.	148	21	0
Butt Welds			
Office of Const.	50	4	0
Nuclear Ops.	0	0	0
Other (specify) - Flare			
Office of Const.	92	24	0
Nuclear Ops	24	2	0

*Weld joints were evaluated not individual weld segments.

13. In the TVA Reinspection Report, a comparison is made between original inspection results and the reinspection results for piping welds. If such a comparison can be made in a quantitative manner for structural welds, please present the data.

    Response: The original inspection was made on an item basis rather than individual weld, consequently, we do not believe possible to make a meaningful weld-by-weld comparison between the reinspection results and the original inspection results for structural welds.

14. Referring to the Legend for Table 4.2, in the Final Resolution column, define the meaning of the letter codes in parentheses.

    Response: The letter codes located within the parenthesis in the legend of the final Resolution of Table 4.2 denote various design soctions within the Division of Nuclear Engineering that had lead responsibility of the resolutions addressed by the code of A1 through A10.

    NEB CSM - Nuclear Engineering Branch - Code Standards & Materials Section
    CEB M2 - Civil Engineering Branch Mechanical Analysis Section #2
    SQEP M3 - Sequoyah Engineering Project Mechanical Design Section #3
    SQEP C3 - Sequoyah Engineering Project Civil Design Section #3

15. There are some employee concerns about various structures not being in accordance with the as-built drawings. Did the TVA reinspection address this issue? If so, report the deviations from the as-built drawings found. Report the deviations in configuration as to type of deviation, the rate of a type of deviation compared to the number in the reinspection population, and if such deviations resulted in not meeting code requirements.

Response: No. This reinspection program was not intended to address deviations in configuration from as-built drawings. This subject is being addressed by TVA's employee concerns program.

16. Table 4.3 shows that a total of 50 structures were reinspected in the TVA reinspection program. However, Table 4.4 shows only 31 structures as having been reinspected. Explain the discrepancy.

-Response: Table 4.3 is correct for number of <u>structures</u>. Table 4.4 shows number of items or what was defined in Phase I as a <u>package</u>. An <u>item</u> may contain only one structure or a number of structures.

To correct the Table 4.3, the title should read "NUMBER OF REINSPECTED STRUCTURES".

There are 31 packages (items) shown in Table 4.4.

Two packages (No. 10 and No. 30) are not reported in Table 4.4. Item #10 was not reinspected and Item #30 is reported in the Mechanical Reinspection (Table 4.2).

The remaining packages breakdown to the following number of structures.

Items 2 thru 9)
12)
14 thru 16) All contain one structure
18)
20 thru 21)
22 thru 29)
31)

Item 1 - 2 structures
11 - 2 structures
13 - 3 structures
17 - 2 structures
19 - 14 structures

Attachment 11

ROBERT D. STOUT, Consultant
141 128 E. Market Street
Bethlehem, Pa. 18018

September 8, 1986

Office Phone: (215) 861-4288

Home Phone: (215) 866-0698

Mr. Carl J. Czajkowski
Brookhaven National Laboratory
Upton, N.Y. 11973

Dear Mr. Czajkowski:

This letter is a statement of my individual reactions as a member of the expert welding team formed by Brookhaven National Laboratory. This team was assigned to review the adequacy of the TVA welding program and the corrective actions taken by TVA in response to expressed employee concerns. The first segment of the work was to examine the program and employee concerns relevant to the Sequoyah Station. The committee embarked on a physical survey of the safety-related Class 3 piping welds and structural welds, and also considered 117 employee concerns pertaining to welding together with the TVA responses supplied.

The physical survey did not raise any serious doubts about the quality and adequacy of the weldments based on visual examination. The fact that the station has been in operation for some six years without significant welding failures supports the adequacy of the welding.

No attempt was made to analyze the documents pertaining to the programmatic aspects of the TVA operation beyond the extensive discussions among the committee and NRC representatives. The examination of the employee concerns and the responses of TVA did not reveal any evidence of gross departures from accepted practices. The chief weaknesses seem to be associated with an overelaborated program which has suffered from the failure of management to maintain careful control of it.

In summary my conclusions are as follows:

1. The Class 3 welded construction at the Sequoyah Station appears from the reinspection reports of TVA and NRC to be of acceptable quality. The visual inspection by the team supported this view.

2. No employee concerns were confirmed which threaten the safety of the plant.

3. TVA must restore full confidence in their management of the welding program.

4. There were no inadequacies revealed in the welded construction which would prevent resumption of operation of the Sequoyah Station.

Very truly yours,

R D Stout

Robert D. Stout

Attachment 12

(813) 542 0074

PAUL E. MASTERS

WELDING ENGINEERING

1951 PALACO GRANDE PARKWAY

CAPE CORAL, FLORIDA 33904

September 13, 1986

Mr. Carl J. Czajkowski
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11793

Subject: Position Statement - Contract Number 225771-S

It is the writer's opinion that the Technical Evaluation Report (TER) relative to the welding concern program at TVA's Sequoyah units 1 and 2, dated August, 1986 and edited by Carl J. Czajkowski, properely reflects the concensus of the Expert Team's evaluation of the utility's response and action plan for addressing the employee welding concerns.

During this evaluation it was quite evident that there was poor over-all management by TVA. They appear to treat each site as an entity. Their documents, which are innumerable, were confusing, overlapping, repetitive, unclear and lacked continuity with regard to each other and to revisions. This certainly results in a lack of understanding by all craft personnel and their supervisors. This also appears to create confusion in the control by NRC in monitoring TVA's work. This situation adds to the public's already poor opinion of the control and safety of nuclear power plants. In the recent meeting with NRC this was quite evident by the discussions within the NRC group relative to the TVA situation.

The writer seriously questions the use of TVA's Visual Weld Accept-Criteria for Structural welding, NCIG-01, for the reinspection or for initial welding. The document correctly states in its introduction paragraph 1.1.1.1 of AWS D1.1 and that it was a new paragraph in the 1985 edition. Also included is part of the Commentary on this paragraph, but fails to include the statement in the preceeding paragraph to the effect that any modifications of the Code deemed necessary by the authorities should be clearly referenced in the construction agreement between the owner and the contractor. In this case, TVA's, the justification for NCIG-01 is being applied many years after the original specifications were written.

It is ludicrous to use a lower weld quality requirement, NCIG-01, than the original, D1.1, as the criterion to reinspect questionable welding. Further justification of acceptance of undersize welds, lack of specified number of welds, etc., by engineering reevaluation shows poor original design work by specifying over welding or a reduction of the safety factor for the connection. Again adding to the public's poor opinion of nuclear power projects.

Position Statement Continued

During the Teams on-site visit to the Sequoyah plant on Feb. 26-28, 1986 it was the writer's opinion that the weld quality was good. However it must be realized that this did not include weld size, length, etc., as we did not have these requirements when viewing the welds.

The writer cannot make a statement relative to the start-up of the Sequoyah units as that must be based on the acceptance of the engineering reevaluation judgement and the acceptance of a lesser weld quality requirement used in the reevaluation and weld reinspection results.

Paul E. Masters

Paul E. Masters

W. H. Munse
1208 Devonshire Dr.
Champaign, IL 61821

October 1, 1986

Mr. Carl J. Czaykowski
Materials Technology Div.
Brookhaven National Laboratory
Upton, New York 11973

Welding Concern Program at TVA's Sequoyah Plants

Dear Mr. Czaykowski:

This letter report is in answer to your request to the expert welding team for individual evaluation of the Weld Evaluation Program at the Sequoyah Nuclear Plants.

Your Technical Evaluation Report of August 1986 includes the expert welding team's evaluations of the detailed Employee Concerns relative to the Sequoyah Units. In addition, I should like to offer the following comments and analysis concerning various aspects of the overall Welding Program. These comments relate to the employee concerns as well as to a variety of other questions, analyses and reports.

1. Evaluation of Construction Welding and Inspection. Both the Aptech and Bechtel audits involved only an examination of the welding and inspection documentation for SQN and not an examination of the actual welds. Based on these studies the Welding Program concludes that (a) "the welding program for the TVA Sequoyah Nuclear Plant is being effectively implemented and that the installed hardware is suitable for service," and, (b) "that TVA had an effective program related to welding and NDE." However, there are many questions raised in these audits that lead one to question these conclusions. For example, it is indicated in the Aptech report that the welding and inspection programs should provide quality welds, if properly implemented. But, were they properly implemented?

An examination of the data on the weld reinspections at SQN shows the existence of numerous welds with rejectable attributes (See Tables 4.2 and 4.3 of the Vol. III report) even though less stringent requirements were employed in the reinspection than had originally been specified. This certainly does not indicate that the original designs and specifications had been properly implemented.

In another portion of the Aptech report it is noted that 46,430 hours of operation at SQN 1 and 2 had not identified any welds which are not of sufficient quality for their intended service. This may be true for operating-service conditions but does not cover the maximum design loadings for which the plants are designed; the operating conditions should not be expected to produce any problems in the welds since they would not stress the welds to the magnitude that would be reached by the application of the maximum design loads.

In the Bechtel report it is observed that, "Many of the referenced implementing procedures in the NQAM were found to be excessively long, ambiguous, and do not give clear and concise instructions to personnel to perform their activity." Again, with such conditions existing it is hard to see how the welding and inspection personnel could properly perform their functions.

Another evaluation which relates to the TVA welding and inspection can be found in the QAE-80-Z report. Although this report applies specifically to the Watts Bar Plants, much of it is of a general nature and no doubt generically applicable or appropriate to the Sequoyah Plants too. Many recommendations are made, including the following:

(a) Disciplinary action should be taken against welders who bypass holdpoints.

(b) Responsibility for meeting QA/QC requirements should be emphasized.

(c) A CONST qualification/certification program for visual weld inspection should be established.

(d) The Welding Engineering Units should supply personnel with information on weld sequencing.

(e) Complete welding procedure specifications should be at the foreman's station.

(f) All necessary tools, gauges, and instruments necessary to determine weld acceptance should be made more readily available.

(g) More surveillance checks should be made on in process welding operations.

(h) A standardized system for continuity of welders' qualification and welding procedure/performance qualification cross-reference should be developed.

(i) A complete rework of distribution, control, content and utilization of G-29 specifications should be made. G-29 needs to be at work stations.

(j) Approximately 50 percent of the site welding engineers have insufficient background, experience and education to perform as qualified welding engineers.

(k) Welding engineers spend the majority of their time as Technical unit supervisor to the welding inspectors because of inexperienced welding inspection personnel.

(l) There is failure by crafts to follow instruction in work packages and other documents, and to bypass holdpoints.

(m) Inspection personnel are not always provided with the basic tools needed to perform the inspection functions.

(n) With the exception of the apprentice program, OJT for welders is not provided.

(o) Nuclear projects are constantly cited by NRC for the lack of control of filler material.

(p) Quality levels on civil structural drawings are confusing and need some type of resolution.

(q) Redundancy in QCI and G-29 on NDE procedures should be eliminated.

With such questions being raised and so many recommendations being made, it is difficult to imagine that the welding and inspection programs are being or have been properly implemented.

Finally, it is noted that the Aptech and Bechtel Audits involved only reviews of records, and the TVA's reinspection is primarily through paint and of V, PT and MT procedures. Little has been done to provide a volumetric evaluation (reinspection) of the Class 1 and 2 Sequoyah welds. This appears to be one of the major shortcomings of the Welding Evaluation Project. In fact there has been no indication of a systematic re-examination of the radiographs for such welds.

2. <u>Specifications</u>. A second area of concern is the application of the AWS D1.1 Code. As a member of the AWS Structural Welding Committee the writer has considered the Structural Welding Code to provide minimum requirements. However, from the TVA documents it is apparent that they have interpreted the Code in the broadest sense and have provided less stringent requirements than in D1.1 (see Table A of Vol. II for the TVA Comparison of G-29 to D1.1 and NCIG-01 to D 1.1) Code. There is less safety provided by G-29 and NCIG-01 than is provided by the D1.1 Code. The D1.1 Code is used primarily for buildings and bridges. In view of the critical nature of a failure at a nuclear plant, one would generally assume that the provisions for a nuclear plant would be <u>more</u> stringent rather than less stringent than for buildings. Furthermore, there has been no sound justification given for the relaxation of requirements. In the commentary of the 1985 AWS Code it is indicated that, "when modifications are approved, evaluation of suit-ability for service using modern fracture mechanics techniques, a history of satisfactory service, or experimental evidence is recognized as a

suitable basis for alternate acceptance criteria for welds." This type of justification for modification has not been clearly demonstrated in the documents available to date.

A second procedure that has been used to justify the use of welds that fail to satisfy the D1.1 is to make an engineering check to establish suitability for service. Such a procedure using NCIG-01, since it is a relaxation of the D1.1 Code, will provide a structure that is not as safe as if the welds met the D1.1 requirements. The necessity to use such a procedure also suggests that the original design may have been poor or overly conservative, both of which indicate poor engineering.

3. Summary and Conclusions. From the TER it is shown that the Sequoyah units have suffered some areas of programmatic breakdown but the hardware itself does not appear to have defects of great detriment or magnitude. However, further study of this question would seem desirable. The evaluation of the Employee Concerns indicates that many have not been substantiated, nor have they all been shown to be groundless or false.

In NSRS report No. I-85-373-NPS interviews with 17 NDE inspectors concerning OJT, it is indicated that most inspectors expressed concerns to varying degrees with regard to the validity of some of the claimed OJT; falsification of records and favoritism are reported. Thus, the concern over OJT was basically substantiated. In addition, it is indicated that some inspectors didn't feel they were qualified for some of their tasks.

The ERT investigations report on OJT also indicates both a programmatic breakdown and falsification of records within the TVA NDE training/certification program. Again, a substantiation of the OJT concern.

Based on the various studies and evaluations made to date it appears that there are a number of shortcomings in the TVA program.

(a) Inadequate training at all levels.

(b) Poor record keeping and control in welding and inspection.

(c) Many procedures are excessively long, ambiguous, and do not give clear and concise instructions to personnel to perform their activities. There are many overlapping documents.

(d) There is excessive redundancy in the various weld related construction documents, the various weld related inspection instructions, the various weld related standard operating procedures, and in the various design, construction and inspection specifications and codes. This causes confusion.

(e) There are so many documents for a project that relate to welding and inspection that it is essentially impossible for the personnel to be aware of all the requirements to which they should be working.

These are all shortcomings that can be corrected. However, the fact that they have existed raises questions as to the quality of the welded nuclear structures.

The Weld Evaluation Program was designed to establish a high level of confidence in all welds at the Sequoyah Nuclear plant. However, with the relaxations in specifications needed for weld acceptance, and the need to use an extra engineering check of suitability for service does not help to instill a high level of confidence in the welding.

In the reinspection program the welds were generally rated as average or better. However, in 345 piping welds 9 were rated unacceptable as to the quality of weld workmanship. In 7368 inches of structural welds reinspected (in 1394 welds), 1040 inches were found with indications, 10.38 percent were undersize and 9 joints had welds omitted. Again, although most welds appear visually to be of average or good structural quality (this was also the writer's general observation from a personal examination of a limited number of welds), the number of indications reported is of such a magnitude that the desired high level of confidence in the welding and inspection does not appear to have been achieved. Acceptance has been achieved only by employing specifications less stringent than originally specified, and through the application of "an engineering evaluation," with little indication of what this evaluation entailed. A greater confidence in such acceptance criteria might be possible if some quantitative measures were presented to such questions as, what percentage the welds were undersize (per the original specified size), and from the engineering evaluation, what is the magnitude of understress in the actual welds (based on the maximum allowable design stresses).

From the above discussion it should be evident that the writer believes further analysis and justification of the Sequoyah welding and inspection would be desirable to demonstrate whether or not the welding at the Sequoyah plants is of a quality to satisfy the desired Code and Specification design requirements.

Very truly yours,

W. H. Munse

W. H. Munse
Professor Emeritus of
Civil Engineering

WHM/jh

September 16, 1986

Mr. Carl Czajkowski
Brookhaven National Lab.
Upton, Long Island
New York 11973

Dear Carl,

Enclosed please find an Executive Summary of my views regarding the adequacy of the TVA SQN Welding Concern Program and the adequacy of TVA's corrective actions in the areas of Welder Certification and Welding Inspection.

I consider these issues to be unresolved by TVA and because these issues potentially adversely affect the safety of the plant, startup should be delayed until TVA has adequately addressed both issues. TVA's denial of documented facts by NSRS/QTC/ERT should be flatly rejected; additional test and evaluation is required to give reasonable assurance that the plant is safe for operation.

I recommend the following additions to the TVA Weld Evaluation Program (WEP) as a MINIMUM requirement for assuring acceptable Safety Related Welds at SQN:

With documented programmatic and implementation failures in visual weld inspection and a high probability of unqualified NDE inspectors at the time of construction and PSI inspections, before restart of SQN weld quality in Class 1 and Class 2 piping welds should be verified by the following:

(1) For inspectors who served at SQN during construction and/or in the PSI but are no longer employed by TVA, make an independent audit of personnel records to determine compliance with SNT-TC-1A recommended practice for NDE Level II Certification of inspectors.

(2) For inspectors still employed by TVA, test each NDE inspector using hands-on weld samples (EG&G is using such samples in their current WEP inspector-qualification program; such testing will give no assurance that the inspector was qualified at the time of construction or PSI but will at least verify that the inspector is currently qualified).

(3) Verify the quality of the radiography as used in the RT of safety related welds by re-examination of film by an independent team of radiographic experts. The assessment of quality should include determining if the films have the required identification, are free of artifacts, have the correct penetrameters and quality level and have correct station markers. Additionally, the verification examination should reassess the weld quality and related documentation to assure that all indications were properly interpreted and that all rejectable indications were corrected.

Sincerely yours,

Carl E Hartbower

SER/TER ATTACHMENT

September 15, 1986

MEMOORANDUM FOR: C. J. Czajkowski
Department of Nuc Energy
Brookhaven National Lab.
Upton, Long Island 11973

FROM: C. E. Hartbower
Consulting Welding Engineer
Fair Oaks, CA 95628

SUBJECT: REVIEW AND EVALUATION OF TVA's REASSESSMENT OF WELD QUALITY AT SEQUOYAH UNITS 1 & 2, EXECUTIVE SUMMARY OF FINDINGS.

STATEMENT OF PURPOSE

The Welding Team was under contract to Brookhaven National Laboratory to independently review TVA's resolution of the issues raised by the numerous welding-related employee concerns at the Sequoyah Nuclear Power Station (SQN) and to make recommendations on the adequacy of TVA's corrective actions, as appropriate.

As a team, our collective findings are encapsulated by Mr. Czajkowski in his Technical Evaluation Report (TER) under the heading EXPERT WELDING TEAM as applied to each of six categories of employee concerns. The findings do not always represent a consensus opinion. I see two issue categories as requiring amplification and further resolution prior to restart of SQN Units 1 & 2; viz., WELDING INSPECTION (40 employee concerns) and WELDER CERTIFICATION/TRAINING (27 employee concerns).

The following memorandum report constitutes an Executive Summary of my findings acting independently as a welding expert.

WELDER CERTIFICATION

Several reports address the employee concerns on matters relating to welder certification/training, including ERT Investigation Report of 9/26/85, QTC/ERT Investigation Report of 2/28/86 and NSRS Report I-85-135-SQN. The following findings may impact the safety of SQN and in my opinion require further consideration by TVA before restart of SQN. Quoting from the NSRS report:

> In the past, Nuclear Power has accepted construction welder performance qualification without retesting.
>
> The SQN Site Director issued a memorandum (Abercrombie to listed recipients, Aug 29, 1985, subject Welders Certification) directing site management to discontinue the practice (of accepting construction welder performance qualification without testing).

COMMENT: Construction was completed at SQN prior to Aug 1985 and, therefore, the corrective action was too late to benefit this plant.

> There appears to be no safety concern since all active welder records were either correct or readily restored to requirements. Also all safety-related welding is in-

dependently inspected per an approved QA program.

COMMENT: Safety may in fact be a concern because construction was completed prior to implementation of a proper welder certification program, and the efficacy of NDE inspection may have been limited as a means to verify the quality of safety-related welds by a programmatic breakdown and falsification of records within the TVA NDE training/certification program.

If one or more unqualified welders worked on safety-relatd welds SRWs), say Cl 1 and/or Cl 2 piping, and if the particular welds made by an unqualified welder were tested by an unqualified NDE inspector, potentially dangerous flaws could be in the plant today.

NRC welding team inspectors (6/2-6/6/86, 6/16-6/20/86 and 7/7-7/11/86 at SQN) confirmed a number of weld deficiencies that had been previously identified and evaluated by TVA in their WEP reinspection effort. Many of these weld deficiencies provide additional evidence of unqualified welders at SQN. Furthermore, the fact that on reinspection there were NCIG-01 rejectable welds inspite of the relaxed acceptance criteria of NCIG-01, welds that in construction had been made to AWS D1.1 acceptance criteria, provides additional evidence of unqualified welders at SQN (as well as unqualified inspectors during construction and PSI).

When the provisions of the TVA QA program that required welder performance qualification testing were relaxed and the work force perceived a loosening of control/standards, workmanship could have suffered. Welder skills, performance and pride in workmanship constitute the first line of defense against flawed welds; the second is the welding inspector who observes the day-by-day performance of each welder. If the welding inspector is to be effective, he or she must be adequately trained/experienced and operate with the full support of management.

WELDING INSPECTION

I believe that before restart of SQN TVA should resolve point by point the findings of three NSRS/QTC/ERT Investigation Reports which concluded that certain employee concerns regarding weld inspection are substantiated.

A Nuclear Safety Review Staff (NSRS) investigation was conducted to determine the validity of an expressed employee concern which stated:

> Sequoyah. Many employees are certified but are not qualified. They do not have enough on-the-job training (OJT) even though it is documented that they do have enough OJT. The concern existed from 1980 to present. Details known to QTC, withheld to maintain confidentiality. NUC PR concern.

NSRS notes that "...it should be recognized that a differentiation can be made between work-time experience, which is what OJT as used in this report is really referring to, and the proper usage of the term OJT which denotes a dedicated, organized, comprehensive and documented system of formal training on actuual work activity and equipment.

The report (NSRS Investigation Report I-85-373-NPS on Documentation of Required OJT for NDE Personnel Certification by C. L. Wilson and M. P. Mills dated 1/31/86, 27 pages) determined that

> NDE management in TVA early on took a very loose interpretation of OJT requirements, and many of the individuals who trained under that policy and were subsequently promoted have continued and extended that practice... A followup investigation by NUC PR will be required to remedy the findings documented herein.
>
> Inspection personnel in both QC and ISI have been placed in a difficult position by a policy which has been originated and promulgated by individuals who are now more than two levels of supervision above them...
>
> It is crucial to understand that there is a direct connection between the personnel practices of the NDE groups and the safety of the plant. This is because the inspectors can only do their critical jobs well when they see that strict completion of technical training requirements, independence and rigorous adherence to procedures are cultivated and rewarded rather than compomised...
>
> ...sufficient certification discrepancies were noted to mandate an extended evaluation by NUC PR of the TVA NDE certification program and resultant inspection activities. ... This followup investigation should begin with Sequoyah Nuclear Plant. NSRS considers this a startup issue for SQN.

Another report, an investigation of nondestructive examination (NDE) certifications at the Sequoyah Nuclear Plant (SQN) and the Power Operations Training Center (POTC), was written at the same time as the NSRS report. This investigation, by QTC, was to determine if NDE certifications had been falsified (QTC/ERT Investigation Report by M. P. Mills on Falsification of OJT Records dated 1/31/86, 7pages).

> The results of this investigation clearly indicate both a programmatic breakdown and falsification of records within the TVA NDE training/certification program. Based on these findings, the following is recommended:
> 1. The turn over of this report to the Office of General Counsel for investigation of legal wrong doing, and
> 2. TVA issue an immediate stop-work order against the certification of NDE inspectors until such time as the situation can be evaluated and corrective action taken.

The third report, QTC/ERT Investigation Report by R. W. Jones dated 2/28/86, 28 pages, dealt with a number of generic concerns that :

> Inspectors are generally untrained or not adequately trained, are unqualified, lack knowledge of weld-acceptance criteria and do not follow procedures.
>
> Training, both classroom and on-the-job is inadequate, certification tests are described as a joke, do not receive the required minimum training. The above concerns encompass all inspector training, qualifications and testing.

> This investigation was performed from July through October 1985.

The generic concerns listed above were substantiated. The finding that inspector training was inadequate, both in the classroom and on-the-job, is most damaging in the case of ultrasonic testing; it is common knowledge that this discipline requires special training and hands-on testing to assure qualified personnel.

Because there are SUBSTANTIATED employee concerns with regard to welding inspection and inspector training/qualification, I believe that all TVA inspections are suspect including visual, NDE MT, PT and RT and, inparticular, the ultrasonic testing (UT) done in construction and in preservice inspection (PSI). For UT inspectors, there is no way that work-time experience can be substituted for the "dedicated, organized, comprehensive and documented system of formal training" which the Nuclear Safety Review Staff found lacking in the TVA program.

APTECH reported in their "Evaluation of the Welds at SQN" AES 8511598AQ-1, Jan 1986, that

> Slightly under 10% of the field welds were inspected (in the PSI) by penetrant (PT). The remaining 90% were inspected ultrasonically (UT), which is a more rigorous volumetric examination than PT, which is primarily a surface examination. The lack of significant numbers of NOIs from the PSI is a strong indicator that the quality of the welds is high.

COMMENT: If TVA UT inspections are suspect, then it is a fallacy to assume that the lack of significant numbers of NOIs is an indicator of high quality in the SQN welds.

Likewise, in the TVA WEP reinspection of 333 piping welds and 1394 structural welds, I take little comfort in the low <u>rejection</u> rate (discounting weld spatter and arc strikes) considering

(1) the high <u>indication</u> rate in Office of Construction (OC) piping welds by visual examination (a 38 to 68% indication rate) see SER Table 1A, p10,

(2) the OC piping welds would have been rejected if the indications had been detected during construction (ASME Section III vs Section XI),

(3) the reportable indications in OC structural welds should have been detected during construction.

(4) at the time of construction, the applicable code for the structural welds was AWS D1.1-72; for purposes of WEP reinspection the accept/reject criteria were based on NCIG-01 which in some respects is much less stringent the AWS D1.1, therefore, if the visual inspectors had been well trained and diligent, TVA at the time of construction should have had a significantly higher rejection rate than found in the WEP reinspection based on NCIG-01.

There is additional evidence of unqualified, unmotivated inspections. There was a high incidence of weld spatter and arc strikes discovered in the WEP reinspections. During construction, at various times TVA Specifications (P.S.3.C.5.2. and P.S.3.C.5.4 after 2/13/81) called for removal of spatter and arc strikes. Furthermore, wherever there was to be UT inspection, removal of weld spatter and arc strikes should have been routine in preparation for inspection. There can be little doubt that much of the spatter and arc strikes occurred during construction and yet inspection did not call for its removal. There is another consideration, viz., I question whether the cracking that sometimes attends an arc strike can effectively/reliably be detected without first grinding the arc strike smooth and flush. Wherever this was not done and followed by PT inspection, there is a possibility of undetected cracking.

With evidence that TVA failed to provide consistently reliable welding inspection during construction and in the PSI inspection, all safety related welds are subject to question except those verified by NRC welding team WEP reinspections. The NRC welding team noted a number of weld discrepancies, most of which were previously identified and evaluated as a result of the TVA WEP reinspection effort. This verification of TVA findings by NRC indicates that TVA inspection was effective in identifying weld deficiencies in the recent WEP reinspections but confirmed ineptness in earlier inspections. Futhermore, the earlier volumetric inspections by TVA that are suspect were omitted from the WEP reinspection program.

Some of the additional irregularities not found in the TVA WEP reinspections but found by the NRC welding team raise additinal questions about the qualifications of the TVA inspectors/inspection program during construction. A number of welds were found by the NRC welding team which deviated from the requirements of the applicable design drawings; TVA inspection should have found these discrepancies in construction or in PSI. One structural platform was inspected by NRC; the TVA inspectors during construction could not have verified conformance to design because there were no weld details on the desgn drawings. Also the NRC welding team found that in some cases the drawing/Specifications failed to specify the Quality Level for inspection; thus, during construction the TVA inspectors had no way to know whether Quality Level 1 or 2 was intended by the designer. These discrepancies are not insignificant and should have been discovered by TVA inspection during construction, in PSI or in the WEP reinspection.

Carl E. Hartbower

DOTY & ASSOCIATES, INC.

ENGINEERING CONSULTANTS

Attachment 15

W. D. DOTY, PH.D., P.E.
PRINCIPAL CONSULTANT
(412) 881-3377

BOX 98243
PITTSBURGH, PA 15227

December 5,1986

Mr. Carl J. Czajkowski
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, Long Island, New York 11973

Subject: Contract Number 225772-S
- W. D. Doty Position Statement

The writer participated as a member of a team of welding experts to assist Brookhaven and the Nuclear Regulatory Commission in evaluating welding concerns at the Sequoyah Nuclear Power Station(SQN). A "Technical Evaluation Report(TER) Related to the Welding Concern Program at TVA'S Sequoyah Units 1 and 2", dated August 1986,was prepared by Carl J.Czajkowski. This report properly reflects the concensus of the team of welding experts.

In addition to the above mentioned report, I should like to offer the following comments:

1- An on-site visit to SQN was made on April 14-16,1986. It was the writer opinion that the weld quality, as judge by visual examination (without weld sizing), was good.

2- The fact that SQN has been operating for six years without significant weld failures supports only the adequacy of the welding for this six year period of reported "normal" operation.

3- Use of NCIG-01 as a relaxation of the D1.1 Code represents a technically acceptable approach for visual inspection of structural weldments of nuclear power plants provide the relaxation is fully justified by an engineering analysis. If the analysis shows that the original design was unjustifiably conservative, the suitability of the structural weldment should not be rejected out-of-hand because the D1.1 visual inspection requirements were not met. The results of such an engineering reevaluation should be a major factor in any decision relative to the start-up of the SQN units.

W D Doty

W. D. Doty

Attachment 16

BROOKHAVEN NATIONAL LABORATORY

ASSOCIATED UNIVERSITIES, INC

Upton Long Island New York 11973

(516) 282
FTS 666 4420

Department of Nuclear Energy

November 20, 1987

Mr. David Smith
Engineering Branch
Office of Special Projects
TVA Project
Mail Stop EWW-325
Washington, DC 20555

Dear Mr. Smith:

Per your request, I have performed a comparison of the TVA "Welding Project Generic Employee Concern Evaluation Reports" and the BNL Technical Evaluation Report (TER) for the Sequoyah Units. The following is a list of employee concerns which appear in the TVA Evaluation Reports but do not appear in the BNL TER:

1. IN-86-019-001 - Inspection through paint
2. IN-85-815-001 - Welder performance qualifications
3. EX-85-008-001 - Welder training
4. WBP-6-007-001 - Box anchor design
5. WI-85-030-007 - Preweld inspections
6. BFM-5-001-001 - Preweld inspections

The previous six concerns although not specifically analyzed by the Expert Welding Team (by number) were all reviewed during the TER evaluation of similar concerns from the same general categories.

7. IN-85-339-005 - Duct Installation

The omission of this concern from the TER was merely a typographical error on BNL's part.

8. JHL-85-003 - Inadequate weld procedure

This concern appears to be more applicable to a QA document control category rather than welding. As such, it does not fall into the original charter of the e)ert welding team.

9. XX- ,-124-001 - Control of Unused Weld Material

This concern described the burial of unused electrode. As such the burial effectively removed the electrode from possible use, eliminating potential problems. This concern was not specifically evaluated by the team, however, although the method of disposition was somewhat novel (burial near a cemetery) the exercise of weld material control was evident.

10. JLH-85-002 - Welder Performance Qualification

This concern revolved about a welder transferred from the Muscle Shoals project who was welding on a nuclear unit with an insufficient number of bend tests performed for his performance qualification. The welder was retested and summarily passed. The evaluation by TVA of the problem and the corrective action to evaluate the potential for other welders similarly transferred to weld without adequate specimen testing is consistent with the suggestions made by the expert welding team for similar Welder Performance Qualification concerns.

11. DHT-85-001 - Weld Metal Substitution

This concern dealt with the substitution and interchangeability of E70S-3 and E70S-6 weld wire at the Sequoyah sites. The response from the utility appears inadequate based upon a BNL review of welding at the valve rooms of Watts Bar #2 (10/87). The attached draft excerpts from the BNL report outline the problem which appears to be generic to the TVA system.

The following is a list of concerns which were addressed in the BNL TER but which were not listed on the TVA documents. It appears that those were addressed generically in various WP's while TVA believed them not to be specific to the Sequoyah units:

1.	XX-85-088-X04	Correction fluid used on welder certifications
2.	-001	
3.	WI-85-030-008	Inspection through paint
4.	XX-85-069-002	NDE Certifications
5.	XX-85-069-006	NDE Certifications
	-X13	NDE Certifications
	-X07	NDE Certifications
6.	IN-85-001-005	Vendor welds
7.	XX-85-083-001	Poor welding inspection at SQN
8.	XX-85-086-002	Poor box hanger design
9.	IN-86-158-006	Weld material accountability
10.	IN-86-047-001	Lack of stub control-weld rods
11.	XX-85-010-001	SQN - nut to baseplate welding

If there are any questions, please contact me at the above listed number.

Very truly yours,

Carl J. Czajkowski

Carl J. Czajkowski

CJC:ts
Enclosure
cc/enclosure:
Expert Welding Team
M. Schuster
P. Soo

- No covers on permanent plant lighting fixtures.

- Water collected in beam pocket.

WELDING

Various piping welds were selected at random in both the north and south valve rooms. These weld numbers were then used to locate the inspection packages for the joint. From the inspection package, the welders identification was determined and then his qualification as well as the procedure qualification was verified. A total of 98 weld packages were reviewed which encompassed 28 welder qualifications.

Personnel Contacted:

H. L. Alsup

S. Bonez

K. Hastings

R. Josse

M. Presley

J. White

Inspection Findings:

During the inspection, it was noted that various welds on the main steam had been installed using E70S-6 filler metal for the weld root passes. TVA Process Specification 1.M.1.2 (R5) dated May 22, 1987, page 8 of 20, paragraph 7.10, states:

"When an electrode of the E70S-3 type is specified on the detail weld procedure, type E70S-6 may be substituted for use in applications not requiring impact testing. The E70S-6 shall have a certified chemical analysis of A Nuclear 1 of Section IX (.15 percent maximum carbon, 1.60 percent maximum manganese, and 1.00 percent maximum silicon). This substitution is not permitted in applications requiring impact testing."

Since paragraph 10.3.2.2 of the Watts Bar Preliminary Safety Analysis Report (PSAR) also states for the WBN main steam lines:

> "The materials for piping and fittings in the TVA Class B Portion of the system are impact tested to plus (+) 40°F, as required by ASME Section III for Class 2 components. The test temperature of plus (+) 40°F is related to a minimum service temperature of plus (+) 70°F (hydro test water temperature)."

it appeared that the welding of these lines was in violation of this specification.

The utility had discovered this procedure violation and issued a Problem Identification Report (PIR) # PIRBLNNEB8607 on 11/26/86. Part of the corrective action in this PIR was the requalification of the procedures used with impact testing. Towards this end, Welding Procedure Qualification Records #GT-SM11-0-2A, GT11-0-1, GT-SM11-0-3, and GT-SM11-0-3C were provided to BNL. Upon review of these procedures, the following conclusions can be drawn:

1. These welding procedure qualifications <u>do not</u> qualify for welding materials of P number 1, Group number 2 (ASME IX) to itself or to P number 1, Group number 1 materials. The following <u>inspected</u> welds would then be affected:

2-001A-D003-10 P1 Gr 2 to P1 Gr 2

2-001A-D003-09 P1 Gr 2 to P1 Gr 2

2-001A-D006-10 P1 Gr 2 to P1 Gr 2

2-001A-D006-11 P1 Gr 2 to P1 Gr 2

2-001A-D006-06 P1 Gr 2 to P1 Gr 2

2-003B-D003-03 P1 Gr 2 to P1 Gr 1

The procedure qualifications GT-SM11-0-2A and GT-SM11-0-3C were made using base material SA 516, Gr 70 (P number 1, Group #2) which the utility states also meets the requirements for SA 516, Gr 65 (P number 1, Group #1). This appears contradictory to ASME Section IX requirements:

"QW-403.5 Welding procedure qualifications for base metals which have specified impact test requirements shall be made using a base metal of the same type or grade or another base metal listed in the same group (see QW-422) as the base metal to be used in production welding. When joints are to be made between base metals from two different groups, a procedure qualification shall be made for the applicable combination of base metals, even though procedure qualification tests have been made for each of the two base metals welded to itself. If, however, the procedure specification for welding the combination of base metals specifies the same essential variables, including electrode or filler metal, as both specifications for welding each base metal to itself such that base metals is the only change, it shall not be necessary to make impact tests to qualify the two together.

When a procedure has been previously qualified to satisfy all requirements other than notch toughness, it is then necessary only to prepare an additional test coupon using the same procedure with the plates only long enough to provide the necessary notch toughness specimens. If a previously qualified weld procedure has satisfactory notch toughness values in the weld metal, then it is necessary only to test notch toughness specimens from the heat-affected zone."

These procedures would need to be requalified using materials from the appropriate P and Group headings.

2. The GT-SM 11-0-2A procedure only qualifies the GTAW portion in thickness range of 3/16"-3/8". What QC requirements will assure that these limitations are not exceeded on repairs/new welds?

3. The same restrictions as 1 above would apply to the use of these procedures on the repair welds in Unit 1 (CAQR WBP 871081 dated 10/26/87).

DOCUMENTS REVIEWED

Detail Weld Procedures

Welding Procedure Qualification Records

Welder Performance Qualification Records

Process Specifications

Welding Operations Sheets