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Safety Evaluation Report
on Tennessee Valley Authority: Browns Ferry Nuclear Performance Plan

Browns Ferry Unit 2 Restart

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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MASTER

ABSTRACT

This safety evaluation report (SER) on the information submitted by the Tennessee Valley Authority (TVA) in its Nuclear Performance Plan, through Revision 2, for the Browns Ferry Nuclear Power Station and in 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 Unit 2. The staff will inspect implementation of those programs. Where systems are common to Units 1 and 2 or to Units 2 and 3, the staff safety evaluations of those systems are included herein. Future supplements to this SER will address open issues identified in Chapter 1.

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ABBREVIATIONS

ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASA	American Standards Association
ASME	American Society of Mechanical Engineers
BFNPP	Browns Ferry Nuclear Performance Plan
BTP	branch technical position
BWR	boiling-water reactor
CAQ	condition adverse to quality
CAQR	condition adverse to quality report
CAR	corrective action report
CCTS	Corporate Commitment Tracking System
CFR	Code of Federal Regulations
CI	concerned individual
CIP	Chemistry Improvement Plan
CNPP	Corporate Nuclear Performance Plan
CRC	corrosion resistance cladded
CRD	control rod drive
DBA	design-basis accident
DBVP	design baseline and verification program
DCA	design change authorization
DG	diesel generator
DNE	Division of Nuclear Engineering (TVA)
DPO	differing professional opinion
ECP	Employee Concern Program
ECP-SR	Employee Concern Program--site representative
EQDP	environmental qualification data package
EPIP	emergency plan implementing procedure
EQ	environmental qualification
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
GDC	general design criteria
GE	General Electric Company
GL	generic letter
HELB-IPC	high-energy line break inside the containment
HELB-OPC	high-energy line break outside the containment
HSW	heat sink welded
IE	Office of Inspection and Enforcement (NRC)
IEEE	Institute of Electrical and Electronics Engineers
IFI	inspector followup item

IGSCC	intergranular stress corrosion cracking
I&H	intimidation, harassment, and wrongdoing (employee concern)
IHSI	induction heating stress improvement
INPO	Institute of Nuclear Power Operations
IR	inspection report
IRG	Independent Review Group
ISI	inservice inspection
JTG	Joint Test Group
LER	licensee event report
LOCA	loss-of-coolant accident
LOP	loss of offsite power
MCC	motor control center
MOV	motor-operated valve
MSA	management self-assessment
NCR	nonconforming condition report
NFPA	National Fire Protection Association
NPRDS	nuclear plant reliability data system
NQAM	Nuclear Quality Assurance Manual
NRC	Nuclear Regulatory Commission
NSRB	Nuclear Safety Review Board
NSSS	nuclear steam supply system
NUMARC	Nuclear Management and Resources Council
OE	Office of Engineering (TVA)
OIG	Office of Inspector General (TVA)
ONP	Office of Nuclear Power (TVA)
PASS	postaccident sampling system
PIR	problem identification report
PORC	plant operations review committee
QA	quality assurance
QC	quality control
QMDS	qualification maintenance data sheet
QTC	Quality Technology Company
RC	radiological control
REP	radiological emergency preparedness
RER	regulatory effectiveness review
RG	regulatory guide
RHR	residual heat removal
RTP	restart test program
RWCU	reactor water cleanup
SALP	systematic assessment of licensee performance
SCL	system checklist
SCR	significant condition report
SDSP	Site Director Standard Practice
SE	safety evaluation
SER	safety evaluation report

SGTS	standby gas treatment system
SI	surveillance instruction
SMART	Senior Management Assessment of Readiness Team
SMPL	site master punchlist
SPL	system punchlist
TE	test exception
TMI	Three Mile Island
TOL	thermal overload
TRD	test requirements document
TS	technical specifications
TVA	Tennessee Valley Authority
UNID	unique identification
URI	unresolved item
UT	ultrasonic testing
VM	vendor manual
VMCP	vendor manual control program

1 INTRODUCTION

On September 17, 1985, the Executive Director for Operations of the Nuclear Regulatory Commission (NRC) issued a letter to the Chairman of the Board of Directors of the Tennessee Valley Authority (TVA or the licensee) pursuant to Title 10 of the Code of Federal Regulations Section 50.54(f) [10 CFR 50.54(f)] requesting information on the actions the licensee 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, (3) the Browns Ferry Nuclear Plant, and (4) the Watts Bar Nuclear Plant. A summary of the concerns raised in the staff's 10 CFR 50.54(f) letter and their status is contained in Appendix C.

TVA's Corporate Nuclear Performance Plan (CNPP), which was prepared in response to the NRC letter, was submitted to the NRC on November 1, 1985. (See Table 1.1 for issue dates of Volume 1 and its revisions.) The NRC staff safety evaluation of the revised CNPP, through Revision 4, was issued in July 1987 as NUREG-1232, Volume 1, "Safety Evaluation Report on Tennessee Valley Authority."

In addition to its corporate plan, TVA prepared 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 listed in Volume 3, Browns Ferry Nuclear Performance Plan (BFNPP) (Rev. 2), specifically for Unit 2 restart. (See Table 1.1 for issue dates of Volume 3 and its revisions.) In many cases, long-term corrective actions, extending beyond startup, are required to fully resolve these issues.

Regulatory performance at Browns Ferry had declined over the years preceding submittal of the BFNPP. Evaluations by the TVA staff, outside contractors engaged by TVA, and the NRC staff have pointed out specific deficiencies in plant performance; but TVA has not always identified and corrected the root causes of these deficiencies.

The root causes and corrective actions taken at the TVA corporate level are described in Volume 1, Revision 5 of the CNPP. The actions include: hiring, developing, and retaining experienced nuclear managers; restructuring the nuclear organization to clarify lines of authority and responsibility and to provide centralized direction and control of nuclear activities; taking steps to restore employee trust in nuclear management; increasing upper management awareness of and involvement in nuclear activities; and improving the nuclear management systems and controls, the nuclear corrective action program, and other programmatic areas of operation, maintenance, welding, design change, and plant modification.

This study of root causes and corrective actions extends to Browns Ferry operations. Corrective initiatives started at the corporate level have been implemented through the Browns Ferry site director as well as through offsite organizations responsible for direct support. These improvements included organizational changes compatible with restructuring of TVA's nuclear power organization, improved management control and involvement, revised conduct of operations and maintenance activities, improved quality awareness, centralized design control, a long-term program for upgrading procedures, and programs to restore employee confidence.

Review of the problems and issues identified at Browns Ferry resulted in TVA determining that the difficulties at that plant have stemmed from three primary causes (BFNPP, Section I.4.0):

- lack of clear assignment of responsibility and authority to managers and their organizations that clearly established accountability for performance

- insufficient management involvement and control in the workplace leading to a failure to adequately establish highest quality

- failure to maintain consistently a documented design basis for the plant and to control consistently the plant's configuration with that basis

Specific functional areas of plant activities that require strengthening on a long-term continuing basis involve operations, maintenance, surveillance, radiological controls, chemistry, security, emergency preparedness, and site scheduling. These cover the functional areas normally reviewed in either Institute of Nuclear Power Operations evaluations or systematic assessment of licensee performance reports.

Special programs were defined in a number of areas to ensure integrated corrective actions dealing with problems created by deficiencies in the past conduct of activities. The following special programs were identified as requiring resolution before restart (BFNPP, Sections II and III):

(1) Establish a program for environmental qualification of safety-related electrical equipment.

(2) Establish and maintain a documented design basis.

(3) Review suspended components for structural adequacy during a seismic design-basis event.

(4) Review electrical, mechanical, nuclear, and civil design calculations for adequacy.

(5) Review fire protection with respect to current NRC and general industrial requirements and recommendations.

(6) Review past welding practices and installed welds for adequacy.

(7) Review the current condition of the primary system pressure boundary and other structural components for adequacy relative to intergranular stress corrosion cracking.

(8) Establish coordinated restart test and operational readiness programs.

(9) Review installations of safety-related instrument-sensing lines for slope, separation, material control, fabrication, and quality assurance.

(10) Inspect suspect areas of piping to ensure that wall loss due to erosion and/or corrosion does not exceed allowable limits.

(11) Develop a summary document that describes changes made in the Browns Ferry probabilistic risk assessment (PRA) and the bases for concluding that the revised PRA conservatively reflects the Browns Ferry configuration.

(12) Review piece-part procurement to ensure that qualification of safety-related equipment is maintained.

(13) Review electrical installations to ensure functionality to mitigate design-basis events described in Final Safety Analysis Report (FSAR) Chapter 14 and provide for safe shutdown.

The programs mentioned above are evaluated in Chapters 2 through 4 of this report. They have been grouped into three sections: adequacy of design, special programs, and readiness for operation.

There are other programs as well to consider: Q-list program, moderate energy line break flooding, containment coatings, platform thermal growth, and heat code traceability. Many of these programs are applicable to Units 1, 2, and 3, although actual implementation for Units 1 and 3 will not be completed until after Unit 2 restart.

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 described in Chapter 5 of this report.

The NRC's evaluation of allegations in accordance with established NRC policies for allegations is discussed in Chapter 6 of this report.

At this time, a number of issues required for restart are still unresolved. Resolution of these issues will be discussed in one or more supplements to this report or in inspection reports to be issued before restart. The purpose of this report is as follows: (1) document the resolution of all restart issues that have been resolved to date (in some cases the programmatic aspects are addressed in this report and staff evaluation of implementation will be the subject of an inspection report) and (2) identify actions that are necessary to resolve all currently open restart issues. Several sections of this SER contain the staff's evaluation of programs as described in the Nuclear Performance Plan which have undergone revision subsequent to the staff's review. The staff will review future revisions to TVA's programs and issue a revised evaluation in a future supplement to this SER as necessary.

Issues identified in this report that cannot be resolved until the licensee provides additional information and the section of the SER that addresses the issue are:

Section	Issue
3.2	The 10 CFR 50.49 equipment qualification (EQ) list
3.11.1	Thermal overload heaters

The following is a list of sections that are not included in this SER because (1) the staff has not yet completed its review or (2) revisions to the licensee's program since the staff's review have made the staff's evaluation inapplicable. The staff's evaluation of these TVA programs will be included in a supplement to this SER:

Section	Title
2.2	Seismic Design Issues
2.3	Heat Code Traceability
2.4	Platform Thermal Growth
3.3	Piece-Part Qualification
3.4	Instrument Sensing Line Issues
3.5	Welding
3.7	Containment Coatings
3.8	Moderate-Energy Line Breaks
3.9	Probabilistic Risk Assessment
3.11.3	Ampacity
3.11.4	Cable Installation Including Cable Separation
3.12	Flexible Conduit
3.13	Cable Splices
4.1	Operational Readiness Review Program
4.2	Management
4.5	Maintenance
4.7	Training
6	Allegations

David H. Moran of the TVA Projects Division of NRC's Office of Nuclear Reactor Regulation coordinated the staff's efforts involved in preparing the safety evaluations for TVA's Browns Ferry Unit 2 restart efforts. Mr. Moran may be contacted by telephone at (301) 492-7000 or by writing to the following address:

Mr. David H. Moran
TVA Projects Division
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Table 1.1 Issue dates of Tennessee Valley Authority Nuclear Performance Plan and revisions

Publication	Date of issue
Volume 1: Corporate Nuclear Performance Plan (CNPP)	
Original	November 1, 1985
Revised (original)	March 10, 1986
Revision 1	July 17, 1986
Revision 2	July 31, 1986
Revision 3	December 4, 1986
Revision 4	March 26, 1987
Revision 5	December 10, 1987
Volume 3: Browns Ferry Nuclear Performance Plan (BFNPP)	
Original (Revision 0)	August 28, 1986
Revision 1	July 1, 1987
Revision 2	October 24, 1988

2 ADEQUACY OF DESIGN

2.1 Configuration Management Program, Design Baseline and Verification Program, and Design Calculations Program

2.1.1 Introduction

A TVA assessment team consisting of design engineers not associated with the Browns Ferry nuclear project and an independent contractor (M. Bender) reviewed the design process at Browns Ferry. In September 1985, this team issued a report (the Bender report) which concluded:

(1) No design policy currently exists or is in the planning stage that would provide a reference basis for deciding how to judge the design requirements for the Browns Ferry plant in view of the fact that the plant was designed at a time when formalized design criteria were not in vogue and the regulatory system that governed new plant design was not in place, and thus, no explicit set of baseline design documents exists.

(2) Design-related activities not controlled by the TVA Office of Engineering (OE) that may infringe on design integrity assurance are not correlated with the OE design control policy being established for the Browns Ferry Nuclear Plant.

The Browns Ferry design baseline and verification program (DBVP) was established to resolve the identified problems related to design control that had occurred at Browns Ferry. These identified problems are summarized as follows (BFNPP, Section III.2.0):

(1) The original design control program allowed an as-built set of drawings to be maintained by plant operations personnel and an as-designed set of drawings to be maintained by engineering personnel.

(2) The plant configuration was not reconciled with the design basis because the plant design basis was scattered among many documents, thus making it not readily usable.

(3) External reviews and studies indicated weaknesses in plant modifications that had been implemented after the plant became operational.

TVA submitted Revision 0 to the Browns Ferry DBVP to the NRC by letter dated March 13, 1987. In a letter dated July 10, 1987, TVA submitted a more detailed version (Rev. 2) of the DBVP that upgraded the program to (1) reconcile design control issues, (2) reestablish the design basis, and (3) evaluate the plant configuration. Revisions 1 and 3 were internal documents not submitted to the NRC. In a letter dated March 25, 1988, TVA submitted to the NRC Revision 4 of the Browns Ferry DBVP which incorporated the DBVP calculational effort.

Program Description

The objectives of the DBVP are to reestablish the design basis and evaluate the plant configuration to ensure that: (1) it satisfies the design basis by verifying the functional adequacy of the plant configuration, (2) the configuration of these systems is supported by engineering analysis and documentation, and (3) confidence exists that the plant configuration is in conformance with licensing commitments.

The essential elements of the overall program are:

(1) verification of plant configuration

(2) reconcilation of the configuration to engineering design documents, including essential calculations

(3) reconciliation of the configuration to the Browns Ferry Final Safety Analysis Report (FSAR) and licensing commitments

(4) performance of system evaluation for the system configuration

(5) issuance of revised key plant drawings for the required systems to be consistent with the plant configuration

(6) implementation of improved design change control

TVA is implementing the Browns Ferry DBVP in two phases: Phase I will be completed before startup and will include the evaluation of systems and portions of systems required for safe shutdown. These systems will be identified by evaluating the abnormal operational transients, design-basis accidents, and special events addressed in Chapter 14 of the Browns Ferry FSAR and by determining the safety functions necessary to mitigate these events. Phase II will be completed after startup and will include implementation of the remaining modifications of systems not required for startup, completion and revision of the design criteria documentation, completion of system evaluations, and implementation of corrective actions to other systems as required.

An improved design change control process will be put into effect to ensure that compliance with the design basis continues.

TVA identified the systems required to accomplish a safe shutdown. A staff review of these required systems revealed that the containment purge valve capability to open against a 30-psig containment pressure was assigned a Phase II priority. The staff has concluded that this item should be assigned a Phase I priority because of the importance of purging the containment atmosphere 30 days into the accident sequence in order to expel the combustible gases that build up in the containment. TVA has provided vendor procurement data and walkdown data which indicate that the installed valves are mechanically capable of opening against the containment pressure. TVA has also demonstrated that electric power can be supplied to the purge valve operator by various methods involving minor repairs that are easily completed within the 30-day period for opening the containment purge valve. On the basis of its review of the containment purge valve operability and the list of safe-shutdown systems, the staff concludes that the Browns Ferry DBVP has identified the systems required for safe shutdown of the plant following design-basis accidents.

2.1.2 Evaluation

From October 26 through October 30, 1987, an NRC inspection team reviewed and assessed the adequacy of the information contained in the Browns Ferry DBVP up to and including Revision 2. The NRC team found that TVA's DBVP contained the essential elements needed to achieve its goals and objectives; however, several weaknesses were identified which required resolution and the team asked TVA to address them. The extent, scope, and findings of this NRC team inspection are documented in NRC Inspection Report 50-259, 260, 296/87-36 dated January 21, 1988.

From April 18 through April 22, 1988, an NRC inspection team reviewed and assessed the adequacy of the Browns Ferry DBVP which incorporated the DBVP calculational effort as described in Revision 4 of the DBVP. The NRC team found that Revision 4 of the Browns Ferry DBVP incorporated the required DBVP calculation effort and, in general, did not contain other significant technical changes. Therefore, the conclusions reached earlier that TVA's DBVP contained the essential elements needed to achieve its goals and objectives were found still valid. The team also reviewed TVA's response dated April 20, 1988 to the previous inspection report finding concerning communication and interaction between the DBVP and ongoing programs at Browns Ferry. TVA initiated a review effort to improve coordination and commmunication among the various special programs established for Browns Ferry as described in Section III of the BFNPP. As a result, TVA developed an output matrix of Browns Ferry programs that are needed for input into other Browns Ferry programs. The responsible TVA program managers use the matrix to improve coordination and interface requirements. The NRC team's review of the TVA response to the interfacing programs issue found that it adequately addressed the team's concerns. The extent, scope, and finding of the NRC team inspection are documented in NRC Inspection Report 50-259, 260, 296/88-07, dated September 8, 1988.

The staff is continuing its review of the implementation of the Browns Ferry DBVP and will provide its findings in future staff inspection reports.

2.1.3 Conclusions

The NRC staff finds that TVA has adequately identified the problems associated with design control and design control changes and has established an appropriate design basis and verification program to reestablish the design basis and to evaluate the plant configuration to ensure its conformance to the plant design basis. The NRC staff concludes that the DBVP, if properly implemented, will ensure that the functional plant configuration is reflected in the plant design documents and drawings, and thus provides confidence that the systems required for safe shutdown of the plant can perform their safety-related functions in a satisfactory manner.

The staff also concludes that TVA has adequately addressed the issues identified in the Bender report since the Browns Ferry DBVP is intended to recapture the design baselines and configurations.

2.2 Seismic Design Issues (To be addressed in a supplement to this volume)

2.3 Heat Code Traceability (To be addressed in a supplement to this volume)

2.4 Platform Thermal Growth (To be addressed in a supplement to this volume)

3 SPECIAL PROGRAMS

3.1 Fire Protection

3.1.1 Introduction

Since the March 22, 1975 fire, the Tennessee Valley Authority (TVA or the licensee) has improved and added to the fire protection systems for all three units at Browns Ferry. The first set of improvements was installed as part of the Fire Recovery Plan that the licensee and the NRC staff had agreed upon. The NRC issued a restart safety evaluation report (NUREG-0061) in March 1976 on the Fire Recovery Plan.

After 10 CFR 50.48 and Appendix R to 10 CFR Part 50 were promulgated, the licensee developed a plan to implement the additional requirements imposed by these regulations. TVA submitted a report entitled "10 CFR 50 Appendix R Submittal Fire Protection and Safe Shutdown Systems Analyses Report for Browns Ferry Nuclear Plant, TVA" on January 31, 1986. Additional information was submitted on November 21, 1986. The staff issued a safety evaluation on the post-fire safe-shutdown analyses on December 8, 1988. This safety evaluation is being amended, however, because the licensee's safe-shutdown analysis has been revised. The licensee also requested 11 exemptions: 2 were withdrawn, 4 were changed to "Engineering Evaluations" in accordance with Generic Letter 86-10. The staff approved the remaining 5 exemptions on October 21, 1988. The staff is planning to inspect Browns Ferry for the Appendix R compliance before restart of Unit 2.

In addition to being in compliance with Appendix R, the licensee has committed in its Browns Ferry Nuclear Performance Plan (BFNPP) to improve organization and staffing in the area of fire protection, to comply with National Fire Protection Association (NFPA) standards, and to replace the Fire Recovery Plan with a new Fire Protection Plan. TVA submitted the Fire Protection Plan on April 4, 1988 as part of its Fire Protection Report.

3.1.2 Evaluation

A complete evaluation of Browns Ferry nuclear plant for compliance with Appendix R will require an onsite compliance inspection and a review of pertinent sections of the Fire Protection Report. The licensee has satisfactorily answered issues raised by the staff during the review of the January 31, 1986 submittal.

In regard to compliance with Appendix R, the following modifications are presently being implemented:

- fire detection (heat detectors and smoke detectors)
- fire suppression (automatic sprinklers)
- compartmentation (water curtains, wall and floor penetration seals, fire dampers, fire doors)

- circuit modifications (prevent spurious operation of residual heat removal [RHR] and reactor water cleanup [RWCU] valves and provide local control switches)

- cable modifications (cable wrapping and rerouting to gain separation)

- breaker and fuse upgrade for associated circuits

- addition of main steam relief valve backup air supply

- battery backup power supply for communication

- emergency lighting

In addition to the modifications presently planned, a long-term program has been developed to ensure continued compliance with Appendix R requirements.

Administrative improvements have included improvements in both organization and procedures. Since June 1985 there has been a Fire Protection Section on site currently staffed with a supervisor, a fire marshal, a fire protection engineer, a mechanical engineer, and appropriate engineering and craft personnel.

Since April 29, 1988, Browns Ferry has had a dedicated fire brigade assume the duties of fire fighting and testing and maintenance of fire protection systems.

As a result of these reorganizations, separate individuals have been assigned responsibilities for compliance, non-engineering aspects of the Fire Protection Program, and engineering aspects of the Fire Protection Program. Other organizational improvements include placing the fire protection function in the plant manager's organization and using the TVA Division of Nuclear Training to improve fire brigade training.

The licensee is also conducting a complete review of administrative procedures related to fire protection. All fire protection surveillance instructions are being reviewed in detail to ensure that they are technically accurate and to verify compliance with Technical Specifications; this review will be completed before restart of Unit 2. Those procedures, particularly those dealing with transient fire loads, have been rewritten.

The licensee's program to comply with NRC guidelines and NFPA standards consists of three steps:

- identification of deviations from NRC guidelines and standards
- evaluation of these deviations
- making appropriate modifications to bring systems into compliance with general industry practice as specified by NFPA

Where NRC guidance differs from NFPA codes or where no significant increase in fire protection would be achieved by the changes, actions specified in the last step above may be excepted. Otherwise all new Appendix R-related modifications to the fire protection systems are being installed in accordance with NFPA codes and NRC's Branch Technical Position (BTP) APCSB 9.5-1. The licensee has also completed a 2-year engineering study of the installed fire protection systems to identify deviations with respect to NFPA standards. The staff received a summary of deviations from NFPA codes dated August 3, 1988 and is reviewing it as part of the Appendix R compliance review and inspection. The staff provided TVA with the NRC position on the NFPA deviations by letter dated December 14, 1988.

The cable spreading room and the intake structure were toured on March 21, 1989 to resolve staff and licensee disagreements over allowable deviations in these areas. This issue will be resolved in a future supplement.

3.1.3 Conclusions

The licensee responded to NRC staff questions resulting from review of the first version of the Fire Protection Improvement Program (BFNPP, Rev. 0). NRC had previously noted deficiencies in the overall fire protection program. In Revision 2 of the BFNPP, the licensee summarized its overall fire protection goals in all areas of the site and discussed its philosophy of excellence for evaluating and addressing all known weaknesses in the Browns Ferry Fire Protection Program. The goals and objectives of the Fire Protection Improvement Program as outlined in the BFNPP are acceptable to the staff. The licensee has also provided a description of the duties and responsibilities of key members of the fire protection unit at Browns Ferry. The staff will inspect Browns Ferry for Appendix R compliance before restart.

3.2 Environmental Qualification of Electrical Equipment Program

3.2.1 Introduction

A licensee must demonstrate that equipment 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 Part 50) is applicable to equipment located inside as well as outside the 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"; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" (which supplements Institute of Electrical and Electronics Engineers [IEEE] Standard 323-1974 and various NRC regulatory guides and industry standards); and "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," prepared by NRC's Division of Operating Reactors and transmitted in a memorandum dated November 13, 1979, from H. Denton to V. Stello.

On August 8, 1985, the staff issued a safety evaluation for the Browns Ferry Nuclear Power Station, Units 1, 2, and 3 on the environmental qualification of safety-related electrical equipment. The staff concluded that the environmental qualification (EQ) program at Browns Ferry was in compliance with the requirements of 10 CFR 50.49 and that the issue of environmental qualification of electrical equipment important to safety was acceptably resolved. During July and August 1985, TVA, assisted by Westec Services, Inc., conducted a management review of EQ activities at Sequoyah, Browns Ferry, and Watts Bar. In this review, completed in August 1985, the licensee concluded that qualification documentation had not been established for the large majority of equipment reviewed. The deficiencies were judged to be significant, and both systematic and pervasive. The problems resulted from an unstructured program, lack of adequate guidance, and an inconsistent approach taken by the fragmented organizations involved in the EQ program. This review led TVA to shut down both units at Sequoyah on August 22, 1985. In a request for information pursuant to 10 CFR 50.54(f) dated September 17, 1985, the NRC included a specific item related to the Browns Ferry EQ program: "Provide a detailed description of (a) the program being implemented to demonstrate compliance with 10 CFR 50.49 and (b) the long-term program to assure continued compliance with regulations. Affirm that the list of equipment required to meet 10 CFR 50.49 is complete."

TVA responded on August 28, 1986, with the BFNPP. Section III.1 of the BFNPP specifically addressed this concern. On October 3, 1986, the NRC requested additional information on the BFNPP; some questions were related to the EQ program. TVA responded with Revision 1 to the BFNPP on July 16, 1987. TVA has yet to affirm that the 10 CFR 50.49 list of equipment is complete. TVA is also expected to certify, before restart, that EQ requirements have been satisfied.

3.2.2 Evaluation

The staff evaluation of the electrical equipment qualification program at Browns Ferry is based on the results of: (1) TVA's compliance with the requirements of 10 CFR 50.49, (2) TVA's BFNPP, and (3) the staff's equipment qualification inspection on May 9-13, 1988. The evaluation included a complete review of the EQ program as described in the BFNPP and implementing program procedures/instructions. This complete review was performed because TVA had made significant changes to its EQ program since issuance of the NRC's August 8, 1985 safety evaluation of the program.

3.2.2.1 Compliance With 10 CFR 50.49

Licensees are required to maintain current a list of the equipment that must be qualified under 10 CFR 50.49. At Browns Ferry, a systems analysis was conducted to identify, for each Chapter 14 design-basis accident (DBA) of the Final Safety Analysis Report (FSAR), a list of those equipment items ("end devices") which must either operate or "stay as is" to ensure completion of safety-related functions as defined in 10 CFR 50.49 (including the TVA commitments to Regulatory Guide [RG] 1.97). This list contained the end-devices (pumps, valves, motors, etc.) which were essential for completing the safety action. A second, more extensive list was generated by researching drawings in order to determine the support equipment such as power supplies, cables, terminations, logic systems, control systems, and electrical distribution systems which are necessary to ensure completion of each end-device's safety-related function. The expanded list was reduced by a failure analysis which eliminated those components whose failure would not prevent achievement of the required safety action. This list was then further reduced by eliminating equipment that is located in a mild environment as defined in 10 CFR 50.49(c). One final attempt was made to eliminate those items located in an environment that becomes harsh for certain accidents but remains mild for other accidents and whose equipment is only required to contribute to the safety function during these "other" accidents.

The DBAs that were evaluated as part of the 10 CFR 50.49 list development include (1) loss-of-coolant accident, (2) high-energy line break inside the containment (which is the intermediate and small-break LOCA), and (3) high-energy line break outside the containment. These are abbreviated as the LOCA, HELB-IPC, and HELB-OPC, respectively.

The 10 CFR 50.49(b)(2) category of non-safety-related electrical equipment whose failure could keep safety-related equipment from performing its safety function was incorporated in the first expansion of the list. During this expansion, a detailed circuit analysis of the drawings was performed to determine the necessary ancillary devices needed to support the required operating mode of the end-device (e.g., a valve required to close and stay closed or a valve required to remain open).

The 10 CFR 50.49(b)(3) category of postaccident monitoring equipment was addressed during the review of instrumentation and control drawings. Instruments shown on control drawings with associated indicators in the control room were correlated with the licensee's submittals of April 30, 1984 and May 7, 1985, regarding compliance with RG 1.97. The staff transmitted its safety evaluation of Browns Ferry's compliance with the guidance of RG 1.97 to TVA on June 23, 1988. The evaluation documented seven variables for which TVA had not provided sufficient justification to exclude them from qualification with 10 CFR 50.49. TVA's August 23, 1988 response provided the necessary information for closing all open items on an acceptable schedule. All variables involved with open items are class 2 and, therefore, not restart items.

The staff found acceptable the TVA commitment to provide environmentally qualified instrumentation by Cycle 7 for Unit 2 for the core spray flow, low-pressure coolant injection flow, residual heat removal (RHR) system flow, and emergency ventilation damper position variables. The staff also finds acceptable the TVA commitment to provide upgraded instrumentation for Units 1 and 3 before their restart.

The above response from TVA provided new justification that instrumentation variables for RHR heat exchanger outlet temperature, cooling water temperature to engineered safety feature (ESF) system components and cooling water flow to ESF system components are not subject to upgrading the environmental qualification. The principal justification is that all variables mentioned above are Category 2 variables and TVA has provided instrumentation to monitor the RHR heat exchanger outlet temperature, cooling water temperature to ESF system components, and cooling water flow to ESF system components. The staff will reevaluate these three variables to determine the validity of TVA's new justification. Staff reevaluation of these three variables is not required for Unit 2 restart.

During the May 9-13, 1988, NRC EQ inspection, the staff identified one inspection followup item (IFI), 50-259, 260, 296/88-11-04, pertaining to specific outstanding items needing correction/clarification in documentation supporting the 10 CFR 50.49 list. This IFI is documented in NRC Inspection Report (IR) 50-259, 260, 296/88-11, dated September 1, 1988. Although the issues of this IFI are not considered significant relative to the 10 CFR 50.49 list process, the issues must be closed before restart.

On the basis of its evaluation and results of the May 1988 inspection, the staff finds that the methods used at Browns Ferry for identifying electrical equipment within the scope of paragraphs (b)(1), (b)(2), and (b)(3) of 10 CFR 50.49 are in accordance with the requirements of those paragraphs and therefore are acceptable.

However, TVA must finalize the 10 CFR 50.49 list and close the issues identified in the EQ IR relative to IFI 50-259, 260, 296/88-11-04 before restart.

3.2.2.2 Qualification Methodology and Documentation

Browns Ferry and TVA's qualification methodology is described in the licensee's Appendix C to DI-125.01, "Program Requirements for Environmental Qualification of Electrical Equipment in Harsh Environments" (DI-125.01 superseded the EQ Project Manual EQP-01 discussed in the BFNPP). As stated in Appendix C, the preferred method of qualification is defined as testing of an identical component under identical conditions or under similar conditions with supporting analysis. Appendix C requires justification for any exception to this method to be included in the qualification file. Detailed guidance is included in DI-125.01 regarding similarity analysis data, extrapolation data, interpolation, and other supporting analyses that would provide acceptable alternatives to the preferred method.

Browns Ferry's EQ program provides for the preparation of an environmental qualification data package (EQDP or EQ binder) for each equipment type to demonstrate that the equipment is environmentally qualified for its application and that design-basis safety functions can be accomplished. An equipment type refers to electrical equipment categorized by manufacturer and model(s) which is representative of all identical equipment in a plant area(s) potentially exposed to the same bounding environmental conditions during and after a design-basis accident (e.g., Rosemount electronic pressure transmitters, Model 1153 Series D, located inside the containment). All auditable documentation that supports environmental qualification for the equipment type is compiled and placed in the EQ binder or is referenced therein. Each EQ binder consists of:

- title page referring to the vendor and equipment types
- revision log
- table of contents
- Tab A--identification of equipment including the equipment type
- Tab B--checklist for evaluating EQ, including summary and conclusion
- Tab C--analyses and justification
- Tab D--qualification documents
- Tab E--miscellaneous documents and correspondence
- Tab F--field-verification data
- Tab G--qualification maintenance data sheets
- Tab H--vendor instruction manual
- Tab I--vendor drawing for equipment
- Tab J--evaluation of NRC circulars, notices, and bulletins, and vendor bulletins

The licensee learned through experience at the Sequoyah Nuclear Plant that the as-built condition of qualified equipment sometimes did not agree with documentation in the qualification binders and that installation and subsequent maintenance activities may invalidate qualification. To alleviate this problem at Browns Ferry, the EQ program includes field verification of environmentally qualified equipment. This field effort covers verification of previously installed equipment and verification following installation of equipment installed by modifications. Activities at Browns Ferry in the area of EQ maintenance are discussed in Section 3.2.2.3.

Although the May 1988 NRC inspection identified some IFIs and unresolved items (URIs) (see IR 50-259, 260, 296/88-11 dated September 1, 1988) in the implementation of the EQ program at Browns Ferry, the staff has determined that Browns Ferry has established a program with appropriate implementing procedures and controls to ensure that all electrical equipment within the scope of 10 CFR 50.49 is qualified to the requirements of 10 CFR 50.49. The licensee must resolve the IFIs and URIs of the May 1988 inspection before restart.

3.2.2.3 Maintenance

In order for a licensee to maintain the qualified status of equipment throughout the equipment's life in the plant, it is necessary to identify qualification maintenance requirements that must be met. At Browns Ferry, TVA has included identification of qualification maintenance requirements (Tab G of the EQDPs) as part of the documentation included in the qualification binders. This information is identified on qualification maintenance data sheets (QMDSs). The QMDSs define all required EQ maintenance requirements and describe qualified spare parts.

The QMDSs are provided to plant maintenance organizations which review all requirements to ensure that required maintenance can be performed (including required warehouse maintenance), that QMDS requirements are merged with other ongoing maintenance activities so that qualification is maintained, that replacement intervals and trending programs are developed, and that all QMDS maintenance is scheduled and performed. The EQ program at Browns Ferry further requires that any desired changes to essential qualification maintenance requirements must be coordinated with the EQ organization before implementation.

The program at Browns Ferry emphasizes that qualification maintenance activities do not drive or substitute for the total overall maintenance program. Qualification maintenance activities are only one part of the plant's overall maintenance program.

In addition to establishing the requirements for qualification maintenance, the EQ program requires that the maintenance status of all equipment within the scope of 10 CFR 50.49 be verified before restart. This includes reviewing past maintenance activities to ensure that they have not invalidated the qualified status of installed qualified equipment.

The staff determined that the licensee has established a program and procedures with adequate controls to ensure that equipment qualified to the requirements of 10 CFR 50.49 is maintained in a qualified status throughout its life in the plant.

3.2.3 Conclusions

On the basis of its evaluation, the staff concludes that the Browns Ferry electrical equipment qualification program for electrical equipment located in harsh environments complies with the requirements of 10 CFR 50.49. Full implementation of the program awaits completion of certain activities such as equipment replacement, modifications, engineering analysis, and documentation. The licensee has implemented a tracking system for these activities and is following their completion. In addition to these activities, the licensee is required to ensure that the following activities identified here are resolved before restart:

- Finalize the 10 CFR 50.49 list. As a confirmatory item, before restart the licensee is required to certify to the NRC that the 10 CFR 50.49 list is complete and all electrical equipment within the scope of 10 CFR 50.49 is qualified to the requirements of 10 CFR 50.49. The NRC staff will continue to monitor implementation of the Browns Ferry EQ program and procedures through future inspections.

- Close the issues related to the IFIs and URIs identified in EQ Inspection Report 50-259, 260, 296/88-11, dated September 1, 1988.

The staff has found acceptable the TVA plan and schedule for closing out RG 1.97 compliance issues. The staff will reevaluate TVA's resolution for three class 2 variables which are not restart items.

3.3 Piece-Part Qualification Program (To be addressed in a supplement to this volume)

3.4 Instrument Sensing Line Issues (To be addressed in a supplement to this volume)

3.5 Welding (To be addressed in a supplement to this volume)

3.6 Intergranular Stress Corrosion Cracking

3.6.1 Introduction

TVA has experienced intergranular stress corrosion cracking (IGSCC) in several components at each of the three units at Browns Ferry. IGSCC in the austenitic stainless steel piping systems for reactor coolant has been identified in boiling-water reactors (BWRs) for about 12 years. Extensive studies have identified the conditions conducive to IGSCC and mitigation methods have been developed for each of the conditions in austenitic steel piping systems. For other systems and components, the items affected tend not to be joined by welding, but are relatively small, discrete parts which can be replaced with materials not susceptible to IGSCC, or if they are weldments, they are relatively small in size or number and can be monitored and replaced/repaired as necessary.

3.6.2 Evaluation

IGSCC Status of Reactor Attached Piping and Safe-Ends

TVA has utilized a number of mitigation methods in treating the 180 welds in the austenitic stainless steel reactor coolant piping greater than 4 inches in diameter which is exposed to reactor coolant whose temperature exceeds 200°F up to the second isolation valve (refer to Generic Letters 84-11 and 84-01). Of the 180 weld joints, there remain 11 welds in non-resistant material which have had no mitigation actions. There are five untreated welds in non-resistant material within penetrations. TVA plans to either remove the welds by design or to overlay cladding on these welds on the inside surface which is exposed to reactor coolant. Although the austenitic stainless steel piping has been replaced with carbon steel piping, there are six welds in the core spray system where austenitic stainless steel fittings are still in use; these bimetallic or dissimilar metal joints must undergo a mitigation action. TVA has indicated that these particular joints will undergo induction heating stress improvement (IHSI) treatment by the conclusion of the next refueling outage.

TVA has not completed post-IHSI inspections of 71 welds that were IHSI treated. Because TVA had not performed sample expansion inspections under the terms stipulated in Generic Letter 84-11 and required in the staff's letter of March 26, 1986, the staff has concluded (in its December 8, 1988 safety evaluation of TVA's response to GL 84-11 and GL 88-01) that all remaining post-IHSI inspections be performed before restart. The staff stipulated that if any crack indications were found after IHSI in the 25-percent sample inspected, another 25 percent of the IHSI-treated welds that had exposure since original licensing should be inspected. By letter dated January 12, 1989, TVA committed to examine before restart the remaining welds that have not received post-IHSI inspection and to carry out the sample expansion process in accordance with guidance in GL 88-01. TVA has replaced recirculation inlet safe-ends on Browns Ferry Unit 2 because of IGSCC. In addition, TVA has arranged to install hydrogen water-chemistry treatment facilities for Browns Ferry Unit 2 and this may take place during mid-cycle or by the end of the next refueling outage.

Jet Pump Hold-Down Beams

In response to IE Bulletin 80-07, TVA performed the required inspections and reported the results by letter dated October 3, 1980. No cracks were found during these inspections. General Electric Company (GE), the vendor, made replacement beam assemblies from material that was more resistant to IGSCC. Since the failure mechanism for jet pump hold-down beams was determined to be IGSCC, TVA replaced all jet pump beam assemblies on Units 1 and 2 as documented in Inspection Report 50-260/84-16 dated June 4, 1984.

Shroud Access Cover Weld Cracking

The NRC issued Information Notice 88-03 on February 2, 1988, "Cracks in Shroud Support Access Hole Cover Welds." The notice alerted BWR licensees to the potential for cracking in these welds. TVA contracted with GE to perform a special ultrasonic inspection to determine if any of these welds were cracked. The inspections were performed as reported in Inspection Reports 50-259, 260, 296/88-06 and 88-15 dated June 14, 1988 and June 6, 1988, respectively. No crack indications were found. The mitigation action of hydrogen water chemistry for the stainless steel piping along with a well controlled water chemistry program may provide the necessary environment to control this IGSCC problem and other such occurrences.

Shroud Head Bolts

Cracked shroud head bolts have been observed in several BWR plants. GE identified the failure mechanism as IGSCC. GE inspected all 48 of these bolts using a special ultrasonic process developed for the specific geometry. TVA reported 13 cracked shroud head bolts.

Before restart, TVA plans to replace all of the cracked bolts with new bolts or with bolts borrowed from another unit. TVA also plans to establish a program for periodic reinspection of these shroud bolts.

Control Blade Cracking

GE has recently identified cracking in control blades due to stress corrosion. However, other factors, such as weld configuration and water chemistry, have been found to be of significance. Various influences, rates of occurrence, means of mitigating or eliminating the cause of the cracking, and the regulatory approach remain to be determined.

TVA is participating in the industry's effort to build an experience data base. A selected number of control blades from Unit 2 will be inspected before restart. The inspection will consist of an in-core remote visual examination of the upper portion of the control blades while they are in the fully inserted position. Participation in generating an industry data base will provide TVA with current and definitive information allowing TVA to make informed decisions in addressing the issue.

Control Rod Drive Collet Tube Cracking

Control rod drive (CRD) collet retainer tube cracking has been observed in BWRs since 1975. TVA has established a CRD rebuild maintenance program that requires periodic inspection of tubes. Tubes are examined by means of liquid penetrant according to the vendor's recommended criteria. TVA has demonstrated an acceptable approach in addressing this problem.

Residual Heat Removal Pump Wear Rings

All of the Browns Ferry Unit 2 residual heat removal (RHR) pumps have had their upper and lower impeller wear rings replaced with material originally specified. Also, two of the crosstie RHR pumps to Unit 2 will have their wear rings replaced before Unit 2 restart. The mitigative action of hydrogen water chemistry for the stainless steel piping along with a well controlled water chemistry may provide the necessary environment to control this IGSCC problem and other such occurrences.

3.6.3 Conclusions

By letter dated December 8, 1988, the staff sent its review of TVA's program for mitigation of IGSCC to TVA. TVA's letter dated January 12, 1989 committed to examining Unit 2 welds that did not receive post-IHSI inspection and to sample in accordance with the guidance in GL 88-01. The staff considers that the program discussed in Section III 7.0 of the BFNPP is acceptable for the restart of Browns Ferry Unit 2.

The staff conducted a GL 88-01 implementation inspection from January 30, 1989 to February 1, 1989. The findings of this inspection have been reported in Inspection Report 50-259, 260, 296/89-05 dated February 21, 1989.

3.7 Containment Coatings (To be addressed in a supplement to this volume)

3.8 Moderate-Energy Line Breaks (To be addressed in a supplement to this volume)

3.9 Probabilistic Risk Assessment (To be addressed in a supplement to this volume)

3.10 Thinning of Pipe Walls

3.10.1 Introduction

On December 9, 1986, Unit 2 of the Surry Power Station experienced a catastrophic failure of a main feedwater pipe caused by wall thinning due to erosion/corrosion of the carbon steel pipe wall. Erosion/corrosion is a form of flow-assisted corrosion. Although pipe failures resulting from erosion/corrosion have occurred in other carbon steel piping, particularly in two-phase piping systems, there have been no previously reported failures in large-diameter piping systems containing high-purity water (single-phase systems).

The basis for this evaluation is NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants." The bulletin requests (1) the code of construction for the piping systems susceptible to erosion/corrosion, (2) a description of the thickness measurement program, (3) the criteria for selecting inspection points, (4) a summary of the inspection results, and (5) a description of future plans.

3.10.2 Evaluation

The licensee answered these five requests in its response to Bulletin 87-01 on September 18, 1987 and in the BFNPP.

(1) Identify the codes or standards for piping design and fabrication

- The piping was designed and fabricated to the 1967 edition of American National Standards Association Standard B31.1

(2) Describe the scope and extent of your programs for ensuring that pipe wall thicknesses are not reduced below the minimum allowable thickness. Include in the description the criteria that you have established for selecting thickness measurement points, frequency of examination, inspection methods and repair/replacement decisions.

- The licensee's basis for selecting areas most susceptible to erosion/corrosion in dual-phase systems is based on EPRI Report NP-3944 entitled "Erosion-Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Program Guidelines." TVA's basis for selecting areas in single-phase systems is based on an EPRI report dated February 19, 1987. The systems selected for examination were the turbine piping, moisture separators, heater drains, steam extraction, feedwater/condensate, and emergency equipment cooling water.

- Procedures were submitted with the licensee's response that describe the scope and extent of the thickness measurement programs. Ultrasonic testing (UT) is used to measure wall thickness with supplemental assistance from visual examination. Procedure TS 09.01.01.14.02 dated March 6, 1984, "Inspection Program--Division of Nuclear Power--Steam/Water Erosion of Piping and Corrosion of Raw Water Carbon Steel Piping," describes the inspection program. Procedure N-UT-26, Revision 4, dated May 14, 1987, "Ultrasonic Examination for the Detection of I.D. Pitting, Erosion and Corrosion," describes the examination procedure in detail.

The licensee plans to use the inspection results for trending analyses. If trending indicates that the wall thickness of the component will approach the design minimum wall thickness before the next scheduled outage, the component will be replaced or repaired.

(3) For liquid-phase systems, state specifically whether the following factors have been considered in establishing your criteria for selecting points at which to monitor piping thickness:

(a) piping material (e.g., chromium content)

(b) piping configuration (e.g., fittings less than 10 pipe diameters apart)

(c) pH of water in the system (e.g., pH less than 10)

(d) system temperature (e.g., between 190°F and 500°F)

(e) fluid bulk velocity (e.g., greater than 10 f/s)

(f) oxygen content in the system (e.g., oxygen less than 50 ppb)

- The licensee stated that only plain carbon steel piping was inspected since small amounts of chromium significantly improve a material's resistance to single-phase flow erosion/corrosion as shown by Unit 2. Fittings less than 10 pipe diameters apart and piping immediately downstream of orifices and flow control valves are considered potential corrosion sites. Studies have shown that erosion/corrosion is more likely to occur in the 200°F to 350°F temperature range for single-phase flow. Locations within this range are inspected as well as areas up to 500°F if other criteria warrant. The fluid bulk velocity of the areas inspected generally exceeds 10 feet per second, however inspections were not limited to those areas.

  The licensee stated that the pH and oxygen are maintained at levels less than those necessary to increase resistance to erosion/corrosion. According to the FSAR, the pH may vary from 7.5 to 8.5. Since the pH and oxygen are assumed to be constant throughout the single-phase flow, they are not criteria for selecting examination points.

(4) Summarize the results of all inspections that were conducted for the purpose of identifying pipe wall thinning and any other inspections where pipe wall thinning was discovered.

(a) Describe the inspection program and indicate whether it was specifically intended to measure wall thickness or whether these measurements were incidental.

(b) Describe what piping was examined and how (e.g., describe the inspection instruments, test method, reference thickness, locations examined, means for locating measurement points in subsequent locations).

(c) Report thickness measurement results and note those that were identified as unacceptable and why.

(d) Describe actions already taken or planned for piping that has been found to have a nonconforming wall thickness. Include the results of any related failure analyses that have been performed. Indicate whether the actions involve repair or replacement, including any change of materials.

- <u>Browns Ferry Nuclear Plant Unit 1</u>

  - The turbine cross-under piping was inspected in 1977 and pits were found that were 60 to 80 mils deep. Some eroded areas were bright and others were covered with a dull, graying oxide which is associated with an actively corroding pit. An area of considerable wear was identified adjacent to moisture separator No. 3 and this and other areas were mapped for future inspections. The 1979 inspection showed additional degradation and straight lengths of pipe had the typical "tiger striping" pattern of pitting. The damage in the turbine exhaust area was completely random.

    High-velocity steam erosion caused the failure of a moisture separator drain pipe in 1982. Stainless steel was recommended as the replacement material.

  - The turbine cross-around piping was inspected in 1983 and there was widespread steam erosion damage. The majority of the corrosion sites were active. UT methods located one spot where the 0.625-in. pipe wall had been reduced to 0.400 in., but was thick enough for continued service. The licensee stated that the wall loss was proceeding at a constant rate, but the staff found that the data points would also justify a curve where the wall loss grew in proportion to the square of the number of hours of operation. The licensee's position should be reviewed at subsequent outages.

  - The miscellaneous drain headers were examined by UT in 1984 and and there was not any appreciable wall degradation. In this report, the licensee based the minimum acceptable wall thickness on the pipe diameter and internal pressure. The staff is of the opinion that this would give an unacceptably thin wall for drain pipes and there should be sufficient thickness to account for the accuracy of the ultrasonic test equipment and the pipe rigidity needed for mechanical loads. The licensee responded that degradation would be detected in the tracking program and corrective actions would be taken before the minimum wall thickness is reached.

  - In 1986, a small section was removed from a portion of pipe to verify UT results. The measured values were consistent with UT results. Tiger striping erosion/corrosion was observed.

- <u>Browns Ferry Nuclear Plant Unit 2</u>

  - The heater drain lines were examined in 1978 and showed the same erosion/corrosion as Unit 1, although not as deep because Unit 2 piping has a slightly higher alloy content. The 1979 inspection of the cross-under piping showed very little erosion/corrosion except in the No. 1 extraction piping and certain areas in the manway cover. Localized attack was seen in the 2B1 moisture separator.

- The 1982 inspection of moisture separators and associated piping showed minor steam erosion damage. In February 1983, the licensee inspected the 2B2 moisture separator drain piping and found erosion-type degradation in the 8-in. tee, 4 x 8-in. increaser, 8-in. pipe, and 8 x 16-in. increaser. Wall loss was measured by visual and UT methods and estimated to be 30 percent. Stainless steel replacement materials were recommended. This damage had not been observed in examinations of the 2A2 and 2C2 piping that has a higher alloy content.

- An examination of the 4-in. turbine exhaust piping in May 1983, showed the maximum wall loss to be 0.097 in. No wall thinning was observed on cross-over piping during a 1985 examination.

- Several reports were written in 1985 on the degradation of the extraction steam piping. UT examination showed wall losses ranging up to 60 percent in the No. 2 lines and up to 35 percent in the No. 1 lines. Calculations showed there was sufficient material remaining, but plans were made for temporary repair and replacements until better materials could be obtained.

As a result of NRC Bulletin 87-01, 32 areas were selected on the feedwater condensate piping for wall-thickness measurements using UT methods. Some minor cavitation damage was detected at the discharge of the main feed pump, but there was not any evidence of wall degradation.

• Browns Ferry Nuclear Plant Unit 3

The cross-around piping, extraction steam piping, and emergency equipment cooling water piping were examined for wall degradation in 1984. There were few localized areas of erosion/corrosion in the 42-in.-diameter cross-around piping and the moisture separators and they were not active corrosion sites. The wall thickness of the extraction steam piping from extractor No. 2 had been reduced from 0.375 in. nominal to 0.291 in. The minimum wall thickness of the first 12-in. line off the main line from extractor No. 2 was 0.301 in.; the surrounding area was 0.398 in. thick. In several unrelated spots, the wall thickness of the 18-in.-diameter emergency equipment cooling water piping had been reduced to 0.304 in. from 0.375 in. nominal. The 1986 inspection of the cross-over piping did not identify any wall thinning.

(5) Describe plans for revising present programs and developing new or additional programs for monitoring pipe wall thickness.

• The licensee furnished Sequoyah Nuclear Plant surveillance instructions and indicated that Browns Ferry would have similar plans. The inspection results will be compared with previous data and serve as the basis for replacement or continued operation of degraded pipe. The licensee plans to participate in the NUMARC (Nuclear Management and Resources Council) initiative regarding selection and inspection of piping for wall thinning. The licensee explained that these plans are not complete and will be modified on the basis of experience.

3.10.3 Conclusions

The staff reviewed the BFNPP and the licensee's response to NRC Bulletin 87-01. The licensee examined the systems most susceptible to erosion/corrosion degradation: turbine piping, moisture separators, heater drains, steam extraction, feedwater/condensate, and emergency equipment cooling water.

Within the exception of the Unit 3 emergency equipment cooling water piping that had localized wall-thickness reductions of 19 percent, the licensee has reduced the number of wall-thickness examinations to likely areas of vapor phase attack via cavitation, erosion, and erosion/corrosion. The only reported failure was the Unit 1 moisture separator drain pipe failure. Severely degraded areas were the Unit 1 turbine cross-around piping, Unit 2 moisture separators and extraction steam piping, and Unit 3 extraction steam piping.

Surveillance instructions have been written, and the licensee plans to monitor susceptible areas and to prepare a trending analysis. The locations of susceptible areas and frequency of inspection may change as experience is accumulated.

The minimum acceptable wall thickness is based on the minimum thickness to accommodate the internal pressure plus a corrosion allowance; the staff feels that additional thickness to account for the sensitivity and accuracy of the UT equipment and the pipe rigidity needed for mechanical loads should also be considered. The licensee makes the assumption that degradation due to erosion/corrosion will be linear with respect to time, but the staff noted that the data also support a conclusion that the degradation will increase with the square of the operating time. However, the surveillance program should detect degradation in sufficient time for corrective actions to be taken.

The NRC staff concludes that the licensee's inspection and surveillance program and the response to NRC Bulletin 87-01 are programmatically acceptable.

3.11 Electrical Issues

3.11.1 Overload Protection of the Motor Control Center Circuits

3.11.1.1 Introduction

In the BFNPP, the licensee described the measures it was taking to improve its nuclear program at Browns Ferry. Specific electrical issues identified in Section III.13.4 of the BFNPP addressed deficiencies associated with thermal overload (TOL) protective devices, which provide electrical protection for the 480-V ac and 125-V dc motor control center (MCC) circuits, including a brief description of the corrective program.

The design drawings for the 480-V ac and the 250-V dc MCCs at Browns Ferry did not specify TOL heater ratings. These TOL heater ratings were specified originally and documented on General Electric Company drawings which were not maintained. As a result, there was no documented evidence that TOL heaters as installed had been selected or reviewed by qualified engineering personnel.

Further, there was no evidence that the TOL heaters selected would adequately protect electrical equipment from overload or prevent the equipment from performing its safety functions.

By letter dated September 23, 1988, TVA provided the following additional information concerning the MCC TOL problem:

- the criteria for sizing the TOL heaters for MCC circuits contained in document QIR-EEB-87031

- the condition adverse to quality reports (CAQRs) that identified inadequately installed TOL heaters found during the plant walkdown

- schematic diagrams and calculation of a representative sample of the new TOL heaters selected

3.11.1.2 Evaluation

TVA's corrective program to resolve the lack of documentation comprised the following activities:

- Perform a plant walkdown, by qualified teams, of all 480-V ac and 250-V dc safety-related MCCs to determine and document the installed TOL heater element sizes and the nameplate data for each load.

- Prepare calculations using revised design standards to specify the appropriate TOL heaters for each application.

- Reconcile the results of the calculations with the walkdown observations.

- Replace or adjust the improperly sized TOL heater elements.

- Document on TVA-issued drawings the properly sized, replaced, or adjusted TOL heater elements to ensure that current and future installations of TOL heater elements are correct and documented.

Document QIR-EEB-87031, provided in TVA's September 23, 1988 submittal, establishes criteria for selecting TOL heaters to protect 480-V ac or 125-V dc continuous-duty or intermittent-duty motor-operated valves (MOVs) for all TVA nuclear facilities.

The criteria from QIR-EEB-87031 for continuous-duty motor ac or dc protection was to protect the motor for a minimum of 125 percent overload to a maximum of 140 percent overload provided the service factor is not less than 115 percent or the temperature rise is not greater than 40° (Celsius). For all other motors not meeting the service factor or temperature rise, the minimum of 115 percent overload to a maximum of 130 percent overload has been specified.

The criteria from QIR-EEB-87031 for intermittent-duty motor overload protection were to protect against stator winding overheating during running overloads and stator and rotor overheating during locked rotor conditions. The following selections would satisfy the criteria:

- motor nameplate full-load current times the service factor but not less than the motor duty cycle (15 minutes typical) except as indicated

- motor nameplate full-load current times 200 percent for minimum time of 2 minutes or more, or maximum time of 8 minutes or less

- motor nameplate locked rotor current for minimum time of 10 seconds or more, or maximum time of 15 seconds or less, with 15 seconds being preferred

If these selection criteria could not be satisfied for a unique application, then priority should be given to the criteria for locked rotor and full-load current. The rated full-load current time cannot be reduced to less than 200 percent of the valve maximum stroke time. The times listed above can be exceeded for low-horsepower MOVs if the smallest TOL has been selected.

Safety-related MOVs whose TOL devices are not bypassed when receiving an accident signal are also required by QIR-EEB-87031 to meet Position C.2 of NRC Regulatory Guide (RG) 1.106. The regulatory guide states that the trip setpoint of the TOL device should be established with all uncertainties resolved in favor of completing the safety-related function. The uncertainties to be considered are:

- variations in the ambient temperature at the installed location of the TOL device and the valve motor

- inaccuracies in motor heating data

- inaccuracies in the TOL trip characteristics

- setpoint drift

The staff has reviewed TVA's selection criteria, QIR-EEB-87031, for safety-related 480-V ac and 125-V dc motors. The staff also reviewed TVA's justification and the constraints to be observed in applying the criteria.

The staff reviewed TVA's Significant Condition Report (SCR) SCRBFNEEB-8536, Revision 2. The SCR identified the root cause of improper selection and documentation of TOL devices. The SCR provided an acceptable corrective action, including actions that are required to prevent recurrence.

The TVA SSFI review also identified a design assumption used in many overload selection calculations which was not conservative, as required by RG 1.106 (Position C.2) and, therefore, could result in the incorrect selection of the overload heaters. The statement given in Calculation ED-Q4219-87314, Revision 0, was: "If the full-load current shown on the vendor's drawing is less than documented from the walkdown data, the use of the vendor drawing information will be conservative." The TVA SSFI review concluded that, in general, the more correct data are the equipment nameplate (walkdown) data (TVA letter, September 23, 1988).

TVA walked down 18 MCCs and documented 298 TOL devices. After calculations were made for these TOL device selections, 35 of the TOL devices did not have to be replaced or reset, 20 had to be reset, and 243 had to be replaced.

The staff has reviewed the following calculations and design change authorizations (DCAs) that document the correct TOL heaters:

- Calculation ED-Q2268-87322, Revision 1, MCC-480-V Reactor MOV Board 2A DCA-H1239-003, -004, -005, and -006

- Calculation ED-Q2268-87324, Revision 1, MCC-480-V Reactor MOV Board 2C DCA-H1239-011, -012, and -013

The staff's review of Calculation ED-Q2268-87324 indicates that MOVs with design voltage of 440 V had their currents adjusted incorrectly for 460-V operation.

3.11.1.3 Conclusions

The staff concludes that TVA has identified the root cause of the correct MCC circuit protection problem. The program for corrective action and action required to prevent recurrence is acceptable. However, the staff also concludes that the problems identified in the CAQR discussed above need to be resolved; the calculations need to be reviewed and corrected; and, if necessary, the TOL devices should be replaced before Unit 2 startup.

3.11.2 Overload Protection of Circuits by Fuses That Limit Current

3.11.2.1 Introduction

The BFNPP described the measures TVA was taking to improve its nuclear program for Browns Ferry. Specific electrical issues were identified in Section III.13. Section III.13.6 addressed misapplication of fuses that limit current in overload protection.

The Browns Ferry Engineering group failed to revise the Browns Ferry specific fuse substitution list to eliminate conflicts with the licensee's new Division of Nuclear Engineering (DNE) fuse substitution list contained in Design Standard DS-E8.1.1 and DS-E8.1.2. As a result there was insufficient evidence that the installed fuses would provide electrical equipment with adequate overload protection and would not prevent the equipment from performing its safety function.

3.11.2.2 Evaluation

TVA's corrective program to resolve the problem regarding protecting electrical circuits by using fuses that limit current contained the following actions:

(1) Revise the Browns Ferry fuse substitution program control document, PSP BF6.12, to reflect the appropriate standards.

(2) Perform calculations using revised design standards to specify the appropriate fuses for each application and document this activity on the fuse tabulation document for incorporation into PSP BF6.12.

(3) Have qualified teams perform a plant walkdown to determine and document the installed fuses for compliance with the fuse tabulation, with the exception of motor control centers, where allowable substitution has been identified.

(4) Compare the results of the fuse tabulation with the walkdown for reconciliation.

(5) Document and resolve by the CAQR process all inadequate fuses identified in item 4.

(6) Delete and replace fuse ratings on design drawings with a fuse identification before restart. The fuse tabulation would be the single source of fuse requirements for the applicable fuses.

The TVA staff used document CAQR BFP 87175, Revision 0, to identify the root cause for misapplication of fuses, and described the corrective action to prevent recurrence. TVA's investigation and resolution of the issue has been found acceptable by the staff.

TVA prepared Procedure PSP BF6.12, Revision 2, to ensure that proper application, replacement, substitution, labeling, and verification of fuses is in conformance with Electrical Design Standards DS-E1.2.3, DS-E8.1.1 and DS-E8.1.2. This document will have fuse tabulation for Unit 2 incorporated in it, and with controlled copies will be located at six locations within Unit 2. The staff found this fuse control program acceptable.

Electrical Design Standard DS-E8.1.1, Revision 7, provides guidance for fuse substitution for low-voltage power and control fuses (600 volts or less) and DS-E8.1.2, Revision 4, provides guidance for fuse substitution for midget and small-dimension fuses. The staff reviewed TVA's design standards DS-E8.1.1 and 1.2, as discussed in this evaluation, and found them acceptable.

As part of the staff request for additional information, TVA in its response by letter dated September 21, 1988 enclosed the following:

- Calculation ED-Q0268-88463, Revision 0, Fuse Program 480V Reactor MOV Boards 3A/B

- Calculation ED-Q0281-88139, Revision 0, Fuse Program 250V DC Reactor MOV Boards 2A, 2B, 2C

- Schematic Diagram 45N714-2, Revision E, 250V Reactor MOV Board 1A and 45799-10, Revision G, 480V Shutdown Auxiliary Power.

- Fuse Tabulation, Drawing Change Authorization DCA-W1569-019, Revision 0

The staff has reviewed these calculations and finds that the criteria used for fuse sizing are based on accepted industry standards.

TVA has informed the staff that the appropriate schematic drawings showing fuses will be revised before Unit 2 restart to remove the fuse rating and provide a unique identification (UNID) for each fuse that can be cross referenced to the fuse tabulation. The staff finds this commitment acceptable.

In its letter of December 9, 1988, TVA informed the staff that the implementation of the fuse program did not include a total walkdown of all Class 1E circuit fuses for verification with the fuse tabulation documents. Instead, individual workplans were developed for systems required for fuel load. The total number of fuses on the fuse tabulation, as of September 19, 1988, was 6060 fuses. The number of fuses replaced in the systems required for fuel load was approximately 1500.

TVA has selected one fuse vendor to improve uniformity and to alleviate vendor supply problems, although other vendor-supplied fuses were determined to be acceptable. Because of this decision, numerous existing fuses were replaced, even though they may have been adequate. Since this fuse program has not been completely implemented, neither the total number of fuses documented during walkdown nor the total number of fuses replaced has been determined. Therefore, the total impact of the fuse program cannot be assessed at this time. TVA has confirmed that the motor control center (MCC) fuses have not been omitted from the tabulation.

TVA has identified the Browns Ferry Site Director Standard Practice (SDSP) procedure SDSP 16.8, Revision 0, "Fuse Control," as the procedure that defines the requirements for controlling procurement, installation, and maintenance of electrical fuses. By letter dated December 20, 1988, TVA provided a copy of this procedure to the staff for review. The staff finds that this procedure provides adequate administrative control for fuses at Browns Ferry Unit 2.

3.11.2.3 Conclusions

The staff concludes that TVA has identified the root cause of misapplication of current-limiting fuses and the program provided by TVA for corrective action is acceptable. Implementation, including the fuse tabulation on drawings, will be completed before Unit 2 restart.

3.11.3 Ampacity (To be addressed in a supplement to this volume)

3.11.4 Cable Installation, Including Cable Separation (To be addressed in a supplement to this volume)

3.12 Flexible Conduit (To be addressed in a supplement to this volume)

3.13 Cable Splices (To be addressed in a supplement to this volume)

4 READINESS FOR OPERATION

4.1 Operational Readiness Review Program (To be addressed in a supplement to this volume)

4.2 Management (To be addressed in a supplement to this volume)

4.3 Quality Assurance

4.3.1 Conditions Adverse to Quality

TVA has acknowledged that it had not always taken timely action to resolve conditions adverse to quality (CAQs) in its nuclear activities. This problem included a lack of upper-level management involvement and a lack of timely processing of CAQs 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 of the new CAQ process
- frequent status meetings
- procedure changes requiring prompt assessment of safety significance when a CAQ is identified

Because the NRC staff has found acceptable the way CAQs are handled at Sequoyah Nuclear Power Plant, and in view of the fact that Browns Ferry Nuclear Power Station is using the same procedures, the staff finds the program for CAQs at Browns Ferry acceptable.

The staff will perform implementation inspections of the handling of CAQs at Browns Ferry before restart. The subsequent inspection report will provide the measure of effectiveness of the implementation of the Browns Ferry CAQ program.

4.3.2 Quality Assurance Program

4.3.2.1 Introduction

The TVA organization for quality assurance (QA) that has been in place 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" (TVA letter, April 29, 1976). This report contains organization charts, a description of the organization, and the QA responsibility assignments. The NRC staff has been informed of changes in the QA organization. The staff has reviewed and approved each organizational arrangement reported 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 quality assurance.

As noted in the revised BFNPP, 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 the engineering discipline was responsible for auditing 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 in an effective, consistent manner.

4.3.2.2 Evaluation

The staff evaluation of TVA's Browns Ferry Quality Assurance Program is based on a review of Section II.2.6, "Quality Assurance," in Revision 1 to the BFNPP.

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 standardized 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 QA program as generally described in the revised BFNPP represent QA programmatic improvements and, if properly implemented, are acceptable.

TVA submitted a revised and improved version of its QA topical report (TVA-TR75-1A) for NRC review on May 1, 1986. The report described the then-current organization and QA procedure system. After reviewing the report and meeting with TVA representatives, the staff forwarded a request for additional information to TVA on August 1, 1986. TVA revised the topical report to address these staff questions and to fully reflect the organization of the Office of Nuclear Power; NRC conditionally approved the submittal in a letter sent to TVA on January 30, 1987.

Only by observing TVA's performance over an extended period can the staff determine if the changes in the TVA QA topical report will resolve past problems. As noted above, the problems in TVA's nuclear activities occurred under a previously approved QA program; however, that program was not implemented in 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 Part 50, Appendix B, as described in Chapter 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.

4.3.2.3 Conclusion

On the basis of its review of the Browns Ferry Quality Assurance Program provided in Section II of the BFNPP (Rev. 1), the staff finds that with proper implementation the quality assurance program is acceptable.

4.4 Plant Surveillance Program

4.4.1 Introduction

The licensee reviewed the surveillance program at Browns Ferry as part of the development of its BFNPP. The resulting program assessment identified seven root causes for surveillance-related deficiencies which had resulted in numerous notices of violations from the NRC. These root causes were grouped into one of two general categories: (1) unclear, difficult-to-use procedures and (2) insufficient attention to detail by personnel performing surveillances and reviewing the surveillance results. In the BFNPP, TVA described four specific programmatic actions which would be implemented to correct the identified root causes.

The staff reviewed the programmatic actions described in Section II.5.0 of the BFNPP to determine if these actions would correct the identified root causes. The actions, and the root causes they are intended to correct, were:

Program	Associated root cause
Surveillance instruction (SI) upgrades	Inconsistent acceptance criteria in SIs
	Difficult-to-use procedures
Vendor manual control program (VMCP)	Inaccurate or outdated technical criteria
Improved management practices	Failure to follow procedures
Implementation of systems engineer concept	Incomplete technical reviews
	Shallow resolution of technical issues
	Delayed corrective actions

4.4.2 Evaluation

The staff provided its review of TVA's upgraded surveillance program to TVA by letter dated September 29, 1988. TVA was asked to address the following issues:

- implementation of the systems engineering concept with respect to the plant surveillance program

- implementation of an expanded third-party or qualified independent observer approach for SI validations to ensure consistency and the quality needed for an effective surveillance program

- review of circuit and piping flow paths identified by SI drafters in developing SIs

- tracking of the programs committed to in the nuclear performance plan as specific licensing commitments

TVA responded to these issues by letter dated October 31, 1988. The staff evaluated TVA's responses and concluded in its letter dated January 3, 1989 that TVA had satisfactorily addressed the programmatic issues raised by the staff. In addition, the staff concluded that TVA's justification on the acceptability of surveillance instructions already developed and in place was acceptable.

The staff did note, however, that there were issues relating to the field implementation of the surveillance program that still required TVA management attention. These issues are presented below:

<u>Systems Engineer Concept in Relation to the Plant Surveillance Program</u>

The staff reviewed TVA's clarification of the BFNPP passages regarding the role of the systems engineers in the review and trending of surveillance instruction (SI) data. The revised division of responsibilities for SI data review and trending, if properly implemented, would effectively correct the root cause of inadequate SI reviews in the past, as identified in the BFNPP (Rev. 2): "In the past, SI reviews were done by engineers who had day-to-day responsibilities other than their assigned system cognizance. This effectively diluted the amount of time which could be spent on system performance evaluations such as SI review."

<u>SI Validations</u>

Although TVA has increased the scope of third-party observations of SI validations, the issue of proper validation of SIs, as reflected in many recent cases of poor SI program implementation, remains. Validation is the final review step in the development of each SI, and its most important function is to ensure that SIs can be performed as written. If an SI is improperly validated, then it may not be workable. One factor that contributed to personnel error associated with the conduct of SIs at Browns Ferry in the past was unworkability of procedures. Several recent instances of personnel error in the conduct of surveillance testing have been cited in licensee event reports (LERs) from Browns Ferry as well as some cases of inadequate SIs resulting in reportable events. Although the specific events in question are closely intertwined with staff concerns about the success of improved management practices in reducing personnel error, the link between personnel error, SI workability, and SI validation indicates that these events may have been partially caused by inadequate SI validation.

<u>SI Verification</u>

The staff agrees that TVA's SI development process is programmatically constructed to allow the development of technically correct SIs. However, there appear to be problems with the ability of this process to produce SIs that are capable of accurately testing those items that they are intended to test. Because the SI development process at Browns Ferry is programmatically sound, staff concerns in this regard rest with the implementation of the program. Staff review of the Browns Ferry SSFI report submitted to NRC on September 23, 1988, noted several cases of items requiring surveillance testing for which SIs either did not exist or did not adequately test all the flow paths, system lineups, or devices requiring testing per the Technical Specifications. These concerns are detailed in SSFI Report No. BFA 88811, observations BF-SMK-4, BF-SFI-4, BF-RB-1, and BF-RB-2. As well, LER 50-259-88-035 identified procedural inadequacy as the root cause of an unplanned initiation of control room emergency ventilation. Although the inadequacies discussed in these SSFI and LER findings are significant in themselves, they also indicate potential generic problems with the implementation of the SI development and review process.

Improved Management Practices to Foster Procedural Compliance Among Personnel

As noted in both the NRC safety evaluation and in TVA's response to the safety evaluation, Browns Ferry management has reemphasized the importance of demanding that instructions be performed as written. These improved management practices should effectively reduce personnel error associated with surveillance testing; yet several recent instances of personnel error cast doubt on the effectiveness of these practices. Specifically, LERs 50-259-88-041, 50-260-88-007, and 50-260-88-011 cite personnel error as the root causes of events related to testing.

Commitment Tracking

It is incumbent on TVA to ensure that all commitments contained in the BFNPP, which is TVA's response to NRC's September 17, 1985 information request pursuant to 10 CFR 50.54(f), are adequately tracked and satisfied in order to prevent unnecessary delays in the resolution of outstanding licensing issues. The staff will review this commitment tracking capability as part of the Quality Verification Inspection before restart.

4.4.3 Conclusions

TVA's upgraded surveillance program is acceptable. The staff concludes that the proper implementation of the corrective actions to the surveillance program will determine the effectiveness of Browns Ferry surveillance testing. Based in part on the above discussion, the staff is planning to perform a final team inspection of the Browns Ferry surveillance program before Unit 2 restart. Although this inspection will be oriented toward assessing the readiness of the surveillance program at Browns Ferry to support an operating nuclear power plant, each of the issues described in the preceding sections will receive particular attention. Any safety-significant issues discovered during this inspection will have to be resolved before the restart of Browns Ferry Unit 2.

4.5 Maintenance (To be addressed in a supplement to this volume)

4.6 Restart Test Program

4.6.1 Introduction

The licensee scrutinized the operability of Browns Ferry Unit 2 safety systems and their capability to perform their safety functions in response to employee-generated concerns, a prolonged plant shutdown, and extensive plant modifications. TVA conducted a major re-review of the Browns Ferry Unit 2 initial design, construction, and operating practices and instituted a restart test program (RTP) to ascertain the functional integrity of the accident-mitigation and safe-shutdown systems. The proposed program is described in TVA letters dated October 7, 1986 and July 13, 1987, and in Section III.8.0 of the BFNPP.

The NRC staff has inspected the implementation of the RTP several times and has documented these inspections in the following inspection reports (IRs):

Inspection Report 50-259, 260, 296/	Date Issued
87-12	March 30, 1987
87-27	September 2, 1987
87-30	September 29, 1987
87-33	November 19, 1987
87-37	December 3, 1987
87-42	January 6, 1988
87-46	February 26, 1988
88-02	March 24, 1988
88-04	March 24, 1988
88-05	May 20, 1988
88-10	June 3, 1988
88-16	September 12, 1988
88-18	September 22, 1988
88-21	October 25, 1988

4.6.1.1 Overall Scope and Objective

The principal objective of the RTP is to engender confidence that certain preoperational tests performed during initial plant licensing and surveillance inspections, routinely conducted following plant licensing, were valid tests that could ensure the current functional integrity of safety systems and components. Where changes in system configuration or inadequate validation of existing preoperational test results dictate a retest, the RTP provides for that testing, developed by a TVA-approved organization with established controls and procedures. The prolonged outage at Browns Ferry and the extensive plant modifications performed there were considered in the RTP. This dictated consideration that all plant systems required for shutdown and cooldown of the reactor under transient and accident conditions be reviewed in terms of the documentation of adequate testing and verification that the required safety systems will perform their intended safety functions.

The scope of testing in the RTP is governed by TVA Browns Ferry Site Director Standard Practice (SDSP)-12.1, "Restart Test Program", Revision 3, dated February 24, 1988 and SDSP-12.2, "Development of System Test Specifications", Revision 4, dated February 24, 1988. The scope of SDSP-12.1 specifically addresses the commitments made in Section III.8.0 of the BFNPP for the RTP. The scope of SDSP-12.2 stipulates the test requirements for the systems selected to be tested as part of the RTP. The scope of the RTP includes individual system testing, as well as integrated system tests, such as a simulated loss-of-coolant accident (LOCA) concurrent with a loss of offsite power (LOP)

(LOP/LOCA test) and a backup control test, which tests the controls required for shutdown from outside the control room. These integrated tests will provide for additional verification of procedures and equipment, and further operator training.

4.6.1.2 Organization

As illustrated by Figure 1 of TVA's July 13, 1987 submittal, the Browns Ferry site director is responsible for the overall implementation of the RTP and for providing the necessary cooperation with other TVA organizations. A Joint Test Group (JTG) is responsible for reviewing the scope and technical adequacy of the RTP and for making recommendations to the plant manager. The plant manager is responsible to the site director. The RTP organization also includes an RTP manager and certified test engineers responsible for ensuring the restart tests are satisfactorily completed. All of the RTP test result packages are reviewed by the JTG.

The JTG membership includes the (1) Unit 2 superintendent (now titled the Operations Superintendent) as the chairman and (2) representatives, as appropriate, from TVA's divisions of Nuclear Engineering (DNE), Maintenance, Modifications, Operations, Quality Assurance, Restart Test, and Technical Support Services, as well as the General Electric Co. (nuclear steam supply system [NSSS] vendor representative). The staff finds the JTG membership diverse enough to provide the technical expertise required to conduct the RTP review functions and to evaluate test results; therefore, the JTG is an acceptable organization to function as a subcommittee to the plant operations review committee (PORC).

The staff has reviewed the membership of the RTP organization and concludes that an adequate representation of technical expertise exists to achieve the RTP objectives.

4.6.1.3 Methodology

The responsibility of DNE, as part of the design baseline and verification program (DBVP), includes the verification and/or generation of the design criteria necessary to document system design functions utilized in satisfying the safe-shutdown analysis. The DBVP-generated test requirement documents (TRDs), produced from the system requirement calculations, are fundamental to the RTP. Browns Ferry Engineering Procedure (BFEP) Project Instruction PI 86-26 governs the generation of the TRDs.

The TRDs are compiled and documented by the RTP test engineer in system test specifications (STS). The STS is a document that specifies the minimum testing to be performed on selected systems for the RTP. From this document, an RTP test instruction is written to implement the test requirements. Many test requirements have resulted directly from the DBVP, as indicated above. However, test requirements may also result from the RTP system review discussed below.

The designated RTP test engineer performs a system review to determine test requirements necessary for reliable system operation. This system review addresses items such as past maintenance and operational history documentation, vendor-recommended testing, the extent of modifications performed during the current Unit 2, Cycle 5 outage, and licensing commitments. As a result of the DBVP and these system reviews, TVA has determined that some systems will require extensive testing while others will need no specific or special testing. A system checklist (SCL) is used to determine the operational status of those systems that have no testing requirements. Items such as system procedures, hold orders, temporary alterations, engineering change notices, and a system walkdown are are included in the SCL.

RTP test instructions allow for the utilization of existing site test procedures, where appropriate, to satisfy testing requirements. Specific test requirements not covered by existing plant instructions require that step-by-step test instructions be written. The RTP, including the STS generation, test instruction development, test conduct, and test results review, is administratively controlled by SDSP-12.1 and SDSP-12.2.

4.6.2 Evaluation

4.6.2.1 Selection of Systems for Testing

(1) <u>RTP Programmatic Exceptions</u>

The RTP must address the systems necessary to support the safe shutdown of Unit 2 following an accident. During a meeting between NRC staff and TVA, on April 26, 1988 (see meeting summary letter dated May 24, 1988), the staff requested additional information regarding the differences between typical industry preoperational test programs, as described in NRC Regulatory Guide (RG) 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, August 1978, and the Browns Ferry RTP. TVA provided the requested information during a meeting on June 21, 1988 (see meeting summary letter dated July 27, 1988).

Position C.1 of RG 1.68 provides industry guidance regarding the criteria used for selecting of plant systems to be preoperationally tested. Positions C.1.b, d, and e of this criterion identify the equipment necessary to support safe shutdown and cooldown of the reactor under <u>transient and postulated accident conditions</u>, the equipment classified as engineered safety features, and the equipment assumed to function or for which credit is taken in the accident analysis of the facility, as described in the final safety analysis report (FSAR). Since the Browns Ferry DBVP evaluations have identified the equipment necessary to support safe shutdown for the FSAR Chapter 14 design-basis events, the required equipment testing identified for preoperational testing in RG 1.68, Positions C.1.b, d, and e would be satisfied if they were included in the Browns Ferry RTP. The Browns Ferry RTP approach developed by TVA is, in fact, consistent with RG 1.68 since the DBVP provides the majority of the input to the RTP.

Therefore, the staff finds the scope of the accident-mitigation and safe-shutdown systems, which are to be tested before Browns Ferry restart, acceptable. Deviations or exceptions that occur during testing of this equipment are required to be evaluated and dispositioned by TVA per SDSP-12.1, Section 6.6, "Test Exceptions," and must be available for NRC staff audit.

RG 1.68, Positions C.1.a, c, and f are excluded in part from the Browns Ferry RTP. These criteria identify those plant structures, systems, and components that are used for shutdown and cooldown of the reactor under <u>normal</u> plant conditions, that are tested under Technical Specifications, and that are used to process, store, control, or limit the release of radioactive materials.

Equipment associated with normal plant cooldown is excluded from the RTP. TVA has identified and evaluated all of the mechanical and electrical systems interactions that exist with the systems required for safe shutdown from transients, accidents, and special events, and has ensured that a failure of normal plant cooldown equipment would not prevent Browns Ferry from achieving safe shutdown. Therefore, since safe shutdown remains assured, the staff finds this RTP exclusion acceptable.

The Browns Ferry Technical Specifications (TS) must be complied with as they are appended to the operating license for the facility. The tests required by the TS are performed regardless of the scope of the RTP. Therefore, since there is no intention to supersede the Browns Ferry TS surveillance requirements, the exclusion of these tests from the RTP is appropriate and acceptable.

RG 1.68, Position C.1.f identifies those plant structures, systems, and components that would be used to process, store, control, or limit the release of radioactive materials. The Browns Ferry RTP includes those systems required to support accident mitigation and safe shutdown as described above. Systems and equipment other than those required for performing the above functions were evaluated for adverse interfacial impact on the required systems as described above. Those structures, systems, and components having no potential adverse impact were excluded from the RTP. The staff has evaluated this exclusion and has found it acceptable since all of the systems required to mitigate the radiological consequences of an accident are included in the RTP.

(2) Selected Systems

By letter dated July 13, 1987, TVA provided, as Attachment 1, an RTP system test list. On this list are the title and associated number of the system as well as the RTP group assignments for all of the systems to be tested as part of the RTP.

NRC Inspection Report 50-259, 260, 296/87-36 (January 21, 1988) documents the staff's review of the information contained in the DBVP for Browns Ferry. The inspection effort included a review of the Browns Ferry safe shutdown analysis (SSA). The SSA includes a list of systems required for Browns Ferry to achieve and maintain safe shutdown during design-basis events. The list of systems to be tested under the RTP agrees with the SSA list. The staff concluded during the inspection that TVA had adequately addressed the areas of mechanical and electrical systems in the DBVP and the SSA. Since the RTP system test list was taken from the SSA, the staff has concluded that this system list is complete and is, therefore, acceptable.

(3) RTP Testing Type Exceptions

Section 1.0 of Appendix A to RG 1.68 gives guidance about the type of testing performed during preoperational test programs. These tests may include manual and automatic operation, and verification of operation following loss of normal power supplies. The scope of testing provided in the Browns Ferry RTP includes consideration for these types of tests with the exception of testing in the degraded mode. This exception is based on TVA calculations that ensure the systems or portions of systems necessary to provide for safe shutdown can perform their required safety functions during the design-basis worst-case conditions. Section 1.0 of Appendix A to RG 1.68 also specifies that tests must include, as appropriate, proper function of instrumentation and controls, permissive and prohibit interlocks, protective devices on equipment whose malfunction or premature actuation may shut down or defeat system or equipment operation, and system vibration, expansion, and restraint testing. The Browns Ferry RTP scope includes consideration of these test types with the exception of vibration, expansion, and restraint testing.

During the June 21, 1988 meeting between NRC and TVA (meeting summary issued July 27, 1988), these exceptions (with particular concern for the exclusion of vibration measurements of systems during testing) were discussed. Piping vibration has been a problem at Browns Ferry in the past; however, corrective actions were implemented in each instance. The licensee stated that vibration testing was not required on the basis of the results of testing performed during the Browns Ferry original preoperational testing program, the plant's operational history since that time, and the corrective actions taken whenever excessive vibrations were observed. The licensee has also indicated that periodic TS surveillance instructions (SIs) require verification of major pump vibrations in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Section XI). Since the licensee has corrected excessive vibration whenever it was identified during Browns Ferry's previous operational history and since the majority of Browns Ferry's required systems that have significant flow velocities (e.g., core spray, reactor core isolation cooling service water, and emergency equipment cooling water) have not historically experienced vibration problems during operation, the staff finds the programmatic exclusion of vibration measurements acceptable. The staff, however, recognizes that in accordance with TS requirements, the licensee will continue to conduct required periodic surveillance instructions to verify acceptable critical pump vibration in accordance with ASME Section XI. The staff also expects the RTP test engineers to rely on their engineering judgment in reporting any detectable excessive vibration during individual or integrated restart tests.

The staff has evaluated the exceptions discussed above and has concluded that they are acceptable because (1) the calculations support the elimination of degraded mode testing, (2) the excepted testing types would provide little or no additional data for system operability assessment (e.g., performing expansion measurements with no significant increase in system temperature), or (3) the excepted testing types will be conducted by the Post Modification Test Section (e.g., restraint testing subsequent to modifications) at Browns Ferry. Piping expansion and restraint testing should be performed subsequent to system modification and/or be incorporated in the power ascension test program, as appropriate.

In summary, various system-specific test exceptions have been taken for the specific reasons provided by TVA. The staff has evaluated these exceptions and finds them acceptable.

4.6.2.2 Implementation

(1) Individual System Testing

TVA's implementation of the Browns Ferry RTP has resulted in the categorization of systems to be tested and the degree of testing before Browns Ferry Unit 2 restart. The systems have been categorized for convenience and clarification into one of three groups. The licensee has defined these three groupings as follows:

- Group 1: Systems determined to be critical to the safe shutdown of the plant

  Testing requirements are determined primarily by the DBVP. A system test specification (STS) will be prepared and an RTP test instruction will be written and conducted. A test results package will be compiled for each system so as to auditably document the test results and system readiness status.

- Group 2: Systems having few test requirements specified by the DBVP but providing direct support to plant operation

  Most of the system is addressed by the RTP review to determine test requirements. An STS will be prepared and an RTP test instruction will be written and conducted. A test results package will be compiled for each system so as to auditably document the test results and system readiness status.

- Group 3: Systems not directly supporting plant operation nor important to safety but, in general, requiring no testing

  No system test requirements are provided from the DBVP. These systems will not have an STS or an RTP test instruction prepared, but will be addressed by the requirements of the system checklist (SCL).

These groupings provide a prioritization methodology for conducting the RTP system tests. The primary difference between groups 1 and 2 is the scope of the DBVP test requirements for the particular system. A group 2 classification for a system may have a few DBVP test requirements; however, most of these system tests result from the RTP organization's system review. The staff has evaluated the grouping of systems in the RTP and has found that groups 1 and 2 receive the same levels of technical review, administrative control, and approval and are, therefore, acceptable. Systems placed in group 3 are not important to safety; therefore, the use of only an SCL to determine system operability status is also acceptable.

(2) Integrated System Testing

In addition to individual system testing, the RTP includes integrated systems tests. These tests, which are outlined below, are being performed to provide added confidence in integrated systems performance, to provide additional verification of plant procedures and equipment, and to further operator training.

LOP/LOCA Test

Plant response to a loss of all offsite power concurrent with a simulated LOCA will be demonstrated.

- Backup Control Test

  The controls required for shutdown from outside the control room (safe alternate shutdown) will be tested by verifying proper operation of local control transfer switches and the functions of the remote shutdown panels.

- Integrated Cold Functional Testing

  Systems will be operated per operating instructions in an integrated manner, as much as possible, before fuel loading or restart. For example, operators would "pull" a condenser vacuum and operate the circulating water, condensate, and feedwater systems together.

(3) Programmatic Implementation

In order for the staff to acquire additional confidence in the programmatic implementation of the Browns Ferry RTP, the staff selected two safety systems for evaluation in terms of the adequacy of the use of DBVP Baseline Test Requirements Document, system test specifications (STS), and associated test instructions. These systems are the standby gas treatment and standby diesel generator systems. The results of these evaluations are provided below.

(a) RTP Implementation for the Standby Gas Treatment System

To establish that the programmatic aspects of the RTP are correctly implemented, the staff and its consultant reviewed the restart test results package for the standby gas treatment system (SGTS) (2-BFN-RTP-065). The package was chosen as a representative sample of the ongoing RTP effort. The test results package contained the following documents:

- test summary
- system checklist (SCL)
- system test specification (STS)
- system test instruction (STI)
- system punchlist (SPL)

As such, the test results package content followed guideline SDSP-12.1.

The staff and its consultant reviewed the completeness and consistency of the following three documents:

- test requirements
- system test specification
- system test instructions

The staff and its consultant confirmed that the test requirements document followed procedure BFEP PI 86-26.

The STS was developed by the restart test group (RTG) by combining the Division of Nuclear Engineering (DNE) test requirements as well as by reviewing the following:

- engineering change notices

- nuclear plant reliability data system (NPRDS) and maintenance request (MR) history

- employee concerns, licensing commitments, and condition adverse to quality (CAQ) programs

- vendor recommendations

The staff and its consultant verified that the STS were written in accordance with SDSP-12.2 and SIL-007.

The STI is a comprehensive document that provides the system acceptance criteria, thereby dictating the steps to be followed during the test. The STI also documents test exceptions and lists surveillance instructions used to satisfy test steps. The staff and its consultant verified that the STIs accurately reflect the tests identified in the STS. The STIs were written in accordance with SIL-003.

Test exceptions (TEs) were included in Appendix B of the STIs. TEs are written when an RTP test instruction step cannot be completed or when the validity of data is questionable. Within the area of dispositioning RTP TEs, the staff has some concerns which are being monitored in order to ascertain the effectiveness of the RTP. Although the staff is concerned about the absolute number of TEs, of more importance are the evaluation, resolution, and effect on the restart test acceptability. The following concerns have been noted by the NRC resident inspectors and are documented in further detail in NRC Inspection Report 50-259, 260, 296/88-21 (October 25, 1988).

The RTP is not a stand-alone activity. The licensee has been reluctant to issue and process a condition adverse to quality report (CAQR) for TEs that clearly should require such reports under TVA's CAQ program. The tendency has been to identify problems and fix them by investigation, analysis, evaluation, and sometimes, resolution of problems identified in the RTP, solely under the TE activity. This action is not consistent with the CAQ program that should cover all plant activities affecting quality. These concerns have been discussed with Browns Ferry managers. TVA has stated that it will provide CAQRs in parallel with TEs where conditions warrant them. The NRC staff will continue to monitor this activity.

Under the Browns Ferry RTP, it is possible to satisfactorily close out a test without closing all TEs against that test. Those TEs should be classified by their significance and tracked on the SPLs. However, the overall program to provide for the appropriate identification, tracking, resolution, and closure of the significant TEs identified in the RTP should include the CAQR.

About 21 TEs were generated by the RTG for the SGTS. The open TEs are listed in the test summary as required by SDSP-12.1. Changes to the RTP tests were handled through the RTP change sheet described in SDSP-12.1. Fifteen change notices were reviewed. Finally, the SCL and SPL were reviewed. The SCL was used to identify open items related to the operational readiness of the system and to establish a punch list of open items and their priorities. The SCL followed procedures SDSP-12.1 and SIL-006. The SPL is used to track and expedite work associated with open items affecting the RTP. The administration and control of the SPL are governed by SIL-005. Therefore, based on the above, the staff finds the RTP programmatic implementation for the SGTS acceptable.

(b) <u>RTP Implementation for the Standby Diesel Generator System</u>

The TVA DBVP and the RTP organization's review has resulted in the identification of the functional test requirements for the standby diesel generator (DG) system, which, in summary, is to demonstrate that this system will perform all required safe-shutdown functions. The integrated plant response to an accident will be verified by a series of special tests consisting of:

- loss-of-offsite-power test
- Unit 2 simulated loss-of-coolant-accident test
- Unit 2 simulated loss-of-coolant accident combined with a loss-of-offsite power with diesel generator D disabled
- Unit 2 simulated loss-of-coolant accident combined with a loss of offsite power with battery No. 2 disconnected

These tests demonstrate that the diesel generators (DGs) will start on automatic and manual initiation signals, provide power to the 4-kV distribution system, and provide verification of proper operation of the equipment necessary to effect the safe shutdown of the plant. In addition to these tests, TVA has also proposed special tests to determine and verify the capability of DGs A, B, C, and D to accept their emergency loads in accordance with the DG loading requirements. TVA has proposed to test DGs A and B close to their full-load requirement and will verify by computer model the actual test results and predict the DG response during full load. All other DGs will be partially loaded and their test results and full-load response will be verified by the computer model.

The staff has had several meetings with TVA to discuss the applicable baseline test requirements, STS, and procedures. On the basis of its review as supplemented by these discussions, the staff has determined that the planned testing of the DGs, in accordance with the above RTP-generated documents, will identify any problems with the DGs. After any identified problems have been resolved, TVA will have demonstrated the functional integrity of the Browns Ferry standby DG systems. Any test exceptions or deficiencies and their resolutions will be available for a staff audit. Therefore, on the basis of its review, the staff finds the RTP programmatic implementation for the standby DG system acceptable.

Although some weakness has been observed in the programmatic dispositioning of TEs, the RTP procedures reviewed constitute an acceptable program for dispositioning TEs. The staff will continue to monitor TVA's activity in this area.

4.6.2.3 Administrative Controls and Implementing Procedures

Section III.8.0 of the BFNPP states that the RTP will be conducted to ensure that plant systems are capable of meeting their safe-shutdown requirements. Also stated is that coordination and development of the RTP are the responsibility of the restart test manager and that an element of the RTP will be the establishment of a program to control test performance and evaluate test results.

As previously discussed, Project Instruction BFEP PI 86-26 governs the generation of the test requirements document (TRD) by DNE. SDSP-12.1, "Restart Test

Program," and SDSP-12.2, "Development of System Test Specifications," were developed to administratively control the RTP processes and the documentation and use of the generated implementing procedures. These procedures govern the RTP in terms of the use of the DBVP-generated baseline test requirements through and including the generation and use of the system test specifications, procedure implementation reviews, and approval of the completed restart test procedures.

To assist in the implementation of the Browns Ferry RTP, eight additional procedures have been written. These are called section instruction letters (SILs) and are numbered 001 to 008. A brief description of these procedures follows:

- SIL-001, "Preparation and Use of Division of Nuclear Engineering (DNE) Need Sheets," prescribes how to prepare and use DNE need sheets. The DNE need sheets are used by the restart test engineer to list items that require DNE action.

- SIL-002, "Training and Qualification of Restart Test Program Personnel," establishes the training and qualification for RTP personnel and specifies work activities allowed for certified personnel.

- SIL-003, "RTP Instruction Example Formats," provides sample forms and pages for use in developing RTP test instruction appendices and signature logs.

- SIL-004, "File Indexes," provides guidelines for filing and indexing RTP correspondence to ensure adequate record accountability and retrievability.

- SIL-005, "System Punch List Program," provides a program for the administration and control of a system punchlist. This punchlist assists tracking and expediting work associated with open items affecting the RTP.

- SIL-006, "System Check List Preparation," specifies the process by which a system checklist (SCL) is completed for systems, as required by SDSP-12.1. The steps involved in using the SCL will identify open items related to the operational readiness of a system and its documentation, and establish a punchlist of open items and their priorities.

- SIL-007, "Review Documentation Reports," outlines a process for documenting and revising the review process used to generate the system test specifications.

- SIL-008, "Restart Test Program Procedure Review Group," establishes a procedure review group whose basic purpose is to review documents generated by the RTP to ensure they are technically accurate, administratively correct, and adequate in scope to support their intent per SDSP-12.1/SDSP-12.2 before being submitted to the Joint Test Group.

The staff and its consultant have reviewed all of these procedures and found them adequate in content and detail so that proper implementation of these procedures would result in an effective RTP. Therefore, the staff finds these documents acceptable.

4.6.3 Conclusion

On the basis of the reviews discussed above, the staff concludes that continued implementation of the Browns Ferry RTP, as currently constructed, will ensure proper verification of the functional integrity of the safety systems at Browns Ferry Unit 2.

4.7 Training (To be addressed in a supplement to this volume)

4.8 Plant Security

4.8.1 Introduction

In Section II.7.0 of the BFNPP TVA identified several initiatives that are intended to improve the security operation and management. Procedural improvements, better tactical training, decreased use of long-term compensatory measures, resolution of employee concerns/allegations, and more extensive self-audits were the more notable initiatives. The NRC staff, in previous systematic assessment of licensee performance (SALP) ratings and escalated enforcement actions, identified these same issues as needing improvement.

4.8.2 Evaluation

The NRC staff has repeatedly inspected Browns Ferry during the current shutdown. At no time during the current Browns Ferry shutdown did the licensee suspend or "devitalize" any of its security restrictions. The NRC staff performed eight routine inspections and four special inspections (one of which was the NRC regulatory effectiveness review [RER]).

With respect to the RER audit at Browns Ferry, the team found several "safeguards inadequacies" relative to camera assessment of alarms and the associated use of long-term compensatory measures. An inadequacy is defined as a safeguard deficiency which, if left uncorrected, would cause the overall security program to fall below the intended level.

The NRC staff performed a special inspection in April 1987 (NRC Inspection Report 50-259, 260, 296/87-17, May 27, 1987) to verify the completion of those initiatives found in the BFNPP (Section II.7.0). Several allegations and employee concerns were also investigated at that time. The allegations and employee concerns related to such issues as shift staffing, inconsistent supervisory guidance, and lack of strict procedural adherence; employees also suggested ways to improve the security organization.

As a result of these inspections, the staff concluded that the licensee has corrected most of the RER findings, reduced the use of overtime, completed an extensive retraining effort, improved access controls, inspected security boundaries, and, most importantly, developed a well-managed security organization at the site.

The licensee has made a commitment by letter dated September 28, 1988 to complete to the satisfaction of the NRC staff the following security-related actions before Unit 2 restart:

- Test security emergency power.
- Reduce the size of the protected area.

- Improve alarm assessment capabilities.
- Reduce compensatory measures.

4.8.3 Conclusions

On the basis of the corrective actions taken to date, the staff concludes that the physical security program at Browns Ferry with the appropriate implementation of the above commitments will be adequate. The staff will inspect the security program before Unit 2 restart. Therefore, the staff concludes that TVA's measures are sufficient to support plant restart.

4.9 Emergency Preparedness

4.9.1 Introduction

TVA identified the restart corrective actions for the emergency preparedness program in Section II.8.0 of the BFNPP. In a September 17, 1985 letter, the NRC staff notified TVA of the results of the NRC senior management team review in relation to the fifth SALP report. Item F, "Emergency Preparedness," discussed the licensee's past performance in emergency preparedness and the NRC review board recommended that TVA management should direct its attention to the resolution of IE Bulletin 79-18, "Audibility Problem Encountered on Evacuation of Personnel From High-Noise Areas," which dealt with the audibility of alarms in high-noise areas. In addition, the NRC staff noted that two items identified in the 1981 emergency preparedness implementation appraisal as needing improvement remained outstanding: (1) Present temperature difference recorders did not allow the user to differentiate among atmospheric stability conditions, and (2) an emergency plan implementing procedure did not adequately reference applicable health physics standard instructions to be used during an emergency.

4.9.2 Evaluation

In response to IE Bulletin 79-18, TVA initiated a design change to improve the public address and evacuation system. An independent consulting firm was hired to determine current noise levels within the plant and to recommend equipment that the plant needed. The consultant's review has provided the basis for design of an upgraded system that will include in-plant areas as well as the other onsite buildings. TVA has committed to install the upgraded system during the Unit 2 cycle 6 outage.

The site radiological emergency preparedness (REP) manager has been made responsible for ensuring that the evacuation alarm system is adequate both inside the plant and in all other areas on site. On September 25, 1987, Emergency Plan Implementing Procedure EPIP-8 was issued, which included provisions for ensuring that high-noise areas are checked during assembly, as needed. In addition, evacuation alarms were installed and operational in Browns Ferry buildings outside the protected area in November 1985.

These administrative controls are in place to verify that personnel are evacuated from the high-noise areas in the plant. The NRC staff closed this bulletin in Inspection Report 50-259, 260, 296/86-01 (February 4, 1986).

Two improvement items identified in the 1981 emergency preparedness implementation appraisal have been closed out satisfactorily after the 1985 radiological emergency preparedness exercise.

Subsequent inspection by NRC identified additional items that require further resolution in the licensee's emergency preparedness (Inspection Report 50-259, 260, 296/88-30, November 15, 1988). The NRC staff conducted this inspection on September 20-22 and 29, 1988. The report discusses the exercise weaknesses regarding inadequate onsite accountability and failure to demonstrate timely and complete emergency information flow within and between the emergency response facilities. As described in the inspection report, corrective action must be completed before restart.

4.9.3 Conclusions

For the reasons stated, the licensee's radiological emergency preparedness is satisfactory. However, certain deficiencies in the program as discussed in this evaluation must be corrected and their adequacy demonstrated to NRC's satisfaction as described in NRC Inspection Report 50-259, 260, 296/88-30 before restart.

4.10 Radiological Control and Chemistry Improvement

4.10.1 Introduction

In Section II.2.6 of the BFNPP, TVA identified the restart corrective actions for the radiological control and chemistry improvement program.

4.10.2 Evaluation

4.10.2.1 Radiological Control

In 1985, within the framework of the SALP program, the staff noted previous improvements made in the Browns Ferry nuclear plant radiological control (RC) program (NRC letter, September 17, 1985). However, the staff criticized the RC program for overreliance on contract technical personnel. In response to these concerns, TVA completed the following improvements:

- Since 1985, the number of technicians employed by TVA in the Browns Ferry RC section who are qualified to Standard 18.1 of the American National Standards Institute (ANSI 18.1) increased from 21 to 97.

- The Browns Ferry RC section has six qualified dosimetry technicians and nine trainees. A minimum of three full-time TVA professional management personnel are assigned to oversee RC technician training for all trainees.

- To provide onshift guidance to RC technicians and trainees, a minimum of one and usually two or more RC supervisors are on site 7 days a week, 24 hours a day. Contracts have been extended to ensure that experienced, well-qualified contract RC personnel continue to be available while TVA technicians acquire work experience. To help minimize personnel turnover, TVA has raised the salaries of RC technicians and supervisors.

- The Browns Ferry RC professional management staff has been strengthened considerably with the addition of a radiological health supervisor, a technical supervisor, and an engineer responsible for controlling radioactivity to ALARA (as low as reasonably achievable) levels.

The licensee developed postaccident sampling system (PASS) capability as a response to post-TMI (NUREG-0737) requirements (BFNPP, Section II.B.3). The NRC staff evaluated this interim capability in 1986 (Inspection Report 50-259, 260, 296/86-18, July 31, 1986) and determined that Unit 2 could restart once two followup items were resolved. The licensee stated that the first followup item (need to retrain laboratory personnel on PASS procedures) has been completed. The second item (need to evaluate the interim PASS under full-power operating conditions against NUREG-0737 criteria) will be completed after Unit 2 achieves full power. A permanent PASS will be installed before restart of Units 1 and 3 and during the next (Cycle 6) refueling outage for Unit 2.

4.10.2.2 Chemistry Improvement Plan

The SALP report (NRC letter, September 17, 1985) noted: "Management involvement and support were evident in the radiological protection area. Performance associated with the chemistry control program, however, was not as aggressive as that demonstrated in the radiological protection program."

In response to NRC comments as well as to internal TVA inspections and audits, Browns Ferry managers initiated the Chemistry Improvement Plan (CIP) to correct programmatic deficiencies. The root cause of the deficiencies was determined to be lack of management attention and response to correct the identified problems. Browns Ferry managers recognized that emphasis in the areas of staffing, training, and facility improvement was critical to the overall success of the program. The following 12 areas of improvement were established by categorizing the audit findings and inspections.

- staffing
- training
- facility
- quality assurance/quality control
- representative sample of the bulk stream
- complete revision and verification of all chemistry procedures
- instrument upgrade
- system chemistry
- environment monitoring and control
- data management
- chemical traffic control
- bulk chemical control

Additionally, the licensee initiated the following improvements in laboratory practice:

- Added requirements for supervisory review and prompt corrective action in case of degraded quality. All laboratory data and instrumentation are reviewed continuously by the shift supervisor and are reviewed frequently by the responsible engineer.

- As for standard sources used for data radiation efficiency calibration, data counting is no longer performed on the proportional counter. The alpha radiation measurement efficiency calibration procedures have been strengthened.

The licensee also introduced programs to reduce the volume of liquid and solid radwaste. To reduce the overall volume of solid radioactive waste, the licensee will minimize its generation by strengthening control over materials entering the regulated area, improving monitoring efforts and increasing employee awareness. Once the radwaste has been generated, effective volume-reduction techniques will be applied. Radwaste-reduction techniques, such as trash segregation and compaction, will be emphasized and reevaluated to pinpoint areas that require improvement. One major area for improvement relative to compaction is the anti-springback device system.

In an effort to reduce the overall liquid volume processed, the licensee has established a Radwaste Inleakage Reduction Program and has assigned an engineer to review this area of radwaste. To date this program has placed emphasis on the floor drain system because of the effect on the floor drain filter of the extensive use of resins. The floor drain system has shown a steady decrease in floor drain inleakage since November 1985. As floor drain inleakage is further reduced, more emphasis will be placed on the equipment drain system.

4.10.3 Conclusion

Improvements discussed above included management involvement in the program; increased staffing, training, and salaries, as well as facility improvements; implementation of NUREG-0737 requirements; and implementation of a program to reduce liquid and solid radwaste by introducing the anti-springback device. These actions strengthen the radiological and chemistry control. The staff concludes that TVA's improvement measures are acceptable to support Unit 2 restart.

5 EMPLOYEE CONCERNS

5.1 Introduction

This evaluation documents the staff's review of the Tennessee Valley Authority (TVA or the licensee) Employee Concern Program (ECP) which went into effect for all of TVA's nuclear power plants on February 1, 1986. The system for handling employee concerns was described in a report titled "TVA Employee Concern Program," which was submitted to the NRC by letter dated November 20, 1985. TVA modified this program by a February 11, 1986 letter. On May 2, 1986, TVA submitted a completely revised report, "Employee Concern Activities," which included a description of the overall ECP for TVA's Office of Nuclear Power (ONP). The program applies to TVA personnel involved in the nuclear program at all of TVA's plant sites and in the corporate offices. The procedure for implementing the ECP was submitted to the NRC by letter dated July 17, 1986; the procedure was subsequently revised up through July 1987.

The NRC staff reviewed the ECP during the periods April 8-11, 1986 (NRC Inspection Report 50-327, 328/86-29, May 15, 1986); June 10-12, 1986 (NRC Inspection Report 50-390, 391/86-15, July 7, 1986); April 6-May 5, 1987 (NRC Inspection Report 50-327, 328/87-24, June 4, 1987); and August 3-5, 1987 (reported herein). Subsequent revisions were reviewed when they were issued.

During June 1986, NRC notified the licensee that TVA employees had raised a number of significant safety concerns to the NRC (NRC letter, June 2, 1986). Included in the expressions of concerns to the NRC were the employees' fear of reprisal by TVA managers if employees raised their views on safety issues within TVA. The NRC expressed its disapproval to the licensee that the TVA policy and program for handling employee issues did not involve a feedback to managers of the types of issues that employees had raised. Further, the NRC believed the potential existed that other nuclear safety issues would remain unexpressed because employees feared reprisal from TVA managers. There were indications that the licensee's programs and policies were not being fully implemented and that the legal and policy protections against reprisal had not gained the full confidence of all TVA employees.

In response to the recognition that existing programs to identify and resolve concerns of employees were not fully effective, TVA awarded Quality Technology Company (QTC) a contract in May 1985 to develop and implement a program for conducting confidential interviews of all TVA employees associated with the Watts Bar Nuclear Plant. QTC also solicited concerns from TVA employees at the licensee's other nuclear sites. This program generated more than 5000 employee concerns which are addressed by an Employee Concern Special Program for which the staff has issued a separate safety evaluation. All employee concerns received on and after February 1, 1986 are handled within the framework of the new ECP which is the subject of this evaluation.

5.2 Evaluation

TVA has taken several actions that are intended to increase employee trust in nuclear management as well as to create an atmosphere that is conducive to quality. These actions include establishing a system to allow employees to express their concerns about quality to TVA nuclear management without fear of reprisal and with the assurance that their concerns will be addressed.

The staff evaluated these actions on the basis of the materials provided by TVA and staff onsite reviews completed over the past year. The ECP provides guidelines for receiving employee concerns and resolving these concerns for all TVA nuclear sites and offices. The staff evaluated the program, scope, organization, and resolution of concerns expressed within the framework of the ECP.

The staff has reviewed the description of the program submitted to the NRC on May 2, 1986. It has also reviewed the implementing procedures submitted on July 17, 1986 and subsequent revisions up through February 1989. Inspections were conducted to review the implementation of the program during the periods April 8-11, 1986 (NRC Inspection Report 50-327, 328/86-29, May 15, 1986); June 10-12, 1986 (NRC Inspection Report 50-390, 391/86-15, July 7, 1986); April 6-May 5, 1987 (NRC Inspection Report 50-327, 328/87-24, June 4, 1987); and August 3-5, 1987 (reported herein). The staff's evaluation of the ECP follows.

5.2.1 Objective

The objective of the licensee's ECP is to ensure that employees can express their concerns without fear of reprisal and that these concerns will receive prompt and effective action.

The program embodies several key fundamentals designed to support high standards of quality and safety in TVA nuclear activities. These include:

- providing for early identification of problems of employee/management relations within the line organization
- eliminating intimidation, harassment, or reprisal actions against employees for raising concerns
- focusing the responsibility for effective operation of the program in a single organization reporting through the Manager of Nuclear Human Resources to the Senior Vice President, Nuclear Power
- providing for confidentiality upon employee request
- developing improved communication between employees and supervisors
- encouraging employee participation in accomplishing program improvements
- encouraging the line organization to solve problems that exist within the line organization
- providing an independent communication channel through ECP site representatives within the line organization for employees to use for reporting concerns outside their work organization
- using the TVA Office of Inspector General as an outlet independent of TVA's Office of Nuclear Power

- utilizing standardized documentation, recordkeeping, trending, and a common database for all locations

The staff agrees that these are key fundamentals for achieving TVA's objectives regarding this program.

5.2.2 Scope

The scope of this program addresses employee concerns that were identified on and after February 1, 1986. The program applies to all nuclear activities. Intimidation, harassment, and wrongdoing are also addressed within the scope of this program.

The staff has reviewed the scope of the program and concludes that it is acceptable.

5.2.3 Organization

The Senior Vice President, Nuclear Power is responsible for all activities in TVA's Office of Nuclear Power (ONP) and establishes policy and general program direction for an effective Employee Concern Program.

The ECP manager, who reports directly through the Manager of Nuclear Human Resources to the Senior Vice President, Nuclear Power, is responsible for management and direction to accomplish the objectives of the overall program. The ONP manager promotes the ECP within the line organization and maintains relations with all levels of line management and with appropriate organizations outside ONP. The ECP manager's organization also provides a means for the receipt and resolution of employee concerns raised within the ECP manager's organization. A key element of the ECP is that an ECP site representative (ECP-SR) is physically located at each major nuclear-power-related location.

The ECP-SR is responsible for identifying and working with senior managers to correct situations brought to the ECP-SR's attention where employee/supervisor/manager communications or relationships fail to establish an environment for free expression of concern. The ECP-SR also serves as a recipient of concerns or differing employee views separate from the line organization in which the employee works. It is the ECP-SR's responsibility to evaluate the nature of the employee's concern and to channel efforts toward proper resolution of the concern. This may include encouraging resolution of the concern with line supervisors, evaluation of the concern by line organization staff at the site, evaluation of the concern by the ONP corporate staff, or evaluation of the concern by a third-party organization.

The staff has reviewed the organization of the program and concludes that it is acceptable.

5.2.4 Management of Employee Concerns

Four distinct steps are involved in managing employee concerns:

(1) receipt of the employee concern
(2) evaluation/investigation of the employee concern
(3) corrective action by the line organization
(4) providing feedback to the employee

Each of these steps is discussed below.

5.2.4.1 Receipt of the Employee Concern

An employee may bring his/her concern to the responsible supervisor or to the ECP site representative at the location where the employee is working. The ONP prefers employees to express concerns to their immediate supervisors. Concerns expressed directly to line managers are not documented in the ECP. An employee concern can be received by the ECP site representative by the following means:

- walk in (unscheduled interview)

- scheduled interview to obtain feedback regarding the ECP or as part of an investigation

- phone call

- referral from line organization

- exit interview of an employee terminated from his/her job or transferred to another job

- mail-in forms available at all major nuclear program locations

Employees may report their concerns to the TVA Office of Inspector General (OIG), independent of the line organization or the ECP Manager. Employees may also report concerns to NRC, to the Office of Safety and Health Administration, and to the U.S. Department of Labor.

The staff considers these methods of receiving employee concerns acceptable.

5.2.4.2 Evaluation/Investigation of the Employee Concern

Should a concern be brought directly to the attention of the line organization, the responsible line supervisor may utilize the normal line organizational units, including discussion with other levels of supervisors or managers to bring resolution. If either the supervisor or manager considers he/she lacks sufficient information or expertise to handle the concern, he/she can either obtain assistance from someone who knows more or who has more experience, or he/she can advise the employee that the concern may be taken to the ECP site representative for resolution.

The ECP site representative is responsible for evaluating and/or investigating all concerns brought to his/her attention using the site representative procedure outlined in a TVA submittal dated July 17, 1986. The basic elements of the procedure for resolving an employee concern consist of the following actions:

- preliminary evaluation of the concern

- evaluation/investigation of the concern or assignment of the concern to an organizational unit for evaluation or investigation

- review of the findings and/or results of investigations assigned to other organizations and accepting or rejecting the report or referring the report to the ECP manager for evaluation

5.2.4.2.1 Preliminary Evaluation of Concern

The ECP site representative will make a preliminary evaluation of the concern and perform the following classifications:

(1) Categorization of Employee Concerns

Each concern will be categorized into one of the nine categories listed below.

- quality assurance/quality control
- materials control
- management and personnel
- intimidation, harassment, and wrongdoing (I&H)
- operations
- welding
- construction
- industrial safety
- engineering.

The staff has reviewed the categorization and definitions of these categories and finds them acceptable. The same nine categories were used for resolving employee concerns expressed to QTC in the Employee Concern Special Program.

(2) Safety Classification

Each concern is classified in one of four categories listed below:

- Nuclear Safety Related--A nuclear safety-related concern is an employee concern that if substantiated could reduce the effectiveness of or eliminate the function performed within the framework of Critical Structures, Systems and Components Part I, Sections 1.3 and 1.7 of the Nuclear Quality Assurance Manual, or the plant Q-list.

- Industrial Safety Related--If the concern pertains to employee or general public health and safety issues and is not classified as nuclear safety related, the concern shall be classifed as industrial safety related.

- Safety Significant--A safety-significant concern is a nuclear safety-related or industrial safety-related concern that if valid could have an immediate effect on unit operability or could create a condition of imminent danger.

- Non-Safety Related--All other concerns are non-safety related.

The staff approved the criteria for classifying concerns as safety significant and non-safety related in its letter dated July 31, 1987. The staff reviewed the other two classifications (nuclear safety related and industrial safety related) and finds them acceptable.

(3) Generic Applicability

When a concern is classified as nuclear safety related, industrial safety related, or safety significant, a determination of generic applicability is performed in accordance with the site representative (SR) procedure.

The generic applicability determination is documented and becomes part of a record of the site representative notifying the other affected site(s).

The staff has reviewed the generic applicability criteria and procedures for addressing generically applicable concerns to other sites and finds them acceptable.

(4) Handling of Intimidation/Harassment Concerns

When a concern is categorized as an intimidation/harassment (I&H) concern, the following special handling by the site representative is required:

- The site representative notifies the ECP manager of the concern as soon as practical and transmits a short summary of the pre-evaluation to the ECP manager.

- The ECP manager determines whether to refer the I&H concern to the TVA Office of Inspector General (OIG) or to assign the investigation to the ECP staff. This determination will be made by considering such factors as the nature of the concern, the complexity, the degree of ONP management involvement in the concern, and the investigative experience or expertise required. The determination may also involve discussions with the OIG.

- If the concern is referred to the OIG, the site representative will transmit a copy of the complete concern file to the OIG. The site representative will notify the concerned individual that the concern has been referred to the OIG and will explain and/or clarify any confidentiality provisions as necessary.

- If an I&H concern is being referred to the OIG, the site representative will review the concern to determine if it contains issues outside the scope of the OIG referral. For example, an I&H concern that is centered around a supervisor/employee relationship could also contain problems related to nuclear safety. Before this concern is transmitted to the OIG, the site representative will identify the technical safety issues and will address them as separate concerns. The transmittal to the OIG identifies any parts of the concern that are to be investigated within the framework of the ECP. Applicable reports issued through the ECP will reference the concern as transmitted to the OIG.

The staff has reviewed the procedure and finds it acceptable.

(5) Alleged Violations of Criminal Law

If during the receipt or pre-evaluation of a concern it appears that criminal laws may have been violated, the site representative shall notify the ECP manager as soon as practical. These alleged violations are to be handled in the same general manner as I&H concerns and in most cases will be referred to the OIG for investigation.

(6) Differing Professional Opinions

A differing professional opinion (DPO) may not violate any regulation or safety concern but may simply represent a way of doing things differently. Also, the role of the site representative for handling a DPO is more one of conciliation than of investigation.

A DPO is defined as: (1) a professional opinion that differs from a TVA manager's decision, a stated position, or an established agency practice or policy; or (2) an employee's opinion that a previously stated concern has not been adequately considered; or (3) an employee's opinion that a decision, position, or practice, if not adopted, may endanger the safe operation of a nuclear power facility.

When an ECP site representative receives an employee concern that is a DPO, the ECP-SR notifies the ECP manager as soon as practical. The ECP site representative is required to inform the concerned individual of his/her right to make his/her views known formally to his/her Vice-President and the Senior Vice-President.

The ECP manager ensures that appropriate measures are taken for resolving the concern and that the existence of the issue is brought to the attention of the ONP manager and to the attention of any other individuals whom the ONP manager designates.

The staff finds the definition of a DPO and the approach for handling DPOs acceptable.

(7) Offshoot Concerns

If during the course of an evaluation/investigation of a concern, the ECP site representative or evaluator becomes aware of other unidentified concerns outside the scope of the concern being investigated, the new concern is to be evaluated in a manner similar to the evaluation/investigation of any other concern.

(8) Immediate Stop Work Order

Should the site representative determine, in accordance with TVA's Nuclear Quality Assurance Manual (NQAM), Part I, Section 2.16, "Corrective Action" (May 16, 1988), that a potential threat exists to the health and safety of the public or to TVA employees, the division director, site director, or senior shift manager will be informed as soon as practical. The site representative will inform the ECP manager of this action as soon as possible.

The staff agrees with the stop work authority provided to the ECP site representative.

(9) Other Significant Actions

The site representative takes immediate action to notify line management if an investigation indicates the potential for a condition adverse to quality or a concern that should be reported to NRC in accordance with 10 CFR 50.55(e), 10 CFR Part 21, 10 CFR 50.72, or 10 CFR 50.73.

Before performing an investigation or assigning the concern to investigation, the SR may determine if the concern has previously been identified or investigated. The results of previous investigations may be used to resolve subsequent concerns.

The staff finds these actions acceptable.

5.2.4.2.2 Assignment of Concern for Investigation

On the basis of a preliminary evaluation of a concern, the site representative is to either resolve the concern by completing any needed investigation or is to determine an appropriate organizational unit to handle the investigation.

5.2.4.2.3 Review of Investigative Reports Prepared by Other Organizations

The ECP site representative is responsible for reviewing all investigative reports, including those prepared by other organizations. The site representative is to specifically examine the investigation report to determine if it addresses the concern and to ensure that findings are supported by facts. When the site representative is satisfied that the investigation report is adequate, the report is submitted to the ECP manager for approval. Once the report has been approved, the ECP manager sends it to appropriate line management.

During April 8-11, 1986; June 10-12, 1986; and April 6-May 5, 1987 inspections (NRC Inspection Reports 50-327, 328/86-29, May 15, 1986; 50-390, 391/86-15, July 7, 1986; and 50-327, 328/87-24, June 4, 1987, respectively), the staff reviewed reports investigating concerns raised by Sequoyah and Watts Bar employees. When the ECP was initiated, the case files were incomplete and the investigative reports were not "stand alone" reports. TVA, in response to staff concerns, upgraded its reports and on review by the staff during the April 6-May 5, 1987 inspection it was concluded that the upgraded reports were acceptable. The reports on I&H-related issues required better documentation; during its August 3-5, 1987 audit (reported herein), the staff found that the documentation was better.

5.2.4.3 Corrective Action by the Line Organization

Upon receipt of the investigation report, the line organization is responsible for implementing corrective actions. Any disagreements between the line organization and ECP manager regarding findings or corrective actions will be escalated through the Manager of Nuclear Human Resources to the Senior Vice President, Nuclear Power or designee for final resolution. Additionally, the line organization will review the report and its findings for generic applicability and reportability to NRC, and will determine as applicable the need for immediate corrective actions such as Stop Work Orders.

The staff finds this approach acceptable and will periodically monitor the implementation of these corrective actions.

5.2.4.4 Feedback to the Employee

The ECP site representative is responsible for providing the concerned individual (CI) with the results of the investigation. This is to be done by arranging a meeting with the employee. At the meeting, the findings of the investigation and the designated corrective action(s) are presented. Employees who have raised concerns that were not resolved before the employee left TVA are notified of the resolution. The CI is provided with the results of the investigation report. If the CI is not satisfied with how the concern was resolved, the ECP site representative is to discuss with the CI other means of having the concern evaluated. The employee may desire to have the concern examined by the OIG. If so, the ECP site representative will provide the OIG with the file on the employee's concern.

The staff finds the method for feedback acceptable and will monitor the progress by assessing allegations brought directly to NRC, by reviewing the results of employee surveys conducted by TVA, and by contacting TVA employees directly.

5.2.5 Training and Orientation of Employees

All employees in ONP are provided training or orientation on the ECP. This includes various training courses, orientation handouts for new employees, and periodically including material about the ECP with the employee's paycheck.

Special training regarding this program is provided to managers and other employees; in this special training the ECP manager and/or the site representatives identify situations that indicate unsatisfactory employee/supervisor contact.

The staff reviewed training documents and training session attendance sheets and finds them acceptable.

5.2.6 Trending

The ECP manager periodically meets with the OIG to discuss trends involving employees using the OIG system instead of the ECP system. Information relating to root causes of undesirable trends as determined by the OIG will be evaluated by the ECP manager to design corrective action in the ECP system.

Trends are also established by reviewing the database maintained by TVA and reviewing certain categories such as number of concerns expressed within the framework of the ECP, number of employees requesting confidentiality, number of employees not expressing specific concerns because they fear reprisal, and other similar indicators.

The staff reviewed these data during the August 3-5, 1987 inspection and found that generally the number of concerns expressed to the ECP has leveled off. Recently, employees have been expressing concerns without asking for confidentiality and the number of concerns expressed by departing employees has been very low. On the basis of its review, the staff finds the trending activity acceptable.

5.3 Conclusion

On May 2, 1986, the staff reviewed the Employee Concern Program; on July 17, 1986, the staff reviewed the procedures for handling employee concerns. Subsequent revisions in the program and procedures (up through February 1989) were also reviewed. The staff audited the implementation of the program including review of investigative reports for Sequoyah and Watts Bar during four inspections in 1986 and 1987. On the basis of these reviews, the staff concludes that the ECP established on February 1, 1986 is an acceptable program for handling employee concerns.

APPENDIX A

LIST OF NRC CONTRIBUTORS

Name	Office
C. Brooks	Special Projects, TVA Inspections Programs Division
P. Castleman	Nuclear Reactor Regulation, TVA Technical Programs Division
P. Cortland	Nuclear Reactor Regulation, TVA Projects Division
M. Fields	Special Projects, TVA Projects Division
H. Garg	Nuclear Reactor Regulation, TVA Technical Programs Division
G. Gears	Nuclear Reactor Regulation, TVA Projects Division
G. Georgiev	Nuclear Reactor Regulation, TVA Technical Programs Division
J. Gilray	Region II
P. Hearn	Nuclear Reactor Regulation, TVA Technical Programs Division
G. Hubbard	Nuclear Reactor Regulation, TVA Technical Programs Division
S. B. Kim	Nuclear Reactor Regulation, TVA Technical Programs Division
W. Little	Nuclear Reactor Regulation, TVA Inspections Programs Division
D. Moran	Nuclear Reactor Regulation, TVA Projects Division
F. Paulitz	Nuclear Reactor Regulation, TVA Projects Division
J. Rotella	Nuclear Reactor Regulation, TVA Technical Programs Division
B. K. Singh	Special Projects, TVA Projects Division
D. Smith	Nuclear Reactor Regulation, TVA Technical Programs Division
W. Tobin	Region II
J. Watt	Nuclear Reactor Regulation, TVA Technical Programs Division
R. Wescott	Nuclear Reactor Regulation, TVA Technical Programs Division

APPENDIX B

REFERENCES

American National Standards Association, Code for Pressure Piping, Standard B31.1-1967, "Power Piping."

Electric Power Research Institute, Report No. NP-3944, "Erosion-Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Program Guidelines," Final Report, April 1985.

Institute of Electrical and Electronics Engineers, Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

NRC Letters

U.S. Nuclear Regulatory Commisison, November 13, 1979, memorandum from H. Denton to V. Stello transmitting "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors."

---, August 8, 1985, letter from D. B. Vassallo to H. G. Parris (TVA), regarding equipment qualification of electrical equipment important to safety.

---, September 17, 1985, letter from W. J. Dircks to C. H. Dean, Jr. (TVA), transmitting 5th SALP review and request to provide information pursuant to 10 CFR 50.54(f).

---, March 26, 1986, letter from M. Grotenhuis to S. A. White (TVA), responding to Generic Letter 84-11 regarding reinspection of stainless steel piping.

---, June 2, 1986, letter from L. Zech to C. H. Dean, Jr. (TVA), regarding a significant number of safety concerns relayed to staff by TVA employees.

---, August 1, 1986, letter from B. K. Grimes to R. L. Gridley (TVA), requesting information on Topical Report TVA-TR75-1A.

---, October 3, 1986, letter from D. R. Muller to Manager, Nuclear Power (TVA), forwarding request for additional information on welding project reinspection.

---, January 30, 1987, letter from B. K. Grimes to S. A. White (TVA), approving QA topical report TVA-TR75-1A, Revision 9.

---, July 31, 1987, letter from J. Zwolinski to S. A. White (TVA), regarding nuclear safety-related employee concerns.

---, June 23, 1988, letter from S. Black to S. A. White (TVA), forwarding staff evaluation of TVA compliance with RG 1.97.

---, September 29, 1988, letter from S. Black to S. A. White (TVA), asking TVA to address issues in surveillance program.

---, December 8, 1988, letter from S. C. Black to O. D. Kingsley (TVA), regarding Appendix R safe shutdown system analysis.

---, December 14, 1988, letter from S. Black to O. D. Kingsley, Jr. (TVA), stating NRC position on NFPA deviations.

---, January 3, 1989, letter from S. Black to O. D. Kingsley, Jr. (TVA), regarding open items on plant surveillance safety evaluation.

NRC Meeting Summaries

U.S. Nuclear Regulatory Commission, May 24, 1988, NRC summary of April 26, 1988 meeting.

---, July 24, 1988, NRC summary of June 21, 1988 meeting.

NRC Reports

U.S. Nuclear Regulatory Commission, NUREG-0061, "Safety Evaluation Report Related to Operation of Browns Ferry Units 1 and 2, Following the March 22, 1975 Fire," March 1976.

---, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," November 1979.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, July 1981.

---, NUREG-1232, Volume 1, "Safety Evaluation Report on Tennessee Valley Authority," July 1987.

NRC Generic Letters

U.S. Nuclear Regulatory Commission, Generic Letter 84-01, "NRC Use of the Terms 'Important to Safety' and 'Safety-Related,'" January 5, 1984.

---, Generic Letter 84-11, "Licensing Actions Proposed to Implement BWR Pipe Crack Reinspection Program," April 19, 1984.

---, Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 25, 1988.

NRC Miscellaneous

U.S. Nuclear Regulatory Commission, IE Bulletin 79-18, "Audibility Problem Encountered on Evacuation of Personnel From High-Noise Areas," August 1979.

---, IE Bulletin 80-07, "BWR Jet Pump Assembly Failure," April 4, 1980.

---, Information Notice 88-03, "Cracks in Shroud Support Access Hole Cover Welds," February 2, 1988.

---, NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," July 9, 1987.

TVA Letters

Tennessee Valley Authority, April 29, 1976, letter from J. E. Gilleland to C. J. Heltemes (NRC), transmitting QA Topical Report.

---, October 3, 1980, letter from L. M. Mills to J. P. O'Reilly (NRC), forwarding response to IE Bulletin 80-07.

---, April 30, 1984, letter from L. M. Mills to H. R. Denton (NRC), regarding detailed evaluation of RG 1.97.

---, May 7, 1985, letter from J. A. Domer to D. B. Vassallo (NRC), regarding conformance with RG 1.97, Revision 2.

---, November 1, 1985, letter from C. H. Dean, Jr. to W. J. Dircks (NRC), forwarding Corporate Nuclear Performance Plan (Volume 1) and Sequoyah Nuclear Performance Plan (Volume 2)

---, November 20, 1985, letter from H. G. Parris to W. J. Dircks (NRC), forwarding "TVA Employee Concern Program."

---, January 31, 1986, letter from R. Gridley to D. R. Muller (NRC), transmitting Appendix R analysis.

---, February 11, 1986, letter from S. A. White to V. Stello (NRC), forwarding "TVA Employee Concern Program" and methodology.

---, March 10, 1986, letter from S. A. White to N. Palladino (NRC), forwarding revised Corporate Nuclear Performance Plan.

---, May 1, 1986, letter from R. Gridley to N. Grace (NRC), forwarding revised QA Topical Report TVA-TR75-1A.

---, May 2, 1986, letter from S. A. White to V. Stello (NRC), forwarding "Employee Concern Activities."

---, July 17, 1986, letter from S. A. White to L. Zech (NRC), forwarding Sequoyah Nuclear Performance Plan, Revision 1.

---, July 31, 1986, letter from S. A. White to N. Palladino (NRC), forwarding Corporate Nuclear Performance Plan, Revision 2.

---, August 28, 1986, letter from S. A. White to L. Zech (NRC), forwarding Browns Ferry Nuclear Performance Plan, Revision 0.

---, October 7, 1986, letter from R. Gridley to G. G. Zech (NRC), describing proposed restart test program.

---, November, 14, 1986, letter from C. C. Mason to B. K. Grimes (NRC), forwarding response to August 1, 1986 request for information about Topical Report TVA-TR75-1A.

---, November, 21, 1986, letter from J. D. Wolcott to M. Grotenhuis (NRC), updating information pertaining to 10 CFR 50, Appendix R.

---, December 4, 1986, letter from C. C. Mason to L. Zech (NRC), forwarding Corporate Nuclear Performance Plan, Revision 3.

---, March 13, 1987, letter from R. Gridley to S. Ebneter (NRC), forwarding Browns Ferry Design Baseline and Verification Program, Revision 0.

---, March 26, 1987, letter from S. A. White to S. Ebneter (NRC), forwarding Corporate Nuclear Performance Plan, Revision 4.

---, July 1, 1987, letter from S. A. White to NRC Document Control Desk, forwarding Browns Ferry Nuclear Performance Plan, Revision 1.

---, July 10, 1987, letter from S. A. White to NRC Document Control Desk, forwarding Browns Ferry Design Baseline and Verification Program, Revision 2.

---, July 13, 1987, letter from S. A. White to NRC Document Control Desk, regarding Browns Ferry Nuclear Performance Plan, Revision 1.

---, September 18, 1987, letter from R. Gridley to NRC Document Control Desk, responding to NRC Bulletin 87-01.

---, December 10, 1987, letter from S. A. White to NRC Document Control Desk, forwarding Corporate Nuclear Performance Plan, Revision 5.

---, March 25, 1988, letter from R. Gridley to NRC Document Control Desk, forwarding Browns Ferry Design Baseline and Verification Program, Revision 4.

---, April 4, 1988, letter from R. Gridley to NRC Document Control Desk, forwarding Fire Protection Plan.

---, August 3, 1988, letter from R. Gridley to NRC Document Control Desk, forwarding response to request for information on National Fire Protection Association Code deviations.

---, August 23, 1988, letter from R. Gridley to NRC Document Control Desk, regarding compliance with RG 1.97.

---, September 23, 1988, letter from R. Gridley to NRC Document Control Desk, responding to request for additional information and overload protection of motor control center circuits.

---, September 28, 1988, letter from R. Gridley to NRC Document Control Desk, regarding plant security.

---, October 24, 1988, letter from S. A. White to NRC Document Control Desk, forwarding Browns Ferry Nuclear Performance Plan, Revision 2.

---, October 31, 1988, letter from R. Gridley to NRC Document Control Desk, responding to plant surveillance program open items.

---, December 20, 1988, letter from R. Gridley to NRC Document Control Desk, forwarding Site Director Standard Practice procedure SDSP 16.8, Revision 0, "Fuse Control."

---, January 12, 1989, letter from R. Gridley to NRC Document Control Desk, committing to adhering to the guidance in Generic Letter 88-01.

TVA Document

Tennessee Valley Authority, Nuclear Quality Assurance Manual, Part I, Section 2.16, "Corrective Action," May 16, 1988.

APPENDIX C

TVA RESPONSES PERTAINING TO BROWNS FERRY 10 CFR 50.54(f) CONCERNS

The NRC's September 17, 1985 50.54(f) letter requested specific information concerning the Browns Ferry Nuclear Plant. The Nuclear Performance Plan, Volume 3, specifically addresses TVA's responses to this letter as it pertains to Browns Ferry Unit 2. The following provides each 50.54(f) concern relating to Browns Ferry with corresponding reference to TVA responses as provided in Volume 3, Nuclear Performance Plan. The appropriate reference to the staff's evaluation of these responses is also provided (NUREG-1232, Volume 3).

1. "Describe the site management changes made subsequent to the SALP [systematic assessment of licensee performance] period to strengthen the regulatory performance at Browns Ferry, including experience and qualifications of newly assigned managers."

   Nuclear Performance Plan, Volume 3: Section II.1
   NUREG-1232, Volume 3: Section 4.2.

2. "Provide a detailed description of the operational readiness plan developed by you to assess the readiness for resuming operation of any of the Browns Ferry units. If this plan does not address all Category 3 areas in the attached SALP report, then your submittal should address these areas. Additionally, because the Regulatory Performance Improvement Program has proven to be ineffective in improving performance, provide an evaluation of the cause of the lack of positive results. Further, provide your rationale for expecting any different results from the Operational Readiness Review."

   Nuclear Performance Plan, Volume 3: Section III.9
   NUREG-1232, Volume 3: Section 4.1

3. "Provide: (a) a detailed description of the Maintenance Improvement Program including improvements for planning and scheduling maintenance activities and (b) a report on progress and results achieved in implementing this program."

   Nuclear Performance Plan, Volume 3: Section II.4
   NUREG-1232, Volume 3: Section 4.5

4. "Provide an updated integrated schedule for all NRC-required plant modifications and improvement modifications which may impact the former."

   Nuclear Performance Plan, Volume 3: Section II.9
   NUREG-1232, Volume 3: Section 4.1

TVA has indicated in Volume 3 that the integrated schedule approach has been superseded by the Volume 3 scheduling effort to support Unit 2 restart. Therefore, the original integrated scheduling effort (submittals of August 14, 1984; September 21, 1984; and April 12, 1985) is no longer applicable. The staff finds this approach responsive to the staff's concern.

5. "Provide analyses that demonstrate that seismic supports with identified deficiencies comply with the seismic design criteria or provide technical justification for interim operation and a schedule for completing any necessary modifications."

   Nuclear Performance Plan, Volume 3, Section III.3
   NUREG-1232, Volume 3: Section 2.2

6. "Provide a detailed description of the design control survey which you are conducting, including a discussion of any generic implications on plant design."

   Nuclear Performance Plan, Volume 3: Section III.2
   NUREG-1232, Volume 3: Section 2.1

7. "Provide your evaluation and proposed disposition of recommendations by contractors, such as General Electric, that have evaluated modifications to Browns Ferry safety systems."

   Nuclear Performance Plan, Volume 3: Appendix B
   NUREG-1232, Volume 3: Section 2.1

The staff will review the licensee's response to this concern as part of its Vertical Slice Team Inspection. The results of inspection will be contained in the team inspection. A brief summary of the team's findings will be contained in a future supplement to NUREG-1232, Volume 3.

8. "Provide a detailed description of (a) the program being implemented to demonstrate compliance with 10 CFR 50.49 and (b) the long-term program to assure continued compliance with regulations. Affirm that the list of equipment required to meet 10 CFR 50.49 is complete."

   Nuclear Performance Plan, Volume 3: Section III.1
   NUREG-1232, Volume 3: Section 3.2

9. "Provide an evaluation of the need to establish an onsite independent safety engineering group to review operational events as they occur."

   Nuclear Performance Plan, Volume 3: Section II.1.2.10
   NUREG-1232, Volume 3: Section 4.2

10. "Provide responses to our requests for additional information and responses to our comments on proposed licensing actions as requested in letters from D. B. Vassallo to H. G. Parris dated November 26, 1984; June 27, 1985; July 22, 1985; July 26, 1985; August 9, 1985; and August 22, 1985."

Nuclear Performance Plan, Volume 3: Appendix A and Appendix E

The staff finds the licensee's responses to this item to be acceptable.

11. "In addition to meeting the requirements of Appendix R, provide an evaluation of your progress and results achieved in implementing an effective fire protection program that conforms to general industry practice and the fire protection standards promulgated by the National Fire Protection Association (NFPA). Specific weaknesses in your fire protection program have been identified in the attached SALP report and in your own audits."

    Nuclear Performance Plan, Volume 3: Section III.5
    NUREG-1232, Volume 3: Section 3.1