EMPLOYEE CONCERNS

SPECIAL PROGRAM

VOLUME 2
ENGINEERING CATEGORY

SUBCATEGORY REPORT 26000
FLUSHING AND PIPING VALVE DESIGN

UPDATED

TVA

NUCLEAR POWER

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TVA EMPLOYEE CONCERNS SPECIAL PROGRAM

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1. Revised to incorporate SRP comments and to incorporate approved corrective action plans.

2. Revised to incorporate SRP comments.

3. Revised to incorporate TAS comments and to add Attachment C.

4. Revised to incorporate SRP and TAS comments.

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EXECUTIVE SUMMARY

Subcategory Report 26000, Flushing and Piping Valve Design, summarizes and evaluates the results of 28 Employee Concern Special Program element evaluations containing 42 separate issues. Employee concerns evaluated address such diverse activities as material selection, code requirements, design philosophy, inconsistent application of design modifications, operational problems, and vendor errors. The concerned employees generally cited a perceived deficiency or inadequacy in the design and construction of the plant fluid systems. Of the 42 findings (from Table 2), 35 were found to require no corrective action. In four of the issues evaluated in this report, the perception was substantiated and corrective action required. Of these, three were in the process of correction prior to this evaluation. In addition, three peripheral issues were identified which require corrective action.

The issues that were substantiated include requirements for documentation to demonstrate adequacy of procured components, piping relocation and insulation requirements, and, in three of the four cases, additional analysis to assess possible modification requirements.

Of the four requirements for corrective action identified by this evaluation, two could potentially require a change in documentation, design margin, or hardware at BFN. One involves piping insulation on austenitic stainless-steel safety-related systems which could contain chemicals that could promote intergranular stress corrosion cracking of the piping, which could potentially lead ultimately to piping failure. The other pertains to the use, in some cases, of piping wall thickness requirement formulas which result in values less (thinner) than required by applicable piping codes. In both cases, further evaluation is warranted.

Although the employee concerns and other issues found during the evaluation did identify some valid problems that require resolution, the relatively small number of negative findings are so diverse in content and their causes are so random that no focused judgment of collective significance is warranted. It is reasonable to conclude that, with respect to the issues evaluated in this report, piping and valve design does not constitute a significant problem for Watts Bar, Sequoyah, Browns Ferry, and Bellefonte nuclear power plants. The incidence of these random discrepancies should be further reduced by implementation of TVA's Corporate Nuclear Performance Plan.

Neither did the grouped evaluation at the subcategory level find any new or broader issues requiring attention. The causes identified and other evaluation results will be reexamined from a wider perspective during the engineering category evaluation.

Preface

This subcategory report is one of a series of reports prepared for the Employee Concerns Special Program (ECSP) of the Tennessee Valley Authority (TVA). The ECSP and the organization which carried out the program, the Employee Concerns Task Group (ECTG), were established by TVA's Manager of Nuclear Power to evaluate and report on those Office of Nuclear Power (ONP) employee concerns filed before February 1, 1986. Concerns filed after that date are handled by the ongoing ONP Employee Concerns Program (ECP).

The ECSP addressed over 5800 employee concerns. Each of the concerns was a formal, written description of a circumstance or circumstances that an employee thought was unsafe, unjust, inefficient, or inappropriate. The mission of the Employee Concerns Special Program was to thoroughly investigate all issues presented in the concerns and to report the results of those investigations in a form accessible to ONP employees, the NRC, and the general public. The results of these investigations are communicated by four levels of ECSP reports: element, subcategory, category, and final.

Element reports, the lowest reporting level, will be published only for those concerns directly affecting the restart of Sequoyah Nuclear Plant's reactor unit 2. An element consists of one or more closely related issues. An issue is a potential problem identified by ECTG during the evaluation process as having been raised in one or more concerns. For efficient handling, what appeared to be similar concerns were grouped into elements early in the program, but issue definitions emerged from the evaluation process itself. Consequently, some elements did include only one issue, but often the ECTG evaluation found more than one issue per element.

Subcategory reports summarize the evaluation of a number of elements. However, the subcategory report does more than collect element level evaluations. The subcategory level overview of element findings leads to an integration of information that cannot take place at the element level. This integration of information reveals the extent to which problems overlap more than one element and will therefore require corrective action for underlying causes not fully apparent at the element level.

To make the subcategory reports easier to understand, three items have been placed at the front of each report: a preface, a glossary of the terminology unique to ECSP reports, and a list of acronyms.

Additionally, at the end of each subcategory report will be a Subcategory Summary Table that includes the concern numbers; identifies other subcategories that share a concern; designates nuclear safety-related, safety significant, or non-safety related concerns; designates generic applicability; and briefly states each concern.

Either the Subcategory Summary Table or another attachment or a combination of the two will enable the reader to find the report section or sections in which the issue raised by the concern is evaluated.

The subcategories are themselves summarized in a series of eight category reports. Each category report reviews the major findings and collective significance of the subcategory reports in one of the following areas:

- management and personnel relations
- industrial safety
- construction
- material control
- operations
- quality assurance/quality control
- welding
- engineering

A separate report on employee concerns dealing with specific contentions of intimidation, harassment, and wrongdoing will be released by the TVA Office of the Inspector General.

Just as the subcategory reports integrate the information collected at the element level, the category reports integrate the information assembled in all the subcategory reports within the category, addressing particularly the underlying causes of those problems that run across more than one subcategory.

A final report will integrate and assess the information collected by all of the lower level reports prepared for the ECSP, including the Inspector General's report.

For more detail on the methods by which ECTG employee concerns were evaluated and reported, consult the Tennessee Valley Authority Employee Concerns Task Group Program Manual. The Manual spells out the program's objectives, scope, organization, and responsibilities. It also specifies the procedures that were followed in the investigation, reporting, and closeout of the issues raised by employee concerns.

ECSP GLOSSARY OF REPORT TERMS*

classification of evaluated issues the evaluation of an issue leads to one of the following determinations:

Class A: Issue cannot be verified as factual

Class B: Issue is factually accurate, but what is described is not a problem (i.e., not a condition requiring corrective action)

Class C: Issue is factual and identifies a problem, but corrective action for the problem was initiated before the evaluation of the issue was undertaken

Class D: Issue is factual and presents a problem for which corrective action has been, or is being, taken as a result of an evaluation

Class E: A problem, requiring corrective action, which was not identified by an employee concern, but was revealed during the ECTG evaluation of an issue raised by an employee concern.

collective significance an analysis which determines the importance and consequences of the findings in a particular ECSP report by putting those findings in the proper perspective.

concern (see "employee concern")

corrective action steps taken to fix specific deficiencies or discrepancies revealed by a negative finding and, when necessary, to correct causes in order to prevent recurrence.

criterion (plural: criteria) a basis for defining a performance, behavior, or quality which ONP imposes on itself (see also "requirement").

element or element report an optional level of ECSP report, below the subcategory level, that deals with one or more issues.

employee concern a formal, written description of a circumstance or circumstances that an employee thinks unsafe, unjust, inefficient or inappropriate; usually documented on a K-form or a form equivalent to the K-form.

evaluator(s) the individual(s) assigned the responsibility to assess a specific grouping of employee concerns.

findings includes both statements of fact and the judgments made about those facts during the evaluation process; negative findings require corrective action.

issue a potential problem, as interpreted by the ECTG during the evaluation process, raised in one or more concerns.

K-form (see "employee concern")

requirement a standard of performance, behavior, or quality on which an evaluation judgment or decision may be based.

root cause the underlying reason for a problem.

*Terms essential to the program but which require detailed definition have been defined in the ECTG Procedure Manual (e.g., generic, specific, nuclear safety-related, unreviewed safety-significant question).

Acronyms

AI	Administrative Instruction
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BFN	Browns Ferry Nuclear Plant
BLN	Bellefonte Nuclear Plant
CAQ	Condition Adverse to Quality
CAR	Corrective Action Report
CATD	Corrective Action Tracking Document
CCTS	Corporate Commitment Tracking System
CEG-H	Category Evaluation Group Head
CFR	Code of Federal Regulations
CI	Concerned Individual
CMTR	Certified Material Test Report
COC	Certificate of Conformance/Compliance
DCR	Design Change Request
DNC	Division of Nuclear Construction (see also NU CON)

DNE	Division of Nuclear Engineering
DNQA	Division of Nuclear Quality Assurance
DNT	Division of Nuclear Training
DOE	Department of Energy
DPO	Division Personnel Officer
DR	Discrepancy Report or Deviation Report
ECN	Engineering Change Notice
ECP	Employee Concerns Program
ECP-SR	Employee Concerns Program-Site Representative
ECSP	Employee Concerns Special Program
ECTG	Employee Concerns Task Group
EEOC	Equal Employment Opportunity Commission
EQ	Environmental Qualification
EMRT	Emergency Medical Response Team
EN DES	Engineering Design
ERT	Employee Response Team or Emergency Response Team
FCR	Field Change Request
FSAR	Final Safety Analysis Report
FY	Fiscal Year
GET	General Employee Training
HCI	Hazard Control Instruction
HVAC	Heating, Ventilating, Air Conditioning
II	Installation Instruction
INPO	Institute of Nuclear Power Operations
IRN	Inspection Rejection Notice

L/R	Labor Relations Staff
M&AI	Modifications and Additions Instruction
MI	Maintenance Instruction
MSPB	Merit Systems Protection Board
MT	Magnetic Particle Testing
NCR	Nonconforming Condition Report
NDE	Nondestructive Examination
NPP	Nuclear Performance Plan
NPS	Non-plant Specific or Nuclear Procedures System
NQAM	Nuclear Quality Assurance Manual
NRC	Nuclear Regulatory Commission
NSB	Nuclear Services Branch
NSRS	Nuclear Safety Review Staff
NU CON	Division of Nuclear Construction (obsolete abbreviation, see DNC)
NUMARC	Nuclear Utility Management and Resources Committee
OSHA	Occupational Safety and Health Administration (or Act)
ONP	Office of Nuclear Power
OWCP	Office of Workers Compensation Program
PHR	Personal History Record
PT	Liquid Penetrant Testing
QA	Quality Assurance
QAP	Quality Assurance Procedures
QC	Quality Control
QCI	Quality Control Instruction

QCP	Quality Control Procedure
QTC	Quality Technology Company
RIF	Reduction in Force
RT	Radiographic Testing
SQN	Sequoyah Nuclear Plant
SI	Surveillance Instruction
SOP	Standard Operating Procedure
SRP	Senior Review Panel
SWEC	Stone and Webster Engineering Corporation
TAS	Technical Assistance Staff
T&L	Trades and Labor
TVA	Tennessee Valley Authority
TVTLC	Tennessee Valley Trades and Labor Council
UT	Ultrasonic Testing
VT	Visual Testing
WBECSP	Watts Bar Employee Concern Special Program
WBN	Watts Bar Nuclear Plant
WR	Work Request or Work Rules
WP	Workplans

CONTENTS

TABLES

1. INTRODUCTION

Subcategory Report 26000 summarizes and evaluates the results of 28 Employee Concern Special Program (ECSP) element evaluations related to piping and valve design. These evaluations had previously been identified as series 21400 and 23200, but because they treat generically similar elements, they have been combined into one subcategory report to facilitate the evaluation review. The concerns noted in these elements address engineering activities associated with material selection, code application, design philosophy, design modification, operations, startup testing, and procurement receipt inspection, as related to the design, purchase, construction, and operation of fluid piping systems.

The employee concerns form the basis for the element evaluations and are listed by element number in Attachment A. The plant location where the concern was originally identified and the applicability of the concern to other TVA nuclear plants are also shown.

The evaluations are summarized in the balance of this report as follows:

- Section 2 -- summarizes, by element, the issues stated or implied in the employee concerns and addresses determinations of generic applicability.
- Section 3 -- outlines the process followed for the subcategory evaluation and cites documents reviewed.
- Section 4 -- provides the discussion, by element, which forms the basis of the evaluation findings and identifies the negative findings that must be resolved.
- Section 5 -- highlights the corrective actions required for resolution of the negative findings cited in Section 4 and relates them to element and to plant site.
- Section 6 -- identifies causes of the negative findings.
- Section 7 -- assesses the significance of the negative findings.
- Attachment A -- lists, by element, each employee concern evaluated in the subcategory. The concern number is given along with notation of any other element or category with which the concern is shared, the plant sites to which it could be applicable are noted, the concern is quoted as received by TVA, and is characterized as safety related, not safety related, or safety significant.

- o Attachment B -- contains a summary of the element-level evaluations. Each issue is listed, by element number and plant, opposite its corresponding findings and corrective actions. The reader may trace a concern from Attachment A to an issue in Attachment B by using the element number and applicable plant. The reader may relate a corrective action description in Attachment B to causes and significance in Table 3 by using the CATD number which appears in Attachment B.

  The term "Peripheral finding" in the issue column refers to a finding that occurred during the course of evaluating a concern but did not stem directly from a employee concern. These are classified as "E" in Tables 1 and 2 of this report.

- o Attachment C -- contains the references cited in the text

2. SUMMARY OF ISSUES

The employee concerns listed in Attachment A for each element and plant site have been examined, and the potential problems raised by the 18 concerns have been identified as 42 separate issues. Evaluation of these issues is presented in 28 element evaluations. The issues evaluated under this subcategory, grouped by element, are summarized below.

214.0, Flushing Particles - Postconstruction checks for particulate matter after piping system flushing are inadequate. Additionally, cleanliness testing as carried out on process piping was not conducted on instrument sensing lines. This issue was raised at Watts Bar and was also evaluated at Bellefonte as construction has not been completed at these two plant sites. For operational sites, this issue was not relevant.

232.1, Accumulator Piping Size Change - Watts Bar unit 2 accumulator piping modifications, performed because of errors found in the original plant calculations, were not carried out on unit 1. This issue was not evaluated for SQN because similar piping modifications were not made there, nor was it evaluated at BFN or BLN as neither plant incorporates the Upper Head Injection (UHI) system.

232.2, Carbon Steel vs Stainless Steel Drainage Piping - Reactor Building drain piping is carbon steel; it should be stainless steel. As QTC identified the area of concern to be in the raceway area, this issue was only evaluated at Sequoyah and Watts Bar because the raceway area is unique to the free-standing containment concept employed only at these two plants.

232.3, Improper Piping Insulation Material - Rock wool insulation used on piping systems is easily damaged and subject to deterioration from vibration. Nonmetallic insulation used on austenitic stainless steel safety-related systems may be of unacceptable quality. These two issues were deemed applicable to all TVA nuclear plant sites and evaluated for each of the plant sites.

232.4, Valve Seat Material Change - The change from hard seats to soft seats after hot functional testing indicates that inferior quality valves are installed at Watts Bar. As the issue was evaluated for Watts Bar and found to be invalid, it was not evaluated on the other three plants.

232.5, Building Columns not on Flow Diagrams - Building columns are not identified on flow diagrams. This issue, common to all TVA nuclear plant sites, was evaluated for Watts Bar where it was found to represent an acceptable practice. It was therefore not evaluated further.

232.6, Rubber Gasket Deterioration - Rubber gaskets installed in cooling water systems flanged joints exhibit rapid deterioration and could affect nuclear safety. This issue was considered applicable to all plants and was evaluated for each plant site.

232.7, Socket Weld Gap Radiation Hot Spots - Socket weld gaps create crud traps resulting in personnel radiation hazards. This industrywide issue was evaluated only for Watts Bar, where it was raised, because it stated a nonplant-specific (NPS) condition that is generically acknowledged and accepted and resulted in no corrective action.

232.8, Piping Wall Thickness Tolerance - In specifying pipe wall thickness for Class 1 and Class 2 stainless steel piping, it was noted that one engineer did not consider the manufacturing tolerance established by Engineering Design. This issue was evaluated at all four plant sites.

232.9, Freezing of Condensate Lines - The proximity of glycol lines to the ice condenser air handling units causes freeze plugging of air handling unit (AHU) condensate drainage piping. This issue was evaluated at the only two TVA nuclear plant sites which employ the ice condenser concept, Sequoyah and Watts Bar.

232.10, Drilled Holes in Branch Header - Small pipe branch connection fabrication technique of welding fitting to header, then drilling a hole in the header through the outlet fitting is a questionable practice. This issue was only evaluated at Watts Bar, where it was raised, as the technique described was found to be well established and generically accepted and required no corrective action.

232.11, ERCW Chiller Piping - Cooling water to chiller condenser trains should be provided with cross connections, and the cooling water supply pilot valves for chiller condensers should be provided with strainers in the trim piping. Neither of these recommendations by the CI were implemented. Also the chiller condenser has electrical contactor problems. These three issues were site-specific to Bellefonte and were only evaluated there because the two issues validated represent factual statements, neither requiring corrective action.

232.12, Butterfly Valve Seats - Circulating water system rubber-seated butterfly valve seats tend to dry and crack. This issue was evaluated at Bellefonte and, as it was found to be invalid, was not evaluated further.

232.14, System Color Coding - The use of the same system color code for three different systems could cause problems. Inasmuch as all TVA nuclear plant sites use similar schemes and the issue was found to be invalid upon evaluation, it was addressed only for Watts Bar where it was raised.

232.19, Excessive Pipe Movement - Piping located overhead of portal on elevation 708 has experienced large displacements during testing. This issue was evaluated only at Watts Bar as it described an operational problem at a specific and unique site location.

232.20, Defective Rockwell Valves - Defective Rockwell valves were discovered in a procurement audit. This issue encompassed valves purchased for four nuclear plant sites. It was evaluated as a Watts Bar issue only as the investigation found that the valves in question were acceptable as-is for use in any of the TVA nuclear plant sites.

The 16 element summaries above contain 42 issues which deal with presumed deficiencies or inadequacies in the design of plant piping systems. More specifically, 18 issues are concerned with the adequacy of design (found in elements 232.1, 232.2, 232.3, 232.4, 232.6, 232.7, 232.9, 232.11, and 232.12); 11 suggest errors or oversights in design (in 232.1, 232.2, 232.3, and 232.8); two relate to operational problems (232.11 and 232.19); two pertain to differences in design philosophy (232.5 and 232.14); four suggest startup procedural inadequacies (214.0); one reports on vendor quality (232.20); and one questions construction procedures (232.10). Additionally, three peripheral issues were identified during the investigation, all requiring corrective action (232.2, 232.8, and 232.9).

As the following sections show, four of the elements evaluated were found to be valid and to require corrective actions (232.3, 232.9, 232.19, and 232.20). One involves a design error (232.3), one involves adequacy of design bases (232.9), one relates to an operational problem (232.19), and one relates to vendor quality (232.20). Thus, this subcategory contains a small sample of valid issues that are basically unrelated to one another. Additional, of the three peripheral issues discovered, one related to lack of design detail (232.2), one related to standards not being followed (232.8), and one related to adequacy of design bases (232.9).

Each issue evaluated in the element reports is stated fully in Attachment B, which also lists corresponding findings and corrective actions that are discussed in Sections 4 and 5 of this report.

3. EVALUATION PROCESS

This subcategory report is based on the information contained in the applicable element evaluations prepared to address the specific employee concerns related to the issues broadly defined in Section 2. The evaluation process consisted of the following steps:

a. Defined issues for each element from the employee concerns.

b. Reviewed current regulatory requirements, industry standards, and TVA criteria documents related to the issues to develop an understanding of the design basis.

c. Reviewed applicable design documents, purchase specifications, drawings, calculations, and conducted facility walkdowns, as appropriate, to develop design understanding and to verify implementation status.

d. Reviewed applicable Preliminary Safety Analysis Report (PSAR), Final Safety Analysis Report (FSAR), Safety Evaluation Report (SER), and SER Supplements to understand scope and basis of NRC review, to determine regulatory compliance, and to identify any open issues or TVA commitments related to the design.

e. Reviewed any other documents applicable to the issues and determined to be needed for the evaluation, such as correspondence, procedures, test reports, Nonconforming Condition Reports (NCRs), Engineering Change Notices (ECNs), evaluation reports, etc.

f. Witnessed system operation to validate issues presented.

g. Interviewed TVA corporate and site personnel in person and by phone to develop understanding of problems noted.

h. Discussed component problems with supplier (vendor) representatives.

Details of the evaluation by element are in Section 4.

4. FINDINGS

The findings from each of the 28 element evaluations for this subcategory are contained in Attachment B. The findings are listed by element number and by plant.

The bases for these findings are discussed in the following subsections.

4.1 Flushing Particles - Element 214.0

As indicated in the concern summary:

- ANSI N45.2.1 (Ref. 1) identifies the cleaning methods
- TVA General Construction Specification G-39 (Ref. 2) identifies the cleanliness requirements
- WBN Construction Specification N3M-890 (Ref. 3) applies the cleanliness requirements to specific WBN fluids systems (N4M-891 at BLN) (Ref. 4)
- WBN Quality Control Test Procedure QCT-4.36 (Ref. 5) defines the procedures for cleaning and flushing fluid handling systems (BNP - CTP-6.1 at BLN) (Ref. 6)

4.4.1 Particulate Volume Testing

This concern questioned the absence of post construction flushing particulative testing in fluid piping systems.

TVA procedures for flushing and testing of piping systems and components were found to agree with the methods given in the ANSI standards. There is no mention of testing for the volume of particulates flushed.

The statement in the concern that the procedures "require a check for size and type of particles flushed from pipes" and the concern over a lack of measurement of the volume of particles imply that the flush test actually measures these parameters. As described in ANSI N45.2.1, the test is only applied to the piping system on a final flushing, after previous flushing has presumably washed most of the particulates out. The flush test results in a conclusion about cleanliness, i.e., that the piping system does not contain particles above the specified size. It does not identify the size or type of particles flushed, nor is the cleanliness test intended to determine what volume of particles was flushed out. The test does implicitly determine the volume of particles above the mesh size that may still be in the piping system. This value is none, or "occasional speckling," if the test is successful.

It is concluded that explicit testing for the volume of particulates flushed is not justifiable and this issue is not substantiated.

4.2.2 Instrument Sensing Lines

The CI questioned the different acceptance criteria applied to piping systems versus instrument sensing lines.

ANSI N45.2.1 contains no specific mention of procedures for instrument sensing lines. The ANSI standard assigns the responsibility of identification of systems and procedures to be used to the originator of the program. It requires methods for verifying cleanliness, "as appropriate." The WBN Construction Specification N3M-890 (Ref. 3) in note 5 on page 3-3 states that "water quality analysis and a check for particulates are not required" for instrument sensing lines. QCT 4.36 explicitly excludes instrument lines from its scope. TVA test procedures (Refs. 7 and 119), identify flushing requirements and the water quality requirements, but do not include any particle size or type tests. The cleanliness criteria are implicitly assumed to be met by the flushing of specified volumes of water, since these procedures do not provide a basis or references to related documents.

The TVA rationale for not testing for particulates was given in a telephone call with a TVA engineer (Ref. 8). The TVA general practice for the installation of instrument lines beyond root values, is to use stainless steel piping and socket weld fittings. These practices and materials produce fewer particles than others (e.g., carbon steel rusts easier). In addition, the instrument sensing lines are nonflow lines, so fewer particles are transferred into the lines during operation. The QCT 3.14 procedure prefers that instrument lines be flushed from the process pipe (after it has been cleaned) towards the instrument (which is isolated or disconnected), further minimizing the transfer of particles. Finally, instrument calibration flushing and venting will also clean contaminants from the instrument sensing lines.

In summary, the procedures are based on the assumption that flushing by itself is sufficient to assure instrument sensing line cleanliness. The evaluation team finds this to be reasonable and finds that the concern is not substantiated.

4.2 Accumulator Piping Size Change - Element 232.1

This employee concern clearly applies to the safety injection system (SIS), and to differences between design of this system on WBN units 1 and 2. The particular aspect of the SIS which resulted in the concern is a subsystem called Upper Head Injection (UHI).

The SIS is described in the WBN FSAR (Ref. 9) and shown on flow diagrams (Refs. 10 and 11). The SIS provides makeup water to the reactor for core cooling following a primary system pipe break, or loss of coolant accident (LOCA), for any break size. The SIS consists of several subsystems one of which utilizes four large accumulators which store sufficient water volume to provide initial core cooling following a large LOCA. Injection force is provided by a pressurized volume of nitrogen in each accumulator. This injection takes place automatically when reactor pressure drops to 373 psi.

As a result of Westinghouse reanalysis of SIS performance relative to NRC emergency core cooling system (ECCS) requirements (Ref. 12), SIS was modified on certain plants. This modification consisted of minor changes to the existing SIS, plus an add-on package called UHI. The UHI package consists of a gas accumulator, a water accumulator, a surge tank and piping to four nozzles on the reactor vessel head. UHI pressure is 1235 psi, allowing earlier flow to the reactor than would be available from the original accumulators. The modification to the original SIS consisted primarily of replacing a portion of 10-inch piping from each of the four accumulators with 6-inch piping incorporating a restricting orifice. This modification was required to balance flows in the existing and new portions of the system. This change was implemented on paper on WBN units 1 and 2. All hardware changes were made on unit 1. On unit 2, only the hardware changes to the original accumulator piping, e.g., the 10-inch to 6-inch size change, were made. The UHI package was not added to Unit 2.

Early in 1985 TVA reevaluated the addition of UHI to WBN unit 2. Recent core cooling analysis computer program modifications, based on NRC test programs, indicated that UHI was no longer essential to plant safety or operability. Also, operating experience at other plants incorporating UHI showed that numerous operational problems had occurred within the UHI subsystem. The bases for this reevaluation are included in TVA internal memos and in letters to the NRC (Refs. 13 and 14). As a result of the reevaluation, UHI was deleted for WBN unit 2. As a result of this decision, it was necessary to restore the unit 2 accumulator piping to its original 10-inch size. ECN 5548 (Ref. 15) was issued to cover this change, and the piping modification was completed. On unit 1, since the entire UHI modification had been completed, TVA decided to leave it as is, rather than to eliminate UHI. This accounts for the unit 1/unit 2 differences, including the 6-inch versus 10-inch piping, which existed at the time of the employee concern.

Subsequently, for unit 1, TVA reevaluated UHI on the bases described above. It was concluded that the potential operational problems were sufficient to justify deletion of UHI on unit 1. Therefore, the UHI package was disabled, and the 6-inch accumulator piping was restored to the original 10-inch size.

The assertion that the piping change was made as a result of a calculation error is incorrect, and the fact that unit 1 modifications were not made was based on a reevaluation of the core cooling analysis.

4.3 Carbon Steel versus Stainless Steel Drainage Piping - Element 232.2

The employee concerns pertain to the suitability of carbon steel piping material used in the WBN and SQN Reactor Building floor drain system. The concerned individuals stated that stainless steel piping should have been used. It was unclear from their statements whether the perceived problem was considered one of construction or design. Therefore, the NSRS investigation and this evaluation have addressed both aspects.

The Reactor Building embedded floor drain piping is TVA nonnuclear safety class L as noted in design drawing 47W851-1 for both plants. WBN and SQN FSAR Section 3 identifies no specific code jurisdiction ("unclassified") applicable to TVA class L piping systems for field fabrication, examination, and testing. The piping design complies with ANSI B31.1-1967, "Power Piping Code," and, since the Reactor Building is a Seismic Category I structure, the nonnuclear safety piping located inside is seismically supported as necessary to prevent unacceptable interactions with safety-related components.

WBN and SQN FSAR Section 9.3.3 design bases identify the Reactor Building floor drain system as a portion of an independent chemical waste collection and disposal system which prevents uncontrolled releases of hazardous materials to the environment. The floor drain piping material selection is based on the fluid to be handled being water and air as identified for WBN and SQN in FSAR Table 9.3.3. The TVA engineering practice, as documented in Division of Engineering Design, Mechanical Design Guide, DG-M8.1.6, "Nonradioactive Building Drainage Requirements," is to use carbon steel pipe and fittings with cast iron drainage fittings unless there are special considerations. In the Reactor Building application, the embedded floor drain piping has been specified in drawings 47W476-1 and 2 as carbon steel in the annulus area (outside of containment) and stainless steel inside containment below elevation 738.0 feet and 761.0 feet at SQN and WBN, respectively. Because of its superior corrosion resistance and ease of decontamination, stainless steel is used for embedded floor drainage piping inside containment in locations where there is a possibility of handling radioactive fluids.

The TVA Nuclear Safety Review Staff (NSRS) and the ECSP evaluation team each conducted evaluations (Refs. 18 and 19) to assess the validity of the concern by identifying the Reactor Building floor drain piping material requirements and by inspection of the installation where accessible for compliance. These investigations reviewed design drawings and bills of material (Refs. 16 and 17) to identify the reactor building drain arrangements and material requirements. Available relevant installation and inspection records were also reviewed and SQN site personnel familiar with the installation procedures were interviewed. Investigators performed visual inspection and magnetic testing of the accessible embedded floor drain pipe ends in the containment raceway of both units 1 and 2. These investigations verified the use of stainless steel embedded floor drainage piping in accessible containment raceway areas as called for on the design drawings and found no evidence that carbon steel pipe was used in these areas, thus concluding the concern was not substantiated.

Neither concern is substantiated, and both issues identified are invalid. However, field verification observations indicated that nonstainless steel material was used for gratings and cover plates. Both plants have committed to replacing the noncomplying carbon steel grates with stainless steel grates (Refs. 115 and 116).

4.4 Incorrect Piping Insulation Material - Element 232.3

4.4.1 Mineral Fiber Insulation

The first concern questions the durability of the soft (rock wool) piping insulation installed at WBN because of its alleged susceptibility to damage from abuse and long-term deterioration when exposed to vibration. Based on discussion with the insulation contractor, the term "SOF" (as quoted on the K-form) does not identify any type or brand name of insulation used at WBN. Therefore, it is interpreted to mean "Soft."

Rock wool is one of several mineral substances used in the manufacture of the fibrous type of mass insulation. "Mineral Fiber" is the ASTM Standard generic term for insulation material composed principally of fibers manufactured from molten mineral substances such as rock, slag, or glass, with or without binders.

The TVA Insulation Design Guide (Ref. 20) provides general and specific requirements, standards, and application guidelines for various types of insulation to be used in nuclear power plants including all metal reflective types for piping and equipment inside containment and mass types for other piping, equipment and ducts. The TVA insulation specifications (Refs. 21, 22, 23, 24, 25, 26, 27, and 31) provide specific requirements for the procurement and installation of piping insulation.

While mineral fiber insulation was widely used at Browns Ferry, the majority of installed insulation at TVA's other nuclear plants is molded calcium silicate. However mineral fiber insulation has been applied to some piping outside of containment only (excluding the mineral fiber block used in the main piping containment penetrations). Typically mineral fiber insulation is installed on heat traced piping containing borated water, essential raw cooling water (ERCW) system piping, raw cooling water (RCW) system piping, and some exposed drainage piping. These mineral fiber insulation forms can be characterized as soft because of their relatively low compressive strength compared to certain other types of commonly used pipe insulation such as molded calcium silicate. ASTM Standard C165, "Standard Method for Measuring Compressive Properties of Thermal Insulation," provides procedures for measuring thermal insulation mechanical behavior under compressive load.

Another of the several mechanical properties which needs to be considered in the selection of insulating material is hardness. Hardness is defined as that property which measures a material's ability to resist penetration. It affects ease of application and is determined by ASTM Standard C569, "Standard Test Method for Indentation Hardness of Preformed Thermal Insulators." The resilience of the wrap-around blanket type is desirable in some piping applications covering small obstructions such as heat-tracing.

Insulations of all types commonly applied to piping are susceptible to some degree of damage if not handled carefully during installation and protected from abuse after installation. Mineral fiber type piping insulation has been widely used in industrial and power plant applications and has provided many years of satisfactory service. If damaged during installation or after to the extent that thermal performance is unacceptable, the damaged section of insulation can be economically and readily replaced. Industry has found that, during installation, a reasonable amount of waste from abusive damage and other causes is anticipated and considered acceptable; it generally is reflected in allowances for such wastage in material specifications.

Mechanical vibration can cause deterioration of piping insulation thermal performance through wearing away, settling, or dusting of the insulation material.

The resistance to vibration of mineral fiber type piping insulation is good, and there are applications where its mechanical performance can be superior to that of a harder material such as molded calcium silicate (Ref. 28). In the judgment of the evaluation team, flow-induced forces resulting from the small amplitude vibrations produced by fluid flow at design velocities in applicable insulated piping at TVA nuclear plants are not of sufficient magnitude to cause significant wearing away or dusting of the mineral fiber or other insulation used. Furthermore, because of the bonding of the fibers, settling is not a problem with rigid or semirigid insulations.

4.4.2 Regulatory Guide 1.36 Compliance

The NSRS investigation revealed that the concern pertained to the potential for nonmetallic insulation causing cracking of austenitic stainless steel. This possible promotion of stress corrosion cracking could arise from contact of austenitic stainless steel with insulating materials containing excessive levels of leachable chloride and fluoride ions as defined in U.S. NRC Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Over an extended period of time the concerned individual had observed the installation on safety-related austenitic stainless steel systems of insulation types that he believed were of unacceptable quality (noncompliance with Regulatory Guide 1.36 requirements).

The TVA Nuclear Safety Review Staff (NSRS) received this concern on March 12, 1985 during a review of maintenance activities at SQN. The NSRS initiated an investigation to evaluate the validity of the concern by identifying the applicable insulation requirements for SQN and the plant's compliance with these requirements. NSRS Investigation Report I-85-106-SQN, issued December 27, 1985, contained corrective action recommendations which were completed by the SQN staff on May 25, 1986.

In conducting the investigation, the NSRS investigator reviewed relevant design drawings and procurement specifications, and interviewed responsible Office of Engineering (OE) personnel to identify the original SQN insulation requirements. Additionally, plant procurement procedures and standard practices documents were reviewed, and responsible plant personnel were interviewed to determine the extent of insulation replacement due to maintenance and modification activities. Also, the investigator evaluated the traceability of replacement insulation from purchase to installation.

The NSRS Investigation Report I-85-106-SQN found that while most of the insulating materials installed at SQN were the same as those used at other TVA plants where compliance with Regulatory Guide 1.36 is required, compliance at SQN was uncertain because certification documentation was lacking, and future compliance was not assured. The report appropriately recommended documenting demonstration of compliance with Regulatory Guide 1.36 requirements for nonmetallic insulation installed in the plant and stored on site. It also recommended procedural changes to assure compliance with the regulatory guide requirements of future replacement insulation purchased by the plant.

The SQN staff response (Refs. 29 and 30) to the NSRS report recommendations committed to Regulatory Guide 1.36 testing of nonmetallic thermal insulation installed on stainless steel in safety-related systems and warehoused, and to maintain testing documentation. Further, the plant response committed to revising SQN Standard Practice SQM35, "Nonmetallic Thermal Insulation-Austenitic Stainless Steel," to ensure that future insulation needs will be procured, stored, and installed in documented compliance with Regulatory Guide 1.36 requirements. These actions have been reported as completed (Ref. 114).

In marked contrast to SQN and WBN, which both employ the pressurized water reactor (PWR) system, BFN utilizes the General Electric boiling water reactor (BWR) concept. Whereas the PWR system employs stainless steel components in all systems in contact with the borated primary coolant, the only safety-related system at BFN, outside the drywell, which was originally designed with stainless steel piping, is the standby liquid control (SLC) system. This safety-related system consists of an atmospheric pressure boron solution tank, two positive-displacement pumps, two explosive-actuated valves, and associated piping. Components are mounted in the reactor building outside the primary containment. Injection piping from the pumps penetrates the primary containment and enters the reactor vessel via the differential pressure and liquid control line. The portion of the system inside the reactor building is maintained at atmospheric pressure and heat traced to 80°F.

Some nonsafety-related systems, most notably the reactor water cleanup (RWCU) system, employ stainless steel piping. The portions of those systems which penetrate the drywell between containment isolation valves are also classified as safety-related for containment pressure boundary purposes only. TVA specification 1067 (Ref. 31), required all metal reflective insulation (purchased from Diamond Power Specialty Corporation) for all piping systems inside the drywell and through the second isolation valve, outside the containment. However, because of the special nature of hot containment pipe penetration design, insulation for penetration piping was furnished by the penetration piping fabricator. Due to the lack of documentation to the contrary, this insulation may not have met Specification 1067 requirements. An investigation has been instigated (Ref. 32) to identify the material properties of the insulation used on this piping.

After BFN commissioning, severe reduction of pipe inner diameter was found in the carbon steel piping of the safety-related emergency equipment cooling water (EECW) system. The EECW system uses river water in a once-through cycle, and the reduction was a result of microbiologically induced corrosion. Subsequently, ECN L-1970 was issued to replace all small diameter piping (4 inches and smaller) of the EECW system with austenitic stainless steel piping (Ref. 33). Insulation requirements were not specified.

Pursuant to Employee Concern Special Program Report CATD 313 07 BFN 01 which states that "Browns Ferry has not evaluated the level of fluorides and chlorides in the nonmetallic insulation used on austenitic stainless steel safety-related piping," the BFN staff has conducted preliminary investigations. In a report prepared on March 13, 1987 (Ref. 34), the staff found that:

> "Nonmetallic insulation currently purchased or contracted for by TVA for BFNP is specified to meet Reg. Guide 1.36 requirements. Possibly some nonmetallic insulation is stored at BFN which does not meet these requirements. Limited document search (for piping 2-1/2 inches nominal diameter and larger) reveals that nonmetallic insulation was used on stainless steel piping on the SLC SYS (63) and the EECW SYS (67). No documentation is available to establish compliance to Regulatory Guide 1.36."

As noted in Attachment B, as a result of previous ECSP findings (Operations Category Element 313.07), Browns Ferry has instituted corrective action programs to ensure compliance with Regulatory Guide 1.36, which has yet to be completed. The evaluation team believes that this CAP should address the situation if "out-of-spec" insulation is found, and should also address any subsequent inspection plans to detect pipe cracking or to confirm the absence of such degradation.

At BLN, safety-related, austenitic stainless steel piping systems were required by procurement documentation (Ref. 24), to be insulated with ". . . materials meeting the requirements of Regulatory Guide 1.36 . . ." Review of receipt documentation for one shipment of calcium silicate insulation material from Johns-Manville Sales Corporation, the insulation supplier and installer, (Ref. 35) contain certified test reports which validate conformance with Regulatory Guide 1.36.

The NSRS report (Ref. 36) noted that ". . . Watts Bar Nuclear Plant's commitment to Regulatory Guide was . . . not clear. Their contract with their insulation supplier, North Brothers, did not require compliance and as a result Watts Bar could not document their compliance with the requirements of Regulatory Guide 1.36." A materials testing program was instituted by OE, and the problem was appropriately resolved as noted in an 50.55(e) report to the NRC (Ref. 37).

The issue was found not to be valid at SQN, WBN, and BLN. At BFN, further evaluation is underway to assess the acceptability of insulation materials.

4.5 Valve Seat Material Change - Element 232.4

This concern asserts that because of a change in valve seat material, the original valves purchased for WBN were inferior. This evaluation concentrated on Kerotest valves because a review of NCRs/DCRs revealed that they were the only valves subjected to the soft-seat replacement program.

In early 1982, the WBN mechanical maintenance section originated Design Change Request (DCR) WBN-DCR 447 (Ref. 38) to replace the installed hard-seated discs in certain 2-inch and smaller Kerotest Y-type globe valves. The requested replacement discs were to be soft-seated. Justification for the change was a reduction of the abnormal maintenance required to obtain tight shutoff (zero leakage) with the hard-seated discs, a condition for which they were not suited. As stated in the DCR, small amounts of leakage through the instrument isolation valves made it difficult to calibrate level transmitters with the system pressurized, and required rework of the valves to correct the leakage. All valve disc changes were made for maintenance reasons only; nuclear safety or systems function not being jeopardized. ECN 4061 was released 10/05/83, to implement this change on unit 1.

The change involved 20 TVA Class B (ASME Code Sec. III, Class 2) 2-inch instrument isolation valves, 16 on the safety injection system accumulator tank level transmitters, and four on the upper head injection system surge tank level transmitters, in each unit. All 20 of these valves are nonactive, (i.e., they are normally open and not required to close to perform their safety function). Also involved were 20 TVA Class G (ANSI B31.1) 1-inch and 3/4-inch valves in the waste disposal system nitrogen piping, six in the unit 1 portion, and 14 in the portion of the system common to both units. The

20 valves include 10 instrument root valves, eight process isolation valves, and two test connection isolation valves. Unit 1 replacement work was completed July 1, 1986 (Ref. 43).

In 1980, two nonconforming condition reports, 2272R (Ref. 39) and 2501R (Ref. 40), were issued at WBN that identified 11 distressed Kerotest 1-inch and 3/4-inch, 1500#, Y-type globe valves. Seven of these were disassembled and inspected after installation and were found to have their bearing assemblies destroyed and diaphragms damaged. The four other valves, taken at random from warehouse storage, were found to have varying degrees of corrosion and pitting in the bearing assembly and internal bonnet and stem surfaces. Three valves could not be operated by hand and the fourth exhibited abnormally high resistance to handle rotation.

The seven severely damaged valves were determined to be isolated instances of handle overtorquing (possibly caused by the resistance due to internal corrosion) and the valves were replaced. To preclude future problems, appropriate personnel were to be advised as to proper valve operation and the valve tee-handles were to be shortened to 4 inches to prevent two-handed operations. The corrosion in the four warehoused valves was determined to be a generic condition with Kerotest Y-type globe valves mainly as a consequence of reinstalling wet packing after factory hydrostatic testing (Ref. 41). TVA judged this to be a significant deficient condition and reported it to the NRC in accordance with 10 CFR 50.55(e) (Ref. 41) since many of the valves are used in safety-related systems.

A thorough investigation of the problem followed, along with development of a systematic maintenance program to dismantle, inspect, and refurbish as required all of the Kerotest Y-type globe valves. This ongoing program intended to preclude further maintenance problems commenced in 1981. In a February 1984 status report it was noted that: ". . . our inspection of over 500 valves has yielded no inoperable valves . . ." (Ref. 42).

The valve seat disc material change was performed to obtain tight shutoff (zero leakage), a condition for which the hard-seated discs were not suited. The disc change was based on maintenance considerations only and was not an indication of inferior valves as implied by the concern. The concern is, therefore, not valid.

4.6 <u>Building Columns Not on Flow Diagrams - Element 232.5</u>

This concern questions the lack of building column identifications on mechanical flow diagrams.

Flow diagrams are schematic in nature and are not intended to convey other than very general physical information. As a practice, flow diagrams show the physical relationship of various system components to each other but not to the structural plant itself. Building columns are part of the structural plant. It is not common industry practice to indicate building columns on schematic drawings as one finds on Physical Construction Drawings (WBN 900 series).

TVA NSRS responded to the employee concern in a newsletter entitled, "Nuclear Safety Update" (Ref. 44). This response read as follows:

> "The nuclear plant design flow diagram demonstrates schematically the functional operation of the system. The use of column lines on flow diagrams would be a very costly step, and the extraneous information would detract from the usefulness and the primary purpose of the drawing. There are no future plans to include column lines on flow diagrams. This practice is consistent for all TVA plants and with industry practices."

This NSRS response summarized a TVA memorandum, prepared by TVA Engineering Design (Ref. 45), which responded to a request for an evaluation of the employee concern.

4.7 Rubber Gasket Deterioration - Element 232.6

The concern relates to short-term deterioration of rubber gaskets installed in flanged piping joints in the essential raw cooling water (ERCW) and the raw cooling water (RCW) systems and in other unspecified systems at WBN. The concerned individual (CI) postulates that such a condition may have a possible adverse impact on plant nuclear safety.

Initially this concern was evaluated at WBN. Interviews with WBN maintenance personnel could not establish a historical problem with rubber gaskets in either safety- or nonsafety-related systems. On the basis of the WBN concern, the TVA SQN Generic Concern Task Force identified "rubber gasket deterioration" as a potential generic issue. An investigation was initiated to determine if a similar problem existed at SQN. The Task Force interviewed responsible SQN mechanical maintenance personnel, but none were aware of any rubber gasket deterioration problem. Additionally, the Task Force examined 1555 maintenance requests (MRs) on the ERCW system covering a 6-year period through April 17, 1986. Five of these MRs identified system leakage; however, none involved rubber gaskets. The Generic Task Force Report GOR-23-23 (Ref. 46) appropriately concluded that while rubber gaskets are used in certain mechanical systems in both the Reactor and Auxiliary Buildings, there has been no verifiable adverse impact on operation.

As part of the BFN evaluation, the review team examined the events found in Nuclear Power Experience (Ref. 47), covering 1,100 events in service water systems for all plants. For TVA plants, the only instance of rubber gasket deterioration was one identified at WBN, in which an unidentified rubber gasket material installed in nonsafety-related systems did not perform satisfactorily and required replacement 6 years ago. No further action was considered necessary at that time, and no further problems of this nature have been noted.

Evaluations determined that:

- o Flanged joints are used in the piping system in those limited cases where frequent disassembly is required for maintenance. Gaskets are used in flanged joints for economy, avoiding the expense of grinding and lapping the joint faces to obtain fluid tight joints. The gasket in a flanged piping joint provides a seal and is not associated with the pressure retaining function of the flanges and bolting. The presence of a deteriorated or imperfectly sealing gasket may result in a leaking joint but is not considered a failure of the system pressure boundary.

- o A leaking flanged joint is a plant maintenance item without impact on plant nuclear safety. The failure of a gasket is not a sudden event. The amount of system inventory that would be lost through joint leakage is not sufficient to diminish a safety system's ability to meet its intended purpose.

For SQN, WBN, and BLN, Section 9 of their FSARs states that the industry code applicable to the ERCW system is ASME Section III, Classes 2 and 3. The relevant section of this code, covering gasket materials, reads as follows:

> "(a) Gaskets shall be made of materials which are not injuriously affected by the fluid or by temperatures within the Design Temperature Range.
>
> (b) Only metallic or asbestos-metallic gaskets may be used on flat or raised face flanges if the expected normal operating pressure exceeds 720 psi or the temperature exceeds 750°F."

The industry code applicable to the nonsafety-related RCW system is ANSI B31.1. The relevant section of this code reads similarly to the section in ASME Section III quoted above.

Two safety-related systems at BFN, the RHRSW and EECW systems, perform the same functions as the ERCW system at WBN. The BFN FSAR states that the industry code applicable to the above systems is USAS B31.1.0, Section 1, 1967 with, again, similar requirements.

As shown in various design documents (Refs. 48, 49, 50), the maximum design pressure is 185 psig and the maximum design temperature is 200°F for the systems reviewed. These design conditions are well within the range of those for which red rubber gasket material is suitable. Red rubber is also suitable for use in systems containing river water.

For the systems reviewed, bills of material were examined (Refs. 51, 52, 53, 54). All gaskets in the ERCW system were specified to be rubber, Garlock ring-type #122 (or equal), 1/16-inch thick. One exception occurred in the RHRSW system, a portion of which required use of neoprene gaskets. In the RCW system, gaskets were generally specified as compressed asbestos, but some rubber gaskets were specified, to the same requirements as those in the ERCW system.

The Garlock #122 ring-type rubber gasket is manufactured to meet the requirements of ASTM D 1330 (Ref. 55) and is the highest quality rubber gasket material available from Garlock (Ref. 56). This gasket material is typically used in low temperature/pressure water services.

The evaluation team thus concludes that this concern is not valid for the rubber gasket material used at TVA nuclear plants and that there is no adverse impact on plant nuclear safety as a consequence of these rubber gasket materials.

4.8 Socket Weld Gap Radiation Hot Spots - Element 232.7

As stated, the concern relates to the gap provided between the end of a pipe and the bottom of the socket weld to allow for thermal expansion during both the welding process and operation. As required by the code, this gap should be approximately 1/16-inch. No maximum gap or tolerance is given in the code; however, as stated in the concern, TVA quality assurance acceptance criteria allows a maximum gap of 5/32-inch. These pipe-to-socket joints are used only in pipe sizes 2 inches and smaller and are recognized throughout the industry as potential crud traps, (i.e., radiation hazards in radioactive fluid systems). Actual, as-constructed variations in the 1/16-inch axial fit-up dimension (prescribed by the TVA welding standards) may have some effect on the volume of radioactive deposits that accumulate in a socket weld joint, but no quantitative data are available for comparative purposes. In responding to an NRC inquiry regarding facility design features in relation to the ALARA (as low as reasonably achievable) radiation protection concept, TVA stated (Ref. 57):

> "Use of butt welds in small pipe as a means for reducing potential crud traps was generally not considered in the Watts Bar design.

Large diameter piping (i.e., 2-1/2 inches and greater) is generally butt-welded as required by various piping codes. Smaller pipe is generally socket welded because:

(a) it was allowed by the piping codes,

(b) socket welded connections for small pipe were more readily available from vendors,

(c) the initial cost and installation cost was less for small pipe socket welds, and

(d) the Sequoyah Nuclear Plant utilized socked [sic] weld connections."

Many other features provide equal, or greater capability to accumulate radioactive fission products such as: valves, orifices, elbows, dead legs, and branch line connections. Obviously, crud traps cannot be totally avoided but care in the design process can minimize these hazards. As stated in the TVA submittal noted above, the following considerations were given in the design regarding the ALARA concept:

(a) Piping runs were generally sloped to aid drainage.

(b) Most tanks were specified to have curved bottom surfaces.

(c) In general, most drain tap-offs were located at low points in piping.

(d) Dead legs were minimized in the layout of piping.

(e) Piping in general was located to minimize run length.

(f) T connections were avoided in piping carrying spent resins or concentrates.

(g) Large radius bends and elbows were generally employed for spent resins and concentrates piping.

As noted in FSAR section 12.1.3, ". . . specific plans and procedures are followed by operating and maintenance staff to assure that ALARA goals are achieved in the operation of the plant." Specifically, ". . . employee radiation exposure trends will be reviewed annually by management staff at the plant and in the central office. Summary reports are prepared that describe (a) major problem areas where high radiation exposures are encountered; (b) which worker group is accumulating the highest exposures; and (c) recommendations for changes in operating, maintenance, and inspection procedures or modifications to the plant as appropriate to reduce exposures."

The concern correctly states the inherent radiation hazard presented by the use of socket welded connections. However, it is accepted industry-wide that there are many systems, including nuclear, where socket weld joints are appropriate.

4.9 Piping Wall Thickness Tolerance - Element 232.8

As stated, the concern appears related to the manufacturing tolerance allowed by ASTM Standard Specification A530 (ASME SA-530) for seamless and welded pipe purchased by schedule number or nominal wall thickness under several ASTM pipe product specifications. The ASTM standard stipulates the minimum actual wall thickness of seamless or welded pipe shall not be more than 12.5 percent under the nominal wall thickness specified in the procurement document. This tolerance is a maximum and provides one acceptance criterion for receiving inspection of commercially available seamless and welded pipe.

The minimum required pipe wall thickness is determined by a prescribed equation in the appropriate piping design code based on the fluid system parameters. These design parameters include internal pressure, pipe diameter, and maximum allowable stress for the selected material at the design temperature. After the minimum required pipe wall thickness is determined by the design code equation, the next heavier commercial wall thickness is selected taking into account the manufacturing tolerance allowed in the purchase specification. Thus the piping installed in the plant is assured of having an actual (manufactured) minimum wall thickness equal to or greater than the required (designed) minimum wall thickness. This consideration is covered in a footnote to Subparagraph NB-3641.1 (1971 Edition) for ASME III, Class 1 piping ("the code"), which states:

> "If pipe is ordered by its nominal wall thickness, the manufacturing tolerance on wall thickness must be taken into account. After the minimum pipe wall thickness, t_m, is determined . . . this minimum thickness shall be increased by an amount sufficient to provide the manufacturing tolerance allowed in the applicable pipe specification or required by the process."

In the equivalent section for ASME III, Class 2 piping, Subparagraph NC-3641.1(a), the footnote quoted above is also stated, further adding these "cookbook" instructions:

> "The next heavier commercial wall thickness shall then be selected from standard thickness schedules such as contained in ANSI B36.10 or from manufacturer's schedules for other than standard thickness." [Emphasis added.]

In certain situations where piping has been procured as a bulk commodity or installed, it could be necessary for an engineer to determine the maximum allowable design pressure for piping having a known (given) minimum wall thickness. In this case, the code provides a formula for calculating this pressure which in this case, consideration of the 12-1/2 percent manufacturing tolerance, is not appropriate. Typical of such situations would be system modifications or additions.

Since the concern was expressed at Watts Bar, a review was conducted of the WBN plant design criteria document WB-DC-40-36, "Classification of Piping, Pumps, Valves, and Vessels," Rev. 0, February 11, 1975, and Table 3.2-4 of the WBN FSAR. This revealed that WBN committed to the ASME B&PV Code Section III for safety-related systems piping design as required by the NRC (Ref. 58). There is no evidence to support the allegation of the concern that TVA Engineering Design established for WBN a piping design criterion more stringent than the ASME code for determination of minimum required pipe wall thickness.

Evaluators reviewed selected piping calculations for TVA nuclear units (Refs. 59, 60, 61, 62, and 63) and verified that appropriate code formulas were used and that they reflect applicable piping manufacturing tolerances.

A review of BFN plant calculations performed in 1967 (Ref. 61), reveals that although appropriate consideration was given to pipe wall manufacturing tolerances, the formula used in some systems to calculate minimum pipe wall thickness was less conservative than the code required. The formula used was an apparent misapplication of a GE formula which was only to be used in calculating piping design pressures and was so noted in GE documents (Ref. 64). A review of the piping bill of material drawings for the core spray system 12-inch piping (Ref. 65) indicates that originally the piping minimum wall thickness specified was derived from use of the GE formula. These Bill of Material Drawings were later corrected and reflect conformance to Code requirements. Indications are that the GE formula was used only in calculations for GE designed systems. A comparison of wall thickness derived from use of the GE formula versus the Code (B31.1.0) is shown in the accompanying tabulation. In the judgment of the evaluation team, this should have a negligible effect on the pressure boundary integrity of piping systems as the errors introduced are small, procurement of piping to standard schedules usually ensures excess pipe wall thickness, and the allowable stress values used in such calculations are well below material yield strengths. However, piping calculations must be reviewed to ascertain code compliance of installed piping.

CORE SPRAY PIPING CALCULATION (MINIMUM WALL THICKNESS)

GE Formula (used by TVA)

$$t_1 = \frac{(P_1 Z + H)\, D}{2S_1 F + 2Y(P_1 Z + H)} + C_r + C_c$$

t_1 = Wall thickness to be used in determining pipe design pressure only

F = Allowable stress factor (1.2)

P_1 = Reactor vessel design pressure (1250 psig)

S_1 = Allowable stress (14,406 psi for SA 376 TP 304)

C_r = Structural stability factor (0)

C_c = Corrosion factor (0.0024")

H = Upstream pump head (0)

Y = Temperature coefficient (0.4)

Z = Pressure factor (1.1)

D = Outside diameter (12.75")

$t_1 = 0.494"$

USAS B31.1.0-1967 [para. 104.1.2(a)]

$$t_m = \frac{PD_o}{2(SE + Py)} + A$$

t_m = Minimum required wall thickness in inches

P = Internal design pressure, psi gauge (1250 psig)

D_o = Outside diameter of pipe in inches (12.75")

SE = Maximum allowable stress (14,406 psi)

A = An additional thickness to account for threading, corrosion, etc. (0.0024")

y = 0.4

$t_m = 0.537"$

Conclusion: Use of the GE formula results in a calculated wall thickness less (thinner) than that required by the applicable code.

On the basis of regulatory and other appropriate system safety criteria, TVA has systematically classified plant equipment for application of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III design rules. Code Subsections NB and NC provide rules for the design of Code Class 1 and 2 items (TVA Classes A and B), respectively. Under the Code-permitted optional use of Code Classes, items classified as Class 2 (TVA Class B) may be designed and certified under the rules of Subsection NB. Thus any Code Class 2 pipe wall thickness determinations made under the rules of Code Subsection NB, as is implied in the concern, are acceptable and do not represent a safety issue.

4.10 Freezing of Condensate Lines - Element 232.9

The EC was initiated at WBN and describes a condition of freezing of the condensate drain lines which connect to the AHU drip pans, apparently resulting in clogging the AHUs with frost buildup. According to the EC, this resulted from proximity of the AHU glycol supply lines to the condensate drain lines.

SQN and WBN incorporate the Westinghouse ice condenser containment concept in their design. This concept utilizes a large bed of ice chips to absorb the energy release of a large pipe break, in a manner analogous to that used in the BWR pressure suppression containment concept. The ice bed is located in the annular space between the "crane wall" (the structure surrounding the primary coolant system components) and the steel primary containment shell. The ice bed is at the level of the upper portion of the steam generators.

To minimize loss of ice, this annular area is separated from the warm primary containment atmosphere by ventilation barriers. There are also wall units, inside the containment shell and outside the crane wall, through which cold air is forced to maintain an ice bed temperature of 10° to 15°F during plant operation. Air cooling occurs in 30 air handling units (AHUs) in two rows adjacent to the containment shell and crane wall, immediately above the ice bed. It should be noted that the quantity of 32 AHUs, as stated in the EC, is incorrect. The correct quantity for WBN and for SQN is 30.

Coolant, consisting of a water/ethylene glycol ("antifreeze") solution referred to as "glycol," is supplied at -5°F to cool the air in the AHUs. The glycol, in turn, is cooled in chiller units outside containment. None of this air and glycol cooling equipment is required to perform during or following an accident; therefore, it is not "safety-related" (Ref. 117).

Because of the very low temperature involved and the presence of moisture in the air cooling the ice bed, frost continually develops on the AHU surfaces. To counter this frost buildup, an automatic timed defrosting system is incorporated in each AHU. This system terminates glycol flow, and initiates heaters on the cooling coil surface and on the drip pan which collects the condensate from the AHU and directs the condensate to the drain lines.

Examination of the SQN drawing of the ice condenser AHU glycol lines and condensate drain lines (47W462-9, Rev. 8) shows that, in the AHU area, all of the glycol lines are above the AHUs and all of the condensate drains are below the AHUs. The minimum distance between glycol and drain piping is more than 6 feet. Also, both the glycol and drain piping are insulated. For SQN, therefore, freezing of condensate drain lines resulting from proximity of the cold glycol lines is not a credible event.

Review of the event reports in Nuclear Power Experience (Ref. 67) indicates that there have been recurrent problems with ice condenser AHU condensate drains. Examples of such events are as follows:

a. Sequoyah 1, October 1980. Water in the drain line froze due to a heat tracing tape failure. From subsequent defrost cycles, water overflowed the drain system.

b. Sequoyah 1, December 1980. A flange gasket in a drain line leaked. Also a flow blockage was caused by freezing, rupturing the drain pipe.

c. Sequoyah 1, January 1981. A flange joint in a drain line leaked due to loose flange bolts. Also a drain line rupture was caused by freezing of the drains, resulting from a failure of power to the heat tracing tape.

d. Donald C. Cook 2, August 1981. A drain line ruptured apparently due to freezing of the drains.

e. Sequoyah 2, July 1982. A drain line ruptured due to freezing, caused by a heat tracing tape failure.

It appears that all of the above failures of the AHU condensate drain lines either were attributed to, or could have been caused by, freezing of the condensate due to heat tracing failures. The possibility of such occurrences is better understood when examining the detail drawings of the SQN drainage piping, 47W462-9 Rev. 8 and 47W462-59 Rev. 1 (Refs. 68 and 69). These show that each AHU drain has an individual loop seal immediately below the flanged drain connection to the AHU. This is necessary to prevent flow of hot air from the primary containment to the AHU area. These individual lines terminate in lengthy, semicircular headers, all of which are contained in the AHU area which is permanently maintained at 10° to 15°F. Even with the drain lines insulated, loss of heat tracing for any length of time is very likely to result in AHU condensate drainage freezing. This necessary piping and equipment arrangement of the ice condenser system is the potential cause of drain line freezing at SQN, not their proximity to glycol lines, as indicated in the WBN EC. Therefore, while the general subject of drain line freezing may be generic to SQN, the WBN cause identified is not.

In parallel with the investigation described above, TVA performed an evaluation of the EC as it potentially applied to SQN (Ref. 70). The resultant report confirms the information above. The report adds information on the current status of the physical condition of the condensate drain line insulation at SQN. It states that portions of the insulation are "either missing or severely degraded." This is contained in a listing of icing problems experienced at SQN around and beneath the AHUs. The report recommends that "this insulation should be replaced with a new insulation compatible with subfreezing temperatures."

An area of conflicting information was encountered in this evaluation. This area was also discussed in the TVA report described in the previous paragraph. The SQN FSAR, and the WBN FSAR as well, in paragraphs 6.5.5.2 and 6.7.6.2, respectively, describe the AHU defrosting system. Both references state that the defrosting system includes heaters for the AHU coils and drip pans, but also lists "a condensate drain heater," all of which are controlled by the defrost timer. The condensate drain heater could be interpreted to include all, or a portion of, the condensate drain piping. This was assumed in the TVA report (Ref. 70). However, review of SQN drawing 47W462-59 shows that the condensate drain lines are heat traced from the point of connection to the AHU. As indicated earlier, the drain lines contain a loop seal immediately below this flange. This loop seal cannot be allowed to remain unheated for more than a few minutes. Power and control for the heat tracing are independent of the AHU defrost cycle controls. It is unclear what was meant by the term "condensate drain heater" in the FSARs, and there is no mention of the drainage piping heat tracing. It appears that the FSARs should be revised to clarify these points.

At WBN the location of the problem area identified in the concern is at the top of a circular stairway which provides access to the compartment in which the ice condenser AHUs are located. The stairway is located at approximately the 300° azimuth. The elevation of the top of the staircase is approximately 806 ft. The glycol supply and return lines, as well as the AHU condensate drain headers, penetrate the "end walls" of the AHU compartment above this elevation. One of the end walls is immediately adjacent to the top of the access staircase.

Outside this end wall the glycol piping and condensate drain piping are in close proximity. Portions of the drain piping, 1-1/2-inch pipe size, run vertically, immediately outside the end wall. Portions of the glycol piping, 4-inch pipe size, also run vertically, immediately outside, and in line with the drain piping. The drain lines inside the AHU compartment are heat traced, but the heat tracing terminates at the outer edge of the end wall. The glycol piping is covered with "anti-sweat" insulation both inside and outside of the end wall. The piping arrangement is shown in TVA drawings 47W462-8 and -9 (Refs. 71 and 72) and insulation details are shown in TVA drawings 47W462-408, -409, and -411 (Refs. 73, 74, and 75).

WBN Mechanical Maintenance investigated the problem stated in the concern, and a brief report was issued by WBN Power and Engineering (Ref. 76). The following is the text of the report in its entirety:

> "This concern was identified in December 1984. Originally, the glycol and drain lines were installed close together (actually touching in some locations) and were insulated together. The installation appears to have been made within construction tolerances.
>
> The glycol line has been reinsulated in an attempt to correct the problem. When icing occurs (about three times in the last year), Mechanical Maintenance (MM) has used heat to melt the frozen portion of the drain line..
>
> This problem has been added to the MM AI for tracking and is the responsibility of Mechanical Maintenance Group B. Concurrent with tracking of the problem, we are continuing to investigate, troubleshoot and develop alternatives to include heat tracing, rerouting of drain line, and reinsulation. The possibility exists that after unit 1 goes into operation, the ambient heat load will be sufficient to prevent freezing."

Subsequent to issuing the report described above, TVA took action to reduce the probability of condensate drain line freezing. The drain line in the problem area was moved away from the glycol line as far as practical, within construction tolerances. This action was covered by a maintenance request (Ref. 77). Also, the insulation was removed from the drain line. Since these actions were taken, the drain line has not frozen, as confirmed in a phone conversation (Ref. 78).

The problem noted in the concern was valid at WBN. At SQN, the problems which occurred resulted from causes other than noted in the EC and do not involve safety-related components. Any corrective active action thus taken at SQN will be for the purposes of ensuring more efficient plant operation or eliminating maintenance problems.

4.11 Drilled Holes in Branch Header - Element 232.10

The CI felt that the technique of welding attachment fittings to header piping prior to cutting the hole in the header piping was questionable.

Evaluation team members surveyed the area noted in the concern and found three branch connections off the 4-inch fire protection system header, two 2-inch and one 1-inch. The branch connections themselves were "threadolets" as specified in the installation drawings (Refs. 79 and 80). Interviews with plant personnel (Ref. 81) present during construction indicate that these branch fittings were welded to the main header piping after first cutting the requisite size hole. This is a different fabrication detail than that presented in the employee concern. Nonetheless, the procedure described in the concern is a well established fabrication technique and utilized in attaching half couplings and the various forged reinforced "weldolet" type fittings. TVA piping bill of material drawings specified "Bonney Forge and Tool Works threadolets, or equal." This supplier notes in its product information bulletins (Ref. 82), under installation procedures:

> "Cut Hole - The hole in the run pipe on reducing sizes can be cut out either before or after the fitting is welded on. The hole can be cut with a torch, a drill or a hole saw. Welding the fitting to the run pipe prior to cutting the hole helps prevent distortion of the run and can be done generally on outlet sizes over two inches.
>
> Layout - The template is the inside of the fitting."

Because of the statement that this technique is generally employed on outlet sizes over 2 inches, Bonney Forge was asked if cutting after welding was improper for piping 2 inches and under (Ref. 83). They stated that the only reason for the over-2-inch caveat was that, because of the small sizes, it is more difficult to scribe the template hole and there is a greater chance of damaging the internal threads. There is no concern in this regard pertaining to encroaching on the pipe pressure boundary integrity. Bonney Forge's experience (Ref. 83) is that, in general, piping fabricators usually cut the hole in the header piping before effecting the attachment weld of 2-inch-and-under branch piping.

The portion of the fire protection system piping under question is classified as TVA nonnuclear Safety Class G, for which the required piping design code is ANSI B31.1, "Code for Power Piping" (Ref. 84). This code provides rules for welding and calculational requirements governing the design of branch connections that use weld outlet fittings, such as forged couplings, adapters, and nozzles. However, the code is silent on the sequencing of hole preparation in header piping.

The concern is invalid because the procedure noted is a well established and acceptable piping fabrication technique.

4.12 <u>ERCW Chiller Piping - Element 232.11</u>

4.12.1 Chiller Piping Cross-Connections

This concern documents the fact that the CI had proposed a design change to provide crossover ERCW ties between redundant trains of various chiller units and the change was not implemented.

In an interview, the CI further amplified on his concerns in regard to BLN control room chillers (system VK):

> "The A-train chiller utilizes A-train ERCW water and the B-train chiller only utilizes B-train ERCW water. There is no crossover to get A-Train ERCW to B-train chiller or vice versa which eliminates the ability to provide backup for control room cooling. For example, the possibility exists that the A-train chiller could be out-of-service for repairs and B-train ERCW water be lost, then there would be no quick way of getting water to B-train chiller. A similar situation has occurred but the cooling problems have always been handled by temporary units, but these units will be removed before the plant goes into operation. [CI] previously suggested a crossover piping tie-in between A-train and B-train ERCW systems with a locked valve to seal the systems. When he proposed this solution to mechanical engineering, he was told the 'code' prevented utilizing a valve but the piping could be installed with a spool piece left out of the line. An engineer [blank] in the Mechanical Engineering Unit was going to consult with design engineering about the changes. [CI] is not aware of any changes having been made. This problem also applies to the VE and VF chillers."

The Control Building environmental control system (VK) is an essential, nuclear safety-related system. The system provides heating, ventilation, and cooling of all areas, including the control room, within the Control Building. Two 100 percent capacity water chiller installations are provided. Each receives ERCW cooling water and electrical power from separate "trains" (A or B) in the classic manner as is required by 10 CFR 50, Appendix A, General Design Criterion 44, Cooling Water. This criterion stipulates a design necessary to meet the single failure criterion (Refs. 87 and 88). Inasmuch as these chillers serve both normal and emergency functions, cross-connections, such as suggested by the CI, would seem desirable to enhance plant reliability. However, to ensure that the system design continues to meet the single failure criterion, features in addition to cross-over piping are required. As a minimum, the cross-connection would require two, redundant power-operated valves, each receiving power from a separate class 1E power supply. Each of these valves and their associated control and electrical power runs would have to be separated sufficiently to meet the additional nuclear safety criterion to prevent common mode failure. As a general rule, because of the added complexities involved in providing cross-connections, such provisions are discouraged as not being in the interest of safety if other means can be provided to ensure system reliability.

The chillers of the VK system (which services components common to both units 1 and 2) are piped to the two redundant trains of the unit 1 ERCW system. Each train of the ERCW system is provided with two ERCW pumps of approximately 70 percent capacity. Although the VK chillers are not directly connected to the unit 2 ERCW system, the corresponding train headers of the two units are cross-connected and normally run in this manner. Thus, as designed and intended to operate, three of the four pumps, in either train A or B for the two units, are required to operate, leaving one pump in each train as an installed spare.

Considering the example presented by the CI, as quoted above, if the A train VK chiller has been taken out of service, how could cooling water flowing to the B chiller "be lost"? Essentially integral to each chiller in the ERCW upstream piping is a pilot-operated control valve and a butterfly valve permanently throttled to a preset position determined during initial system balancing tests. Either of these valves could go closed for some reason; however, inasmuch as they're considered integral to the chiller due to function, any train cross-connections provided would have to be upstream of these two valves and would serve no useful purpose given the example noted. Upstream of these two valves, and also upstream of a suggested cross-connection, is a manually operated, normally open, butterfly valve. Failure of this valve in the fully closed position is deemed highly unlikely. Upstream of all components served in the 36-inch supply headers is a motor-operated butterfly isolated valve. This valve could be inadvertently closed. However, it could be almost immediately reopened either by its motor operator or by the manual handwheel. The possibility exists that the disc could separate from the valve stem, resulting in the disc orienting itself to the stream flow. One could also postulate that the valve could be jammed closed so that it could not be opened. In this case, however, loss of the whole train would necessitate immediate plant shutdown, and individual chiller cross-connections would be of little value.

Loss of one or more ERCW pumps can also be postulated. Should one pump be lost, three would remain; thus design flow is immediately assured. Should both pumps in one unit be lost because of an electrical fault, the two pumps in the other unit would remain, which would provide almost sufficient cooling capacity. However, as noted in the case of a header isolation valve failure, such a condition would require immediate plant shutdown. Here again, individual chiller isolation valves would serve no useful purpose.

The Auxiliary Building common zone water chillers system (VF) and the Auxiliary Building water chillers system (VE) are configured similarly.

Although the addition of cross-connections in some systems is a valid engineering solution, in the opinion of the evaluation team, little reliability enhancements could be achieved at BLN considering the unit cross-connections incorporated in the original design.

4.12.2 Chiller Pilot Valve Strainer

This concern relates to the fact that the CI had proposed adding a strainer in the ERCW system which had not been implemented.

The CI in his interview goes on to further state:

> "The ERCW line (for VE and VK chillers) is full of rust, dirt, etc. An orifice comes off the ERCW line to a 1/2-inch copper line which leads to a Freon-operated metrics pilot valve. The trash in the ERCW line clogs up the orifice which keeps the valve open and affects the operation of the chillers. [CI] proposed a strainer be put in the line to keep the orifice from clogging up. The problem was presented to design by Bellefonte engineer [blank]. The design engineers indicated that hangers would have to be added to the system for the strainer. [CI]) believed the strainer could be mounted from the pipe at minimal expense. The problem will continue to exist without a strainer and the operation of the chillers will continue to be affected."

Such a problem was noted in NCR 2086 on November 17, 1982 (Ref. 85), which stated:

> "The 6-inch freon activated raw cooling water flow control valves . . . failed to operate under activation. Investigation revealed . . . the buildup of rust and scale on the valve's pilot seat contact surfaces, inhibiting the movement of the valve stem."

NCR-2086 further states:

> "In addition large deposits of loose flaking rust were found on internal surfaces of the valves. It is suspected that loose rust particles broke free of the valve wall and became trapped in the valve pilot seat. The excessive amounts of rust discovered suggests that this type of failure will be recurrent."

Initial corrective action suggested was noted as follows:

> "In order to prevent recurrence of rust deposits in the control valves, the control valves will be drained and flushed with fresh water immediately after each system test until the system is fully operational. This procedure will be followed for all valves in the ERCW system and will be employed anytime the system is shut down."

In a subsequent revision to NCR 2086, the disposition was revised to:

- Add in-line strainer and isolation valve to the water-chiller condenser water control valve trim piping.
- Replace diaphragm plate with a noncorrosive plate.
- Add a rust-preventive coating to the upper internal valve area to prevent plugging of the control valve pressure sensing ports.

Inasmuch as it was noted that the corrosion products were the result of corrosion of valve internals, an additional strainer in the control valve trim piping would be of no practical value. The disposition was again revised on June 6, 1984 to:

"1. Add rust preventive coating to the valve diaphragm plate.

2. Add rust preventive coating to the upper internal valve area to prevent plugging of the control valve pressure sensing ports."

In discussions with plant maintenance personnel (Ref. 86), the evaluator confirmed that the corrective actions noted above have solved the problem noted in the concern. Yearly valve overhaul is also necessary to clean the valve internals of buildup of microbiologically induced corrosion (MIC) products.

The evaluators agree with the plant's assessment that the addition of strainers in the pilot valve piping would not solve the pilot valve failure problem which resulted from corrosion of the valve internals.

4.12.3 Electrical Contactors

This concern refers to contactor problems with the Control Building and/or Auxiliary Building chiller compressors VE, VF, and VK. No specifics as to the nature of the problems were given, other than a comment by the CI that ". . . there was an electrical contactor problem on VE, VF, and VK chillers that needs to be reviewed. The system is advanced technically, but it is unreliable."

A search of RIMS files for 1980 through 82, 1983 through 85, and 1986 to the present, and informal discussions with TVA personnel, disclosed the following items as possible objects of the concern:

- Trane, manufacturer of the Auxiliary Building common area chiller compressors, designated VF, expressed concern over the use of circuit breakers as motor controllers for their large (1350 hp, 6.9 kV) compressor motors (Ref. 89). Trane recommended motor starters (sometimes called "contactors") for the purpose. One of the arguments brought up by Trane against the arrangement was the potential loss of control circuit fuses, resulting in loss of motor control and protection.

- o TVA maintained that the control and protection provided by the existing breakers and their associated relaying were satisfactory and declined procurement of the starters recommended by the vendor (Ref. 90).

- o Before Trane's communication, during a test, an inoperative fuse disabled the trip circuit of a Control Building class 1E chiller compressor motor. Attempts to shut down the unit failed, and the motor burned out (Ref. 91). The event was considered significant and was reported to the NRC.

- o In evaluating the event, TVA concluded that the cause of failure was human error. The subject fuses were erroneously installed in the inactive maintenance holders in the switchgear furnished by Westinghouse. These holders are side by side with the active fuse holders, where the fuses should have been installed to be operative. This arrangement was viewed as a key factor leading to the event. Administrative procedures to double-check the fuse positions following maintenance were considered, as well as rework that would have physically blocked the spare holders against the insertion of fuses. This adopted resolution was to paint the active fuse holders, so that the absence of fuses from the operational position is readily visible, thus making recurrence of the error unlikely.

On the basis of the foregoing, the evaluation team made the following assessment:

- o In general, the use of breakers for the larger motors (above the 100 to 150 hp range) is an established practice in power station distribution system design. Frequent start-stop operations, for which starters have a distinct advantage over breakers, are uncommon in power station applications.

- o Occasionally, chiller packages are procured with starters for the main motors. Reduced voltage (e.g., wye/delta) starting may be necessitated when system capability is exceeded by the power required for direct starting of relatively larger compressor motors. Also, designers may elect to leave the somewhat involved controls in the vendor's area of responsibility. None of this means that starters for the main compressor motors are necessary under all circumstances.

- Starters of larger sizes tend to be expensive, bulky, and require local mounting (i.e., outside the electrical distribution equipment). Their reliability may also be of concern.

- Control power for tripping is inherently required for all power circuit breakers. The chiller compressors are not unique in this respect. The corrective action for the incident above does not prevent fuse burnouts. A failed fuse or any other failed element associated with a class 1E chiller is considered a single failure and should be covered by redundance and periodic testing. Review of this area has been considered to be outside the scope of the response to this concern.

Electrical chiller package contactor problems cannot therefore be substantiated as a valid concern at BLN.

4.13 Butterfly Valve Seats - Element 232.12

This concern relates to the assertion that the condenser circulating butterfly valve seats tend to dry and crack.

Review of plant files turned up one rejected work request (Ref. 92) to repair a leaking unit 1 main condenser 'A' waterbox discharge valve. No investigation work appears to have been conducted, thus the cause of this problem remains unknown. No other documentation related to butterfly valve seat problems in the condenser circulating water system (KH) could be located. This search did reveal however, five nonconforming condition reports (Refs. 93, 94, 95, 96, and 97), which report on butterfly valve seat problems for valves installed in the ERCW system (KE). Problems described were:

a. Seats cut and portions missing

b. Approximately 1 inch of the two valves seats was cut

c. Approximately 6 inches of the valve seat was pulled from the valve body

d. Seats had been rolled back and pulled from their seats [sic]

e. A 10-inch section of the seat groove was corroded which formed a 1/4-inch deep trench in the seat groove

f. Approximately 1/2 of the valve seat was pulled loose from the valve body

Causes of the individual problems noted above were given as:

a. [Seat cut] apparently caused by the seats being crimped during closing of the valve discs

b. [Seat pulling loose from valve body] apparently caused by failure of the epoxy to bond the seat to the valve body

c. Glue used to put the seats in . . . has a limited shelf life

d. Cause of the damage was not apparent

e. Damage to the seat groove was apparently caused by corrosion in conjunction with flow erosion

In its final 10 CFR 50.55e report to the NRC (Ref. 98), TVA reported:

a. Valve seats deteriorated during storage or were damaged due to improper storage or damaged during system flushing

b. [Vendor] informed TVA that the subject seats have a finite shelf life that is bounded by storage conditions

On the basis of its investigation, the valve vendor concluded (Ref. 118):

> "To date, the seat failures that have occurred have been attributed to damage (Cuts) by foreign objects or misuse/mishandling during site storage or installation. Since the majority of the . . . valves at Bellefonte Nuclear Plant are installed, and since we have had no similar problems at other Nuclear facilities, we can assume that the crucial period is over."

In a discussion with plant/site personnel on the subject (Ref. 99), it was noted that components with rubber seats and linings (valves, tanks, etc.) currently in dry layup show signs of deterioration and cracking.

Acknowledging project schedule extension, TVA reports (Ref. 100): "It is expected that the shelf life of the above valve seats will be exceeded at BLNP fuel load. It was determined that it is uneconomical to implement a shelf-life extension program at this time." Present plans (Ref. 99) are to test and repair deteriorated components prior to startup.

The evaluation team thus concluded that the materials selected for use in cooling water system butterfly valve seats are appropriate.

4.14 System Color Coding - Element 232.14

Because the same colors were used to identify a number of different systems, the CI felt that this could cause adverse operational problems.

The specifics of this particular employee concern posed some difficulty in the evaluation process. Examples of this difficulty are as follows:

- No "piping color code and lettering board" could be identified at WBN.
- No WBN color coding system could be identified which used the same color for plant items to which the numbers 6, 8, and 9 apply.

The evaluator determined that the numbers "6, 8 and 9" as applied to "piping" could only refer to TVA's system identification numbers. The systems numbers for WBN are found in a TVA electrical design standard (Ref. 112). The systems apparently referred to in the concern are as follows:

- 6 - Heater Drains and Vents System
- 8 - Miscellaneous Turbine Connections
- 9 - Miscellaneous Turbine Vents System

There are several color coding systems which apply to various portions of TVA power plants. The only systems which apply to piping are those which apply either to identification of systems containing hazardous materials or identification of specific process or mechanical systems. The TVA hazard control standards (Refs. 23 and 113) contain no numbering system. It is, therefore, highly unlikely that this aspect of color coding was the subject of the employee concern. Furthermore, no potential "problem" could be identified relative to the application of the TVA hazardous material identification system.

TVA's system color coding is also found in the standard referred to above. This color coding is used to identify systems for plant operations, such as for control board switch nameplates. Since there are 70 mechanical system designations at WBN, several similar systems, up to a maximum of six, use the same color. Turbine-generator auxiliaries, such as systems numbers 8 and 9, miscellaneous turbine connections, and miscellaneous turbine vents, use the color gold. System number 6, feedwater heater drains and vents, along with five similar systems, uses the color medium blue. In only four cases is a color uniquely assigned to a system.

The evaluation team determined that the issue raised in the employee concern was the use of the same color code for several different systems or subsystems, which might cause confusion to plant operators. The TVA grouping of systems/subsystems by color code was reviewed. In all cases, the evaluator was able to assign a brief title common to the systems included in the color code grouping. Examples are as follows:

- Color code: orange
  Systems included: main and reheat steam, auxiliary steam, extraction steam, safety and relief valves
  Common title: high and intermediate pressure steam systems

- Color code: light blue
  Systems included: raw cooling water, service water, condenser circulating water, high pressure fire protection
  Common title: non-safety-related raw water systems

As indicated above, the systems listed in the employee concern, although somewhat related, were actually from two different color code groups. These particular color code groups were somewhat more diverse than the groups tabulated above. The groups were as follows:

- Color code: medium blue
  Systems included: condensate, demineralized water, makeup demineralizers, heater drains and vents, water treatment, moisture separator drains

- Color code: gold
  Systems included: turbogenerator and auxiliary controls, miscellaneous turbine vents and connections, central lube oil, central hydrogen cooling, miscellaneous generator controls
  Common title: systems designed by main turbine-generator supplier

The color coding system used at WBN is also used at SQN and BLN. An earlier plant, BFN, used a more complex system, allowing more differentiation of systems by use of additional colors and shades, and color combinations by the addition of bands to switch nameplates. It is the evaluator's opinion that the simpler system used on the later plants is preferable and that the system color coding selections were reasonable and appropriate.

4.15 Excessive Pipe Movement - Element 232.19

This concern notes the CI's observation that piping "...move[d] drastically during testing" at one specified location.

The portal on elevation 708 is an area occupied by operations and security personnel. High temperature and high pressure piping (main steam, feedwater) runs directly above this regularly occupied area. Under certain modes of operation (prestart deaeration, long cycle feedwater recirculation), a 10-inch-diameter steam generator blowdown line to the condenser has experienced flashing flow conditions in a regulator valve located close to portal 708. This non-safety-related line has a history of severe vibration and displacement problems caused by these thermodynamic conditions.

The evaluation team witnessed system operation with one hotwell pump running under cold conditions (Ref. 101). The piping was observed to be vibrating (moving) horizontally approximately 1/2 inch.

Inspection of piping in the area also noted that a vertical drop of approximately 2 feet would occur only with a broken pipe support. Nuclear plant test deficiency report PT-174 (Ref. 102) noted the failure of the original designed restraints that were provided to reduce vibration, but neither the exact dimensions of this movement nor the encroachment on allowable stress levels could be established.

Corrective actions have been implemented via workplan 4711 (Ref. 103) and recent changes to operating and technical instructions TI-56.3, as suggested in a memorandum reporting on this problem (Refs. 104 and 105) and SOI-2 and 3.3 (Ref. 106) to mitigate deleterious operating conditions.

In a separate review of a similar issue regarding steam line failures (Ref. 107), it is stated:

> "TVA developed the 'power block' security concept in 1982 to reduce the NRC-required security area, and make the security operation more efficient. These secured areas are collectively known as the 'power block.'
>
> After study, the decision was made to locate a personnel access portal between columns M and K and T-1 and T-2 on elevation 708. . . of the turbine building. This is a search and security check-point similar to an airport security station which regulates access to the power block areas.
>
> This portal location is directly beneath four 36 inch diameter main steam lines that carry steam from the steam generators to the main turbine.
>
> . . . The hot steam released by [the postulated 'worst case' steam line rupture] accident would severely burn or kill anyone in the area of rupture, and would be of sufficient force to damage the portal."

This report also noted that a formal assessment of the hazards (catastrophic major steam line rupture) of the access portal location was conducted and it was determined that for ". . . those employees spending 8 hours in the portal or nearby the probability of death occurring from a pipe rupture is 7.34×10^{-8} or one in 13,623,978." Thus, it was concluded that "the access portal location represents an acceptable level of risk."

Although satisfactory resolutions have been achieved in those cases of vibration where component failures have occurred, inadequate attention has been given to the personnel risk involved with location of a personnel station in the area. TVA has agreed to conduct a risk assessment on the 10-inch steam generator blowdown line similar to that discussed above. Appropriate corrective action will be taken if it is deemed that there is an unacceptable risk. Pipeline rerouting, or relocation of the personnel station may be appropriate. This report has not yet been issued.

The concern is valid inasmuch as piping within the region of concern has experienced severe vibration in operation.

4.16 Defective Rockwell Valves - Element 232.20

This concern documents an NRC finding following review of the QTC files that a procurement audit had mentioned "defective Rockwell valves."

In an internal TVA communication (Ref. 108), a procurement department spokesman states:

> "The only problem that we are aware of that might lead somebody to believe that Rockwell supplied defective valves to TVA was the incident in which valves were hydro tested as [sic] ASME III class I pressures in lieu of ASME class 2 and/or class 3 pressures."

The incident referred to above was noted in an Office of Quality Assurance (Audit) deviation report (Ref. 109) which summarized:

> "Rockwell supplied to TVA in 1981 approximately 1600 valves for Sequoyah, Watts Bar, and Hartsville/Phipps Bend under Contract Nos.:
>
> 78KA2-824413; 78KA3-824497-2, 79K82-824770-3, and 77K53-820721-3
>
> Rockwell subsequently reported to TVA (after shipment) that Rockwell had performed the hydrostatic shell test 25 lb/in^2 lower than required [2150 psig versus 2175 psig]."

All four contracts noted above included the supply of ASME III, Class 2 and 3, 2-inch and smaller 600-pound class, carbon steel, welded end valves. Carbon steel, welded end 600-pound rated valves are suitable for a maximum system design pressure rating of approximately 1,480 psig at a design temperature of 100°F, or at somewhat lower pressures at elevated temperatures (1,350 psig at 200°F). Shop shell hydrostatic test pressure is specified by the code as 1.5 times the 100°F pressure rating, rounded up to the next 25 pounds. Thus for the case noted above, the hydrostatic test pressure would be 2,220 psig rounded up to 2,225 psig. Pertinent contract information is given in the following tabulation.

P.O. Date	Contract No.	Plant	Code ASME III	System	System Design Press., (psig)	System Design Temp. °F
05/13/77	77K53-820721-3	HTN/ PBN (a)	1974 edition, Summer 1976 Addenda	Various	Various (d)	
07/18/78	78KA2-824413	SQN	1977 edition, Summer 1977 Addenda	ERCW (b)	160	130
09/26/78	78KA3-824497-2	WBN	ditto	ERCW	160	130
10/25/78	79082-824770-3	WBN	ditto	ERCW/ HPFP (c)	275	130

(a) General Electric boiling water reactor NSS
(b) ERCW - essential raw cooling water
(c) HPFP - High Pressure Fire Protection
(d) 1420 psig (max) at 150°F

In dispositioning the above-mentioned deviation report, TVA engineers noted that in the years preceding these contracts, the ASME Code requirements for valve hydrostatic testing were in a state of flux. In a request to the ASME Code Committee seeking testing requirement clarification (Ref. 110), the code history was reconstructed as follows:

"1. In the 1971 Edition, when valves were first introduced into Section III as components, Class 1 valves 4-inch and under could be designed and hydrostatically tested to the flanged end requirements (2175 psi). A [sic] Class 2 and 3 valves were designed to ANSI B16.5-1968. Hydrostatic test pressure was specified at 2175 psi. The Summer 1973 Addenda changed the hydro test pressures to 2250 psi and weld end flanges (from MSS SP-66) and to 2150 psi for flanged end valves for Class 1 applications.

2. In the 1974 Code, Class 1 and Class 3 valves required a 2250 psi shell hydrostatic test, and Class 2 valves could be hydroed to either the requirements of ANSI B16.5-1968 (2175 psi) or MSS SP-66 (2250 psi). Class 1 valves 4 inches NPS and smaller could be designed and hydroed to the flanged end valve requirements (2150 psi).

3. The Winter 1974 Addenda to NC [Class 2] changed its requirement to 2250 psi by adding Table NC-3512(c)-2. Now all classes were the same at 2250 with the 4-inch exemption still available for Class 1 valves.

4. The Winter 1976 Addenda to NC changed NC-3512 to reference ANSI B16.5-1968, which called out 2175 psi as the hydrostatic test for 600-lb valves, but the revision did not delete Table NC-3512(c)-2, which still read 2250 psi for carbon steel. There was no parallel change to ND-3512 [Class 3], but this change transpired (probably editorially) with the printing of the 1977 edition of Subsection ND.

5. The Winter 1977 Addenda deleted all the rating tables and the hydrostatic test pressure tables in NC and ND and referenced ANSI B16.34-1977, which specified 1.5 times the 100°F rating pressure which is 1480 psi, or 2220 psi, rounded up to 2225 for welding end carbon valves.

6. In the Winter 1977 Addenda, the 4-inch exemption was deleted when NB-3513 [Class 1] was rewritten, and the hydro requirements in NB-3513 did not provide the exemption previously permitted for four inches and under Class 1 valves.

Therefore, on the TVA contract date, all Class 600 valves had to receive a shell hydrostatic test of 2250 psi, and Class 1 valves 4 inches NPS and under could be hydroed at 2150 psi. Rockwell International chose to manufacture and hydro the valves to Class 1 for the NPS 4 and under size, as permitted by NA/NCA-2134. However, the valves were stamped on the nameplates as Class 2 or Class 3 when so ordered, and Class 1 stress reports were not submitted with them. The NPV-1 Data Report Forms reflect the 2150 psi hydrostatic test pressure, although still showing Class 2 or 3 within the form. They were signed by an Authorized Nuclear Inspector."

In responding to TVA's clarification request, the ASME Codes and Standards Committee responded (Ref. 111):

"Our understanding of the questions in your inquiry and our replies are as follows:

Question 1: For Class 600 valves manufactured in accordance with the 1971 Edition with Summer 1973 Addenda up to and including the 1977 Edition with Winter 1977 Addenda, is it the intent of the Code that shell hydrostatic test pressure tables for Class 1 flanged end Class 600 valves be acceptable for all types of valves NPS 4 and less in lieu of the specified hydrostatic test pressure for welded end valves?

Reply 1: Yes.

Question 2: Are any changes required to the documentation and nameplates supplied with the valves described in Question 1?

Reply 2: No."

Note that the reference to Class 1 flanged end valve hydrostatic test pressure (Question 1) calls for 2150 psig.

Although this concern was technically valid at one time, the governing industry standards body agreed that the valves in question were acceptable for use in TVA nuclear power applications.

4.17 Summary of Subcategory Findings

The classified findings are summarized in Table 1. Class A and B findings indicate that there is no problem and therefore corrective action is not required. Class C, D, and E findings require corrective actions. The corrective action class, defined in the Glossary Supplement, is identified in the table by the numeral combined with the finding class. For example, the designation D6 in Table 1 indicates that the evaluated issue was found to be valid (finding Class D) and that a corrective action involving some type of evaluation is required (corrective action Class 6).

The summary of findings by classification is given in Table 2. Of the 42 findings identified by a classification in Table 2, 35 require no corrective action. Of the remaining, three issues had corrective action initiated before the ECTG evaluation, one had new corrective action identified, and three were peripheral issues uncovered during the ECTG evaluation. This table shows that at Watts Bar, where most of the issues were originated, three of the original 18 issues were found to be valid and require corrective action. Of those three, two had corrective action begun before the ECTG evaluation.

Of the seven negative findings noted: one identified a supplier quality problem which was resolved through correspondence concerning code interpretation (232.20); one involved nonconforming drainage piping (nonpressure boundary) attachments (232.2); two involved air handling unit drainage piping ice plugging, operational problems which one would expect to normally encounter and solve during commissioning (232.9). The remaining three require further investigation (as initial corrective action) to assess significance and to determine whether further corrective actions, such as design modifications are necessary. Of these latter three, one involves a perceived fear of pipe rupture in an area next to a personnel security station, which had been located in the vicinity of high energy piping as the result of a backfitting decision following initial plant design. The other two are potentially significant (232.3 and 232.8) inasmuch as the pressure boundary integrity of safety-related systems could be compromised. In the judgment of the evaluation team, two of these seven findings could have been avoided by more careful attention to design standards (232.8) and careful observance of design installation details (232.2). Early attention to NRC regulatory guides and adherence to Division of Engineering directives could have precluded the finding at Browns Ferry (232.3).

5. CORRECTIVE ACTIONS

Table 2 identifies seven findings where corrective action is required or has previously occurred. The corrective actions, along with their finding/corrective action classifications, are summarized in Table 3. Additional corrective action information is provided in Attachment B. The plant or plants to which a corrective action is applicable can be determined by reading the Corrective Action Tracking Document (CATD) column where the applicable plant is identified by CATD number, or in parentheses if no CATD exists.

From the Finding/Corrective Action Classification column of Table 1, it can be seen that of the seven corrective actions identified, two require hardware changes, one requires documentation in the form of an ASME Code interpretation, and four require further analysis to assess problem validity and to make modifications if and as necessary. Finally, with respect to corrective actions, Table 1 shows that of the 16 elements in this subcategory, 10 require no corrective action.

In all cases, the evaluation team found the corrective action plans to be acceptable to resolve the findings.

6. CAUSES

Table 3 identifies the cause for each problem requiring corrective action. In most cases, the experience of the evaluation team was used to establish the cause. However, when direct evidence linked a cause to a problem requiring corrective action, such evidence was taken into account.

For the seven corrective action descriptions listed in Table 3, six causes have been checked. They are shown in the table and totaled at the bottom. Rationale for selection of each of the causes follows:

Element 232.2 - There was a lack of design detail inasmuch as the required material for certain drain pipe attachments was not identified on installation drawings (WBN).

Element 232.3 - Because of the lack of documentation, compliance with the requirements of Regulatory Guide 1.36 cannot be established (BFN).

Element 232.8 - Certain piping calculations were identified which used a formula less conservative than that required by the applicable code (BFN).

Element 232.9 - Because of incomplete design bases, inadequately insulated drainage piping was routed through areas of subfreezing temperatures (SQN).

Element 232.9 - Because of incomplete design bases, condensate drain piping was allowed to be routed near the glycol supply piping (WBN).

Element 232.19 - The bases or judgments made in locating a personnel security station in the vicinity of high energy piping were not documented (WBN).

Element 232.20 - The valve manufacturer erred in the shop hydrostatic testing pressure set pressure selection (WBN).

As Table 3 shows, only the two findings noted in Element 232.9 were attributable to the same cause. In these two cases the cause was listed as "inadequate design bases," which apparently resulted from the application of a new technology; i.e., the ice condenser containment.

7. COLLECTIVE SIGNIFICANCE

Of the 42 issues expressed and evaluated in this subcategory, one issue required corrective action as a direct result of the employee concerns. Three issues had corrective actions in progress at the time of the evaluation, and three peripheral issues requiring corrective action were identified during the ECTG evaluation.

Because of the relatively low number of negative findings in this subcategory and the random nature of the causes, it can be concluded that the piping and valve design, limited to the issues evaluated in this report for the four nuclear plants reviewed, does not represent a significant technical problem. No broader issues can be identified in this area.

As has been noted in Section 4 above, two findings (232.3 and 232.8) could potentially result in changes to documentation, design margin, or hardware. Review of the TVA Corporate Nuclear Performance Plan did not establish any direct relationship with these findings. However, it is reasonable to conclude that the improvements suggested by the Plan in areas of commitment tracking and timely implementation of corrective actions should generally diminish the frequency and nature of the findings noted above. Evaluation team members did note, however, that some Browns Ferry engineering department members still were of the belief that piping calculations (232.8) were valid, even though non-code formulas had been identified.

The results of this subcategory will be combined with the other subcategory reports and reassessed for the Engineering category report.

TABLE 1

CLASSIFICATION OF FINDINGS AND CORRECTIVE ACTIONS

	Element	Issue/ Finding**	Finding/Corrective Action Class* SQN	WBN	BFN	BLN
214.0	Flushing Particles	a	-	A	-	A
		b	-	A	-	A
232.1	Accumulator Piping Size Change	a	-	A	-	-
		b	-	A	-	-
232.2	Carbon Steel vs. Stainless Steel Drainage Piping	a	B	B	-	-
		b	B	B	-	-
		c	-	E6	-	-
232.3	Incorrect Piping Insulation Material	a	A	A	A	A
		b	A	A	C6	A
232.4	Valve Seat Material Change	a	-	A	-	-
232.5	Building Column Not On Flow Diagrams	a	-	B	-	-
232.6	Rubber Gasket Deterioration	a	A	A	A	A
232.7	Socket Weld Gap Radiation Hot Spots	a	-	B	-	-
232.8	Piping Wall Thickness Tolerance	a	A	A	A	A
		b	-	-	E6	-

*Classification of Findings and Corrective Actions

A. Issue not valid.
No corrective action required.
B. Issue valid but consequences acceptable.
No corrective action required.
C. Issue valid. Corrective action initiated before ECTG evaluation.
D. Issue valid. Corrective action taken as a result of ECTG evaluation.
E. Peripheral issue uncovered during ECTG evaluation. Corrective action required.

1. Hardware
2. Procedure
3. Documentation
4. Training
5. Analysis
6. Evaluation
7. Other

**Defined in Attachment B.

TABLE 1 (Cont'd)

	Element	Issue/ Finding**	Finding/Corrective Action Class* SQN	WBN	BFN	BLN
232.9	Freezing of Condensate Lines	a	A	C1	-	-
		b	E1	-	-	-
232.10	Drilled Holes in Branch Connections	a	-	B	-	-
232.11	ERCW Chiller Piping	a	-	-	-	B
		b	-	-	-	B
		c	-	-	-	A
232.12	Butterfly Valve Seats	a	-	-	-	A
232.14	System Color Coding	a	-	B	-	-
232.19	Excessive Piping Movement	a	-	D6	-	-
232.20	Defective Rockwell Valves	a	-	C3	-	-

*Classification of Findings and Corrective Actions

A. Issue not valid. No corrective action required.
B. Issue valid but consequences acceptable. No corrective action required.
C. Issue valid. Corrective action initiated before ECTG evaluation.
D. Issue valid. Corrective action taken as a result of ECTG evaluation.
E. Peripheral issue uncovered during ECTG evaluation. Corrective action required.

1. Hardware
2. Procedure
3. Documentation
4. Training
5. Analysis
6. Evaluation
7. Other

**Defined in Attachment B.

TABLE 2

FINDINGS SUMMARY

	Classification of Findings	Plant SQN	Plant WBN	Plant BFN	Plant BLN	Total
A.	Issue not valid. No corrective action required.	5	9	3	8	25
B.	Issue valid but consequences acceptable. No corrective action required.	2	6	0	2	10
C.	Issue valid. Corrective action initiated before ECTG evaluation.	0	2	1	0	3
D.	Issue valid. Corrective action taken as a result of ECTG evaluation.	0	1	0	0	1
E.	Peripheral issue uncovered during ECTG evaluation. Corrective action required.	1	1	1	0	3
	Total	8	19	5	10	42

TABLE 3
MATRIX OF ELEMENTS, CORRECTIVE ACTIONS, AND CAUSES
SUBCATEGORY 26000

ELEM	FINDING/ CORRECTIVE ACTION CLASS.**	CORRECTIVE ACTION	CATD	CAUSES OF NEGATIVE FINDINGS *: MANAGEMENT EFFECTIVENESS: 1 Frag-mented Organ-iza-tion	2 Inade-quate Q-trng	3 Inade-quate Proce-dures	4 Proce-dures Not Fol-lowed	5 Inade-quate Com-muni-cation	6 Un-timely Res of Issues	7 Lack of Mgt Atten	DESIGN PROCESS EFFECTIVENESS: 8 Inade-quate Design Bases	9 Inade-quate Calcs	10 Inade-quate As-blt Recon-cil.	11 Lack of Design Detail	12 Engrg Judgmt not Docu-mented	13 Design Crit/ Commit Not Met	14 Insuf. Verif Docu-menta-tion	TECHNICAL ADEQUACY: 15 Stds Not Fol-lowed	16 Engrg Error	17 Vendor Error	Signifi-cance of Corrective Actions*: D	M	H
232.2	E6	Replace noncomplying floor drainage system components.	WBN 01											X							-	-	-
232.3	C6	Ascertain and ensure the adequacy of nonmetallic insulation.	BFN 01****														X				P	-	P
232.8	E6	Review piping calculations to ascertain code compliance regarding minimum pipe wall.	BFN 01															X			P	P	P
232.9	E1	Provide adequate insulation of air handling unit condensate drainage piping.	SQN 01								X										-	-	-
	C1	Relocate piping and provide adequate insulation of air handling unit condensate drainage piping.	(WBN)***								X										-	-	-
232.19	D6	Conduct risk assessment analysis regarding steam generator blowdown piping movements.	WBN 01												X						-	-	-
232.20	C3	Verify adequacy of procured Rockwell valves.	(WBN)***																	X	-	-	-
			TOTALS								2			1	1		1	1		1			

* Defined in the Glossary Supplement.

** Defined in Table 1.

*** No CATD exists.

**** Operations Category Report 313.07

GLOSSARY SUPPLEMENT FOR THE ENGINEERING CATEGORY

Causes of Negative Findings - the causes for findings that require corrective action are categorized as follows:

1. Fragmented organization - Lines of authority, responsibility, and accountability were not clearly defined.

2. Inadequate quality (Q) training - Personnel were not fully trained in the procedures established for design process control and in the maintenance of design documents, including audits.

3. Inadequate procedures - Design and modification control methods and procedures were difficient in establishing requirements and did not ensure an effective design control program in some areas.

4. Procedures not followed - Existing procedures controlling the design process were not fully adhered to.

5. Inadequate communications - Communication, coordination, and cooperation were not fully effective in supplying needed information within plants, between plants and organizations (e.g., Engineering, Construction, Licensing, and Operations), and between interorganizational disciplines and departments.

6. Untimely resolution of issues - Problems were not resolved in a timely manner, and their resolution was not aggressively pursued.

7. Lack of management attention - There was a lack of management attention in ensuring that programs required for an effective design process were established and implemented.

8. Inadequate design bases - Design bases were lacking, vague, or incomplete for design execution and verification and for design change evaluation.

9. Inadequate calculations - Design calculations were incomplete, used incorrect input or assumptions, or otherwise failed to fully demonstrate compliance with design requirements or support design output documents.

10. Inadequate as-built reconciliation - Reconciliation of design and licensing documents with plant as-built condition was lacking or incomplete.

11. Lack of design detail - Detail in design output documents was insufficient to ensure compliance with design requirements.

GLOSSARY SUPPLEMENT (Cont'd)

12. Failure to document engineering judgments - Documentation justifying engineering judgments used in the design process was lacking or incomplete.

13. Design criteria/commitments not met - Design.criteria or licensing commitments were not met.

14. Insufficient verification documentation - Documentation (Q) was insufficient to audit the adequacy of design and installation.

15. Standards not followed - Code or industry standards and practices were not complied with.

16. Engineering error - There were errors or oversights in the assumptions, methodology, or judgments used in the design process.

17. Vendor error - Vendor design or supplied items were deficient for the intended purpose.

Classification of Corrective Actions - corrective actions are classified as belonging to one or more of the following groups:

1. Hardware - physical plant changes

2. Procedure - changed or generated a procedure

3. Documentation - affected QA records

4. Training - required personnel education

5. Analysis - required design calculations, etc., to resolve

6. Evaluation - initial corrective action plan indicated a need to evaluate the issue before a definitive plan could be established. Therefore, all hardware, procedure, etc., changes are not yet known

7. Other - items not listed above

Peripheral Finding (Issue) - A negative finding that does not result directly from an employee concern but that was uncovered during the process of evaluating an employee concern. By definition, peripheral findings (issues) require corrective action.

Significance of Corrective Actions - The evaluation team's judgment as to the significance of the corrective actions listed in Table 3 is indicated in the

GLOSSARY SUPPLEMENT (Cont'd)

last three columns of the table. Significance is rated in accordance with the type or types of changes that may be expected to result from the corrective action. Changes are categorized as:

- Documentation change (D) - this is a change to any design input or output document (e.g. drawing, specification, calculation, or procedure) that does not result in a significant reduction in design margin.
- Change in design margin (M) - This is a change in design interpretation (minimum requirements vs actual capability) that results in a significant (outside normal limits of expected accuracy) change in the design margin. All designs include margins to allow for error and unforeseeable events. Changes in design margins are a normal and acceptable part of the design and construction process as long as the final design margins satisfy regulatory requirements and applicable codes and standards.
- Change of hardware (H) - This is a physical change to an existing plant structure or component that results from a change in the design basis, or that is required to correct an initially inadequate design or design error.

If the change resulting from the corrective action is judged to be significant, either an "A" for actual or "P" for potential is entered into the appropriate column of Table 3. Actual is distinguished from potential because corrective actions are not complete and, consequently, the scope of required changes may not be known. Corrective actions are judged to be significant if the resultant changes affect the overall quality, performance, or margin of a safety-related structure, system, or component.

ATTACHMENT A

EMPLOYEE CONCERNS
FOR SUBCATEGORY 26000

Attachment A -- lists, by element, each employee concern evaluated in the subcategory. The concern number is given along with notation of any other element or category with which the concern is shared, the plant sites to which it could be applicable are noted, the concern is quoted as received by TVA and characterized as safety related, not safety related, or safety significant.

ATTACHMENT A

EMPLOYEE CONCERNS FOR SUBCATEGORY 26000

ELEMENT	CONCERN NUMBER	PLANT LOCATION	APPLICABILITY SQN	WBN	BFN	BLN	CONCERN DESCRIPTION*
214.0	IN-85-638-001	WBN		X		X	"Local procedures QCT 4.36 and 3.14 require a check for size and type of particles flushed from pipes, but do not require checking on the volume of particles, or the size of particles flushed from sensing lines. Both procedures are based on G-39 and N3M.890 which are based on ANSI 45.2.1-1973 which does not require you to check for volume of particles. CI questions the ignoring of the particulate." (SR)
232.1	EX-85-002-002 (shared with 20300)	WBN		X			"Accumulators on Unit #2 had a 6" pipe going into accumulator. Error found in flow [calculation] and piping changed out to 10". Unit #1 still has 6" pipe. Reactor building pipe chase area, elev. 716'-745'. (Not all accumulators)." (SR)
232.2	IN-85-021-006	WBN	X	X			"The floor drain piping is carbon steel and should be stainless steel in reactor building 1 and 2." (SR)
	XX-85-127-002	SQN	X	X			"Sequoyah - floor drain piping is carbon steel and it should be stainless steel. Reactor Bldg." (SS)
232.3	EX-85-089-002	WBN	X	X	X	X	"CI feels that improper insulation materials were installed in many aspects of WBNP construction. CI expressed that most of the insulation installed was 'SOF' insulation ('Rock Wool'), covered by a metal sheath. CI stated that this type of insulation is easily damaged, and is subject to deterioration due to vibration over long periods of time. CI expressed that a 'Harder' type of insulation should have been used." (NO)
	I-85-106-SQN	SQN	X	X	X	X	"The employee was concerned about the quality of nonmetallic thermal insulation being installed on austenitic stainless steel safety-related systems. He stated that in past experience at other nuclear plants the types of insulation that TVA is using at Sequoyah would not be acceptable. He stated that he had questioned the use of these types of insulation about two years prior to the plant staff and had been told that everything was okay. He was looking for an independent assessment." (SR)
232.4	IN-85-301-003	WBN		X			"Valves are inferior at Watts Bar. Seats were already changed from hard seats to soft seats after hot functional testing." (SR)

* SR/NO/SS indicates safety related, not safety related, or safety significant per determination criteria in the ECTG Program manual and applied by TVA before evaluations.

ATTACHMENT A

EMPLOYEE CONCERNS FOR SUBCATEGORY 26000

ELEMENT	CONCERN NUMBER	PLANT LOCATION	APPLICABILITY SQN	WBN	BFN	BLN	CONCERN DESCRIPTION*
232.5	IN-85-388-008	WBN		X			"Building columns are not identified on the flow diagrams." (NO)
232.6	IN-85-400-002	WBN	X	X	X	X	"Rubber gaskets used in the ERCW, RCW, and other unspecified systems, exhibit deterioration over short periods of time: which could adversely impact plant nuclear safety due to gasket failure. No further specifics or other details are available." (SR)
232.7	IN-85-532-001	WBN		X			"The gap specified for socket welds is approx. 1/16 inch. No specific acceptance criteria is specified. Welding QC accepts installations from 1/32 inch - 5/32 inch. The larger gaps will allow contamination to accumulate making the socket welds hot spots for future rework or modifications." (SR)
232.8	IN-85-545-X06	WBN	X	X	X	X	"12-1/2% was the criteria established by Engineer Design for minimum wall thickness since 1978. One engineer was found to be using the less stringent formula contained in Code Section NB. The above applies to QA Class 2, and some Class 1, stainless steel pipe in Units 1 & 2." (SR)
232.9	IN-85-772-005	WBN	X	X			"Condensate line and glycol line running next to each other is not a good design. Air handlers in July/August '84 were filled with ice, causing icing of drains and clogging. Location: Ice Condensers (32 air handlers) located at the Head [of] circular stairway up from elev. 755'-0" in Reactor Building both units." (NO)
232.10	IN-86-085-003	WBN		X			"Holes were drilled in pipe subsequent to welding of branch connections, to permit flow through the branch connection. CI feels that this is a questionable practice. Pipe is 4" fire protection, aux. building, Unit 1, 692' elevation, at bottom of stairs, 8 off of floor, halfway between stairs and elevators." (SR)
232.11	BNP-QCP-10.35-11	BLN				X	"CI proposed design change to provide crossover ERCW tie between redundant trains of Chillers in VE, VF, and VK systems that were not implemented. Also, CI proposed adding a strainer in ERCW system which was not implemented. Also, there is an electrical contractor problem on VE, VF, and VK chillers that needs to be reviewed." (SR)
232.12	BNP-QCP-10.35-8-20	BLN				X	"KH system butterfly valves have rubber seats that tend to dry and crack." (NO)

* SR/NO/SS indicates safety related, not safety related, or safety significant per determination criteria in the ECTG Program manual and applied by TVA before evaluations.

ATTACHMENT A

EMPLOYEE CONCERNS FOR SUBCATEGORY 26000

ELEMENT	CONCERN NUMBER	PLANT LOCATION	APPLICABILITY SQN	WBN	BFN	BLN	CONCERN DESCRIPTION*
232.13			-	-	DELETED		
232.14	WBN-0245	WBN		X			"On the piping color code and lettering board the colors for number 6, 8 and 9 are the same. This could cause a bad problem." (NO)
232.15			-	-	DELETED		
232.16			-	-	DELETED		
232.17			-	-	DELETED		
232.18			-	-	DELETED		
232.19	IN-86-027-001 (shared with 30100)	WBN		X			"Overhead pipes in 708 portal move drastically during testing. On one occasion (Oct. 84) a pipe dropped approximately 2 f. CI has no further information." (SR)
232.20	HI-85-077-N08	WBN		X			"NRC identified the following concern from review of the QTC file: 'Defective Rockwell Valves.' Per review of the file, the concern appears to deal with defective Rockwell valves discovered in a procurement audit." (SR)

* SR/NO/SS indicates safety related, not safety related, or safety significant per determination criteria in the ECTG Program manual and applied by TVA before evaluations.

ATTACHMENT B

SUMMARY OF ISSUES, FINDINGS, AND CORRECTIVE ACTIONS FOR SUBCATEGORY 26000

Attachment B -- contains a summary of the element-level evaluations. Each issue is listed, by element number and plant, opposite its corresponding findings and corrective actions. The reader may trace a concern from Attachment A to an issue in Attachment B by using the element number and applicable plant. The reader may relate a corrective action description in Attachment B to causes and significance in Table 3 by using the CATD number that appears in Attachment B.

The term "Peripheral finding" in the issue column refers to a finding that occurred during the course of evaluating a concern but did not stem directly from an employee concern. These are classified as "E" in Tables 1 and 2 of this report.

Issues	Findings	Corrective Actions
****************** Element 214.0 - Flushing Particles ******************		
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. The present TVA approach to system cleanliness proof testing is questioned with the absence of particulate volume testing.	a. No basis for testing for the volume of particulates was found in the TVA design guidance identified in the concern, in NRC Regulatory Guide 1.37, or in ANSI standards. The QCT Procedure 4.36 system flushing and particulate "Proof Flush Sampling" test implicitly require particulate volume levels to be acceptable; i.e., the particulates collected on the wire mesh screen used for testing must exhibit only "occasional speckling."	a. No corrective action is required.
b. The basis for making a distinction between cleanliness testing of piping systems vs not testing instrument sensing lines is questioned.	b. Flushing requirements exist for both piping and instrument lines. Testing for particulate size is not required for instrument lines by TVA, NRC, or ANSI standards requirements. Instrument lines are nonflow lines, thus minimizing the quantities of particles being transported into the lines; also the instrument calibration and testing processes involve flushing the lines. Thus an actual test for particulates in instrument lines is not necessary.	b. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
a. The present TVA and ANSI N45.2.1-1973 approach to system cleanliness proof testing is questioned in regard to the absence of particulate volume testing.	a. ANSI N45.2.1-1973 represents the industrial standard for nuclear power plant fluid system cleanliness. This standard does not require testing for particulate volume, but does supply other means for establishing satisfactory piping cleanliness. BLN procedures conform to ANSI N45.2.1-1973.	a. No corrective action is required.

Issues	Findings	Corrective Actions
Element 214.0 - BLN (Continued)		
b. The basis for making a distinction between cleanliness testing of piping systems vs not testing instrument sensing lines is questioned.	b. ANSI N45.2.1-1973 does not specifically address instrument sensing lines. TVA procedure BNP-CTP-4.4 defines instrument sensing line flushing requirements for BLN. TVA's material and construction practices, coupled with the fact that nonflow instrument sensing lines are less susceptible to particulate contamination related problems, provide reasonable assurance that implementing procedure BNP-CTP-4.4 will produce acceptably clean instrument sensing lines.	b. No corrective action is required.
****************** Element 232.1 - Accumulator Piping Size Change ******************		
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. Calculation error resulted in WBN Unit 2 piping size change.	a. The Unit 2 piping size change resulted from an engineering evaluation to determine the costs and benefits of installing UHI on Unit 2, and the decision not to make this installation. The change did not result from a calculation error.	a. No corrective action is required.
b. Piping changes on Unit 2 were not retrofitted to Unit 1.	b. Piping size differences between Units 1 and 2 at the time of the EC resulted from the fact that UHI installation had been completed on Unit 1 before the Unit 2 reevaluation was performed. It was decided at that time to leave the Unit 1 UHI in place. The Unit 1/Unit 2 piping size differences resulted from having UHI on Unit 1 and not having UHI on Unit 2. Subsequently, UHI was disabled on Unit 1, and the accumulator piping was restored to 10 inch size, the same as on Unit 2.	b. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

Issues	Findings	Corrective Actions
****************** Element 232.2 - Carbon Steel vs. Stainless Steel Drainage Piping ******************		
SQN	SQN	SQN
a. The reactor building floor drain piping design drawings may have required the use of carbon steel instead of stainless steel piping material.	a. Both carbon steel and stainless steel materials have appropriate applications in the reactor building floor drainage piping systems, and both have been used in the design and installation. The superior serviceability of stainless steel piping is justified in those reactor building drainage systems where there is a potential for handling radioactive or borated water, and stainless steel material has been specified in the design for use in those systems.	a. No corrective action is required.
b. Carbon steel pipe may have been installed in the reactor building floor drain system instead of stainless steel as required by the design drawings.	b. Stainless steel piping has been installed in the reactor building drainage systems where required by the design. However, some installed floor drain gratings have been identified that are not in compliance with the installation drawings. This discrepancy is being corrected by TVA. The reactor building embedded drainage piping material is not a nuclear safety issue.	b. No further corrective action is required.

Issues	Findings	Corrective Actions
Element 232.2 - WBN	WBN	WBN
a. The reactor building floor drain piping design drawings may have required the use of carbon steel instead of stainless steel piping material.	a. Because of its superior serviceability, stainless steel piping is justified in those Reactor Building floor drain systems where there is a potential for handling radioactive or borated water. Stainless steel material has been specified in the design for use in those areas.	a. No corrective action is required.
b. Carbon steel pipe may have been installed in the reactor building floor drain system instead of stainless steel as required by the design drawings.	b. Carbon steel and stainless steel materials have appropriate applications in the Reactor Building floor drainage piping systems, and both have been used. Stainless steel piping has been installed in the Reactor Building floor drain systems where required by the design. Some carbon steel is used for nonradioactive service applications such as air cooling unit drains. The Reactor Building floor drain piping material is not a nuclear safety issue.	b. No corrective action is required.
c. Peripheral issue.	c. Some installed floor drain gratings and solid cover plates have been identified that are not in compliance with the installation drawings.	c. The problem description of CATD 232 02 WBN 01 states: "Some installed reactor building floor drain gratings and solid cover plates have been identified to be of nonstainless material contrary to installation drawings." TVA's Corrective Action Plan (CAP) (CATD 23202-WBN-01) states: "A more detailed investigation has revealed the following conditions: Unit 1 Temporary nonstainless steel (magnetic) covers have been installed over some of the floor drains in the Reactor Building on elevation 716.0. These temporary covers have been installed for the purpose of preventing debris from entering the drainage piping during ongoing modification work. This temporary measure is being controlled by

Issues	Findings	Corrective Actions
Element 232.2 - WBN (Continued)		
		Temporary Alteration Control Form, TACF, Nos. 1-83-4-77 and 1-84-191-77. The temporary covers are to be removed prior to hot functional testing of Unit 1 as indicated in the TACFs. An error was noted on drawing 47W476-2 in that it did not specify the required material for covers on square drains on elevation 702.78. A condition adverse to quality report (CAQR) will be issued to document this CAQ.* The corrective action will be to replace the existing gratings with gratings made of austenitic stainless steel. Unit 2 Temporary nonmagnetic covers are installed over floor drains on elevations 702.78 and 716.0 in accordance with note 4 on drawing 47W476-1. *A CAQR will be issued to document and correct the same condition in Unit 2." The evaluation team concurs with this corrective action.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

Issues	Findings	Corrective Actions
****************** Element 232.3 - Incorrect Piping Insulation Material ******************		
SQN	SQN	SQN
a. The physical durability of the soft (rock wool) type piping insulation material widely used at WBNP and at SQNP may not provide as satisfactory a service life as would a "harder" material.	a Commercially available mineral fiber type piping insulation has appropriate and economic applications in the SQN plant. The general serviceability of mineral fiber piping insulations is satisfactory when installed and maintained with reasonable care. At SQN mineral fiber piping insulations exposed to the vibration created by fluid flow in the insulated pipes during normal operation will not experience a significant loss of thermal performance over the long term when installed and maintained in accordance with the manufacturer's instructions.	a. No corrective action is required.
b. The types of nonmetallic thermal insulation being installed at SQN plant on austenitic stainless steel components in safety-related systems may be of unacceptable quality.	b. Regulatory Guide 1.36 requirements were recognized by TVA and all TVA nuclear plants were appropriately instructed in 1975 by Division of Power Production procedure DPM No. N75M9. TVA SQN Plant Standard Practice SQM35 (01/18/83) transcribed the Regulatory Guide 1.36 requirements from DPM No. N75M9 and was incorporated by reference into SQN Plant Procurement Procedure SQA 45, which governed the purchase of replacement insulation at the time of the NSRS investigation but did not require quality certification documentation. Subsequent to the NSRS investigation, the SQN plant staff completed a program of testing nonmetallic insulation installed and in Power Stores storage areas which verified and documented compliance with Regulatory Guide 1.36 requirements.	b. No corrective action is required.

Issues	Findings	Corrective Actions
Element 232.3 - SQN (Continued)		
	The SQN plant staff revised procedure SQM35 to incorporate NSRS report recommendations for documenting compliance with Regulatory Guide 1.36 requirements for new nonmetallic insulation purchases at SQN.	
WBN	WBN	WBN
a. The physical durability of the soft (rock wool) type piping insulation material widely used at WBNP and at SQNP may not provide as satisfactory a service life as would a "harder" material.	a. Commercially available mineral fiber type piping insulation has appropriate and economic applications in the WBN plant. The general serviceability of mineral fiber piping insulations is satisfactory when installed and maintained with reasonable care. At WBN mineral fiber piping insulations exposed to the vibration created by fluid flow in the insulated pipes during normal operation will not experience a significant loss of thermal performance over the long term when installed and maintained in accordance with the manufacturer's instructions.	a. No corrective action is required.
b. The types of nonmetallic thermal insulation being installed at WBN plant on austenitic stainless steel components in safety-related systems may be of unacceptable quality.	b. Regulatory Guide 1.36 requirements were recognized by TVA and all TVA nuclear plants were appropriately instructed in 1975 by Division of Power Production procedure DPM No. N75M3. TVA has incorporated procedures for documenting compliance with Regulatory Guide 1.36 requirements for new nonmetallic insulation at WBN. The WBN plant staff completed a program of testing nonmetallic insulation installed and in Power Stores storage areas which verified and documented compliance with Regulatory Guide 1.36 requirements.	b. No corrective action is required.

Issues	Findings	Corrective Actions
Element 232.3 - BFN	BFN	BFN
a. The physical durability of the soft (rock wool) type piping insulation material widely used at WBN and at BFN may not provide as satisfactory a service life as would a "harder" material.	a. Commercially available mineral fiber piping insulation has appropriate and economic applications in the BFN plant. The general serviceability of mineral fiber piping insulations is satisfactory when installed and maintained with reasonable care. At BFN, during normal operation, mineral fiber piping insulation, exposed to the vibration created by fluid flow in the insulated pipes, will not experience a significant loss of thermal performance over the long term when installed and maintained in accordance with the manufacturer's instructions.	a. No corrective action is required.
b. The types of nonmetallic thermal insulation originally installed and that procured for replacement at BFN for austenitic stainless steel components in safety-related systems may be of unacceptable quality.	b. Regulatory Guide 1.36 requirements were recognized by TVA, and all TVA nuclear plants were appropriately instructed in 1975 by Division of Power Production Procedure DPM N75M9. Although these requirements were noted and subsequently imposed on all TVA nuclear plants by DPM N75M9, it has been noted in the problem description of CATD No. 31307-BFN-1, (10/16/86) addressing this finding in ECSP Report No. 313.07 BFN that: "Browns Ferry has not evaluated the level of fluorides and chlorides in the nonmetallic insulation used on austenitic stainless steel safety-related piping." Thus, compliance with Regulatory Guide 1.36 has not been established. TVA's proposed action plan states: "(1) Establish onsite storage procedures to ensure that no unqualified insulation is available for use on stainless steel CSSC piping.	b. No further corrective action is required.

Issues	Findings	Corrective Actions
Element 232.3 - BFN (Continued)		
	(2) Complete Document Review (for piping 2" nominal diameter and smaller) in order to complete identification of CSSC stainless steel piping with non-metallic insulation.	
	(3) Document that non-metallic insulation installed on CSSC stainless steel piping meets requirements of Regulatory Guide 1.36 or replace with qualified insulation, where required. A 'walkdown' verification will probably be necessary for this documentation. Documentation will be provided on a safety analysis.	
	(4) Provide TVA drawings and construction specification for BFNP to document and control insulation installed on CSSC stainless steel piping. (The drawing and specification provision is also a corrective action response to BF-CAR-87-0012).	
	Items (1), (2), and (3) to be completed before U2C5 ~~startup approximately October 1, 1987, for unit 2 only.~~ The balance of the work to be completed by June 30, 1988."	
BLN	BLN	BLN
a. The physical durability of the soft (rock wool) type piping insulation material widely used at WBNP and at BLNP may not provide as satisfactory a service life as would a "harder" material.	a. Commercially available mineral fiber type piping insulation has appropriate and economic applications in the BLN plant. The general serviceability of mineral fiber piping insulations is satisfactory when installed and maintained with reasonable care.	a. No corrective action is required.
b. The types of nonmetallic thermal insulation being installed at SQN and BLN plant on austenitic stainless steel components in safety-related systems may be of unacceptable quality.	b. Insulation specified, procured, and installed at BLN for safety-related austenitic stainless steel systems is in conformance with the requirements of Regulatory Guide 1.36.	b. No corrective action is required.

Issues	Findings	Corrective Actions
****************** Element 232.4 - Valve Seat Material Change ******************		
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. The change from hard seats to soft seats after hot functional testing indicates inferior quality valves are installed at WBN plant.	a. Hard-seated discs on approximately 40 Kerotest 2-inch and under Y-type globe valves were replaced with "soft" disc material. The disc change was based on maintenance considerations only and was not an indication of inferior valves. Affected valves in nuclear safety related systems serve tank level transmitters and consequently are normally open. Closure of these valves is required only during permitted maintenance of the affected level transmitter. Severe damage to valves was determined to have been caused by overtorquing of the valve handle. Corrosion found in warehoused valves was determined to be the result of reinstalling wet packing after factory hydrostatic testing. No inoperable valves have been identified as a result of the valve refurbishment program.	a. No corrective action is required.

Issues	Findings	Corrective Actions
Element 232.4 - BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

****
Element 232.5 - Building Columns Not on Flow Diagrams
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Issues	Findings	Corrective Actions
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. Building column identifications, used as location references on drawings, were omitted from one or more flow diagrams.	a. Although this EC correctly states the current practice, the concern does not constitute a technical problem needing resolution.	a. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A).	(N/A)	(N/A)

Issues	Findings	Corrective Actions
****************** Element 232.6 - Rubber Gasket Deterioration ******************		
SQN	SQN	SQN
a. Rubber gaskets, used in flange joints, installed in safety-related raw water piping systems such as Essential Raw Cooling Water (ERCW), exhibit rapid deterioration and could impact plant nuclear safety.	a. The rubber gasket material specified and installed in certain systems at SQN plant complies with Section III of the ASME B&PV code for class 2 and 3 piping systems. The rubber gasket material installed at SQN plant is typical of the best industrial standard rubber gasket material commercially available. There is no evidence of rubber gasket failure (premature or otherwise) at SQN plant. Deterioration of gasket material does not present a safety issue.	a. No corrective action is required.
WBN	WBN	WBN
a. Rubber gaskets, used in flange joints, installed in safety-related raw water piping systems such as essential raw cooling water (ERCW), exhibit rapid deterioration and could impact plant nuclear safety.	a. The evaluation team's findings are as follows: o The rubber gasket material specified in certain safety-related systems at WBN complies with piping code requirements applicable to WBN. o The rubber gasket material installed at WBN is typical of the best industrial standard rubber gasket material commercially available. o There is no evidence of rubber gasket failure (premature or otherwise) at WBN, other than the instance covered in the employee concern. o Deterioration of gasket material does not present a safety issue.	a. No corrective action is required.

Issues	Findings	Corrective Actions
Element 232.6- BFN	BFN	BFN
a. Rubber gaskets, used in flange joints, installed in safety-related raw water piping systems such as essential raw cooling water (ERCW), exhibit rapid deterioration and could impact plant nuclear safety.	a. The evaluation team's findings are as follows: o The rubber gasket material specified in certain safety-related systems at BFN complies with piping code requirements applicable to BFN. o The rubber gasket material installed at BFN is typical of the best industrial standard rubber gasket material commercially available. o There is no evidence of rubber gasket failure (premature or otherwise) at BFN. o Deterioration of gasket material does not present a safety issue.	a. No corrective action is required.
BLN	BLN	BLN
a. Rubber gaskets, used in flange joints, installed in safety-related raw water piping systems such as essential raw cooling water (ERCW), exhibit rapid deterioration and could impact plant nuclear safety.	a. The evaluation team's findings are as follows: o The rubber gasket material specified in certain safety-related systems at BLN complies with piping code requirements applicable to BLN. o The rubber gasket material installed at BLN is typical of the best industrial standard rubber gasket material commercially available. o There is no evidence of rubber gasket failure (premature or otherwise) at BLN,other than the instance at WBN covered in the employee concern. o Deterioration of gasket material does not present a safety issue.	a. No corrective action is required.

Issues	Findings	Corrective Actions
****************** Element 232.7 - Socket Weld Gap Radiation Hot Spots ******************		
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. The internal crevice inherent with the use of a socket end weld fitting creates a "crud trap" for accumulating corrosion products present in fluid systems. The accumulation of radioactive deposits in the socket weld crevice results in a radiological "hot spot" and a radiation exposure hazard to plant personnel.	a. The concern does not identify any particular system where socket welds joints are used. There are many systems, including nuclear, where socket weld joints are appropriate. Elimination of socket welds in small pipe as a means of reducing potential crud traps was generally not considered in the WBN plant design. Variations in the magnitude of the radiation hazard resulting from socket weld gap size variations would not mitigate the attention such a hazard would receive under the WBN Plant Radiation Surveillance Program for maintaining data on exposures of and doses to station personnel.	a. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

****
Element 232.8 - Piping Wall Thickness Tolerance
****

Issues	Findings	Corrective Actions
SQN	SQN	SQN
a. The ASME Section III code formula for calculating required minimum wall thickness of stainless steel Class 1 and 2 piping may be less conservative than the 12-1/2 percent criterion established for WBN units 1 and 2 in 1978.	a. Piping design criteria established by TVA Division of Engineering Design has consistently required use of appropriate piping code rules to determine the minimum required pipe wall thickness of piping in safety and nonsafety class systems. 12-1/2 percent is the maximum permissible reduction of nominal wall thickness allowed by several ASTM standard specifications for furnishing seamless and welded carbon steel and stainless steel piping products. The manufacturing tolerance on nominal pipe wall thickness specified for material purchase is not an element in the piping code equation for determining minimum required pipe wall thickness and, thus, has no impact on the design conservatism.	a. No corrective action is required.
WBN	WBN	WBN
a. The ASME Section III code formulas (Subparagraphs NB/NC-3641.1), for calculating required minimum wall thicknesses of stainless steel Class 1 and Class 2 piping, respectively, may be less conservative than the 12.5 percent criterion established for WBN units 1 and 2 in 1978.	a. Piping design criteria established by TVA Division of Engineering Design WBN have consistently required use of appropriate piping code rules to determine the minimum required pipe wall thickness in safety and nonsafety class systems. A maximum permissible reduction of nominal wall thickness of 12.5 percent is allowed by several ASTM standard specifications for furnishing seamless and welded carbon steel and stainless steel piping products. The manufacturing tolerance on nominal pipe wall thickness specified for material purchase is not an element in the piping code equation for determining minimum required pipe wall thickness and, thus, has no impact on the design conservatism.	a. No corrective action is required.

Issues	Findings	Corrective Actions
Element 232.8 - BFN	BFN	BFN
a. The ASME Section III code formulas (Subparagraphs NB/NC-3641.1) for calculating required minimum wall thickness of stainless steel Class 1 and 2 piping may be less conservative than the 12-1/2 percent criterion established for WBN units 1 and 2 in 1978.	a. Piping design criteria established by TVA for the procurement of replacement piping require the use of appropriate piping code rules to determine the minimum required pipe wall thickness of piping in Class 1 (TVA Class A) stainless steel systems. A maximum permissible reduction of nominal wall thickness of 12.5 percent is allowed by several ASTM standard specifications for furnishing seamless and welded carbon steel and stainless steel piping products. The manufacturing tolerance on nominal pipe wall thickness specified for material purchase is not an element in the piping code equation for determining minimum required pipe wall thickness and, thus, has no impact on the design conservatism.	a. No corrective action is required.
b. Peripheral finding.	b. Calculations available for review indicate that the appropriate Code formula required for determining minimum pipe wall thickness for some safety-related systems were not used during initial plant design. Instead, a less conservative formula was used which yields pipe wall thickness values lower than required. Allowing the individual who originated the work to check that work is contrary to industry practice. This issue will be addressed separately in BFN Element 204.5, "Organization or Operating Procedures."	b. The problem description of CATD 232 08-BFN-01 states: "Calculations performed for some safety-related systems used a formula to calculate minimum pipe wall thickness which yields values less than allowed by the applicable industry code (B31.1.0-1967)." TVA's corrective action plan (CAP) (CATD 23208-BFN-01) states: "The Mechanical Calculation program for BFN which supports the Design Basis & Verification program will perform a technical adequacy review of the existing BFN mechanical calculations via sampling plan. A random sample of each type/area of essential mechanical calculations is planned. Types/areas of essential

Issues	Findings	Corrective Actions
Element 232.8 - BFN (Continued)		
		calculations that are identified to have common cause deficiencies (i.e., show a trend) will have their sample size increased. Since a question has been raised concerning some piping minimum wall thickness calculations performed by TVA, on NSSS systems, a review of the required essential piping minimum wall thickness calculations (performed by TVA) for NSSS systems is planned. This will include all piping connecting to the reactor coolant system and all other piping for which General Electric piping and valve specification 22A1045AB applied as addressed in ESCP Report No. 232.8(C). Other areas of TVA piping minimum wall thickness calculations will be sampled (such as non-NSSS systems). Essential calculations performed by TVA that are found to be inadequate per code requirements will be either revised, superseded, or obsoleted (if TVA is not responsible for the design of such systems). Calculations for which TVA is responsible for the design will be checked against the bill of materials to verify that the piping schedule which was ordered and installed meets the required minimum wall thickness. The calculation program (which includes this CAP) has been scheduled in P2. CAQs will be written for all essential calculations found to be deficient (either CAQR's or PIR's, as appropriate)."
		The evaluation team concurs with this corrective action.

Issues	Findings	Corrective Actions
Element 232.8 - BLN	BLN	BLN
a. The ASME Section III code formulas (Subparagraphs NB/NC-3641.1), for calculating required minimum wall thicknesses of stainless steel Class 1 and Class 2 piping, respectively, may be less conservative than the 12.5 percent criterion established for WBN units 1 and 2 in 1978.	a. Piping design criteria established by TVA Division of Engineering Design have consistently required use of appropriate piping code rules to determine the minimum required pipe wall thickness in safety and nonsafety class systems. A maximum permissible reduction of nominal wall thickness of 12.5 percent is allowed by several ASTM standard specifications for furnishing seamless and welded carbon steel and stainless steel piping products. The manufacturing tolerance on nominal pipe wall thickness specified for material purchase is not an element in the piping code equation for determining minimum required pipe wall thickness and, thus, has no impact on the design conservatism.	a. No corrective action is required.

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Element 232.9 - Freezing of Condensate Lines
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Issues	Findings	Corrective Actions
SQN	SQN	SQN
a. Proximity of cold glycol lines to ice condenser air handling units (AHUs) can cause freezing of condensate water drain lines from the AHUs, thus preventing drainage and causing clogging of AHUs.	a. The ice condenser AHU glycol and condensate drain lines are insulated, and the latter are heat traced. Proximity of the glycol and condensate drain lines is not the cause of condensate drain line freezing. Furthermore, at SQN the minimum distance between the glycol and condensate drain lines is more than 6 feet, which cannot be considered as "running next to each other" as stated in the WBN EC. The identified problems do not involve safety-related components.	a. No corrective action is required.

Issues	Findings	Corrective Actions
Element 232.9 - SQN (Continued)		
b. Peripheral finding.	b. The most probable cause of condensate drain line freezing would be related to the extensive amount of piping surrounded by a 10° to 15°F environment. Actual experience indicates that the most frequent direct cause of such freezing has been heat tracing failure. SQN has experienced a number of occurrences of freezing of AHU condensate drain lines, has identified specific causes for recent occurrences, and recommended corrective action accordingly. The SQN FSAR discussion related to freeze prevention provisions for the condensate drain lines is not clear.	b. The problem description of CATD 232 09 SQN 01 states: "Portions of the ice condenser AHU condensate drain lines may have insulation missing or in poor condition. The type of insulation used may not be suitable for the environmental conditions present during operation. Evaluation and repair or replacement may be necessary. Also the FSAR may need revision to clarify operational requirements of the AHU defrosting." TVA's corrective action plan (CAP) (TCAB-086, 04/12/87) states: "ECN L6468 will add insulation on the radial beams in the upper bay of the ice condenser. Modification will help maintain temperature in the upper bay, and reduce condensation which contributes to ice buildup. ECN L6468 has been issued, but is scheduled for implementation post U2 restart. In addition to L6468, work requests will be placed on portions of the condensate drain line which require new insulation. SI-108.1 & 108.2 requires weekly inspection for ice buildup in the upper bay. Any ice found is removed. Note 1: The proposed CAP will significantly reduce the amount of ice buildup found in the upper bay of the ice condenser. However, because of the subfreezing environment in the upper bay, this area will continue to serve as a condensing medium for warm moist air in upper containment. If any significant frost or ice buildup should occur after the CAP is implemented, then SI-108.1 & 108.2 will provide a suitable method for removing the frost or ice buildup." The evaluation team concurs with this corrective action.

Issues	Findings	Corrective Actions
Element 232.9 - WBN	WBN	WBN
a. Proximity of cold glycol lines to ice condenser air handling units (AHUs) can cause freezing of condensate water drain lines from the AHUs, thus preventing drainage and causing clogging of AHUs.	a. The concern correctly states the condition of proximity of the ice condenser air handling unit glycol and condensate lines at the time of the concern. The arrangement tends to minimize exposure of the condensate line to ambient air, and to maximize the effect of the cold glycol line on the condensate line in the particular area cited. This arrangement could cause freezing of the condensate line as described in the concern. Subsequent to the concern, insulation was removed from the condensate drain line to allow heating by ambient air. Also, the drain line was moved closer to the end wall, away from the glycol line, reducing the effect of the latter. These changes have apparently eliminated the freezing problem.	a. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

****
Element 232.10 - Drilled Holes in Branch Connections
****

SQN	SQN	SQN
(N/A)	(N/A)	(N/A)

Issues	Findings	Corrective Actions
Element 232.10 - WBN	WBN	WBN
a. Fire protection piping branch connection fabrication by welding an outlet fitting to the run pipe (header) then drilling a hole in the run pipe wall through the outlet fitting may not be adequate.	a. The installation procedure of attaching "threadolet" type branch connection fittings in the fire protection piping system by boring the hole in the header (run pipe) following attachment welding may have been employed. The procedure of drilling following welding is a well established technique approved by pipe-fitting fabricators. Applicable piping codes do not prohibit this procedure. Acceptable branch connections can be provided on small piping if care is taken in execution.	a. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

****
Element 232.11 - ERCW Chiller Piping
****

SQN	SQN	SQN
(N/A)	(N/A)	(N/A)

Issues	Findings	Corrective Actions
Element 232.11 - WBN	WBN	WBN
(N/A)	(N/A)	(N/A)
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
a. CI's proposed design change to provide crossover ties between redundant ERCW trains for chillers in the VK, VE, and VF systems was not accepted.	a. The ERCW system as designed meets regulatory requirements as to the single failure criterion. ERCW system cross-connections provided between units 1 and 2 provide essentially the same degree of system reliability as would the proposed cross-connections between the separate, redundant trains. Piping cross-connections as suggested would add little to the reliability of the HVAC systems. Cross-connections as suggested would add additional complexity because of regulatory considerations, such as the single failure and common mode failure criteria.	a. No corrective action is required.
b. CI's proposed design change to add a strainer in the ERCW system was not accepted.	b. Strainers have not been installed in the chiller ERCW control valve trim piping, because it has been determined that the problem noted was the result of corrosion of the valve internals, not of the carbon steel system piping. To address the scale and rust buildup on the pilot valve contact surfaces, nickel plating has been applied to the surfaces of the upper diaphragm plate and the internal surfaces of the cover plate, both internal to the valve.	b. No corrective action is required.
c. Electrical contactor problems exist in chiller units of the VE, VF, and VK systems.	c. Although a technical disagreement with the chiller package vendor did occur, no physical problems with these components can be identified.	c. No corrective action is required.

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Element 232.12 - Butterfly Valve Seats
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Issues	Findings	Corrective Actions
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
(N/A)	(N/A)	(N/A)
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
a. Condenser circulating water system rubber-seated butterfly valve seats tend to dry and crack.	a. No generic problem/s with rubber-seated butterfly valves in the condenser circulating water system (KH) could be identified. Reported butterfly valve seat problems in other cooling water systems were the result of problems encountered during plant warehousing, installation, or startup and not due to improper material selection. Components containing rubber components (valve seats, tank linings, etc.) currently in dry layup show signs of deterioration and cracking. (conditions known to prematurely age rubber components) A formal program to refurbish deteriorated rubber components at some future date has not been established. TVA will review replacement requirements on an as needed basis.	a. No further corrective action is required.

Issues	Findings	Corrective Actions
******************* Element 232.14 - System Color Coding *******************		
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. The use of the same system color code for three different systems could cause confusion.	a. The WBN system color coding selection was found to be reasonable and appropriate. It is unlikely that plant operators would be confused by the grouping of different systems under one color coding group.	a. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

****
Element 232.19 - Excessive Piping Movement
****

Issues	Findings	Corrective Actions
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. Piping located overhead in portal 708 has experienced large displacements during testing.	a. The evaluation team witnessed a 10-inch steam generator blowdown line within the area of question moving horizontally approximately 1/2 inch. The line noted above has a history of severe vibration and displacement problems caused by flashing flow conditions. Vibration problems associated with the feedwater recirculation piping within this same area have been resolved. Recent experiences of high pressure piping failures at Surrey and Mojave Power Plants warrant consideration of personnel safety concerns at the portal 708 area.	a. The problem description of CATD 232 19 WBN 01 states: "Sufficient justification has not been provided from a personnel safety standpoint for location of access structure near high pressure piping." TVA's corrective action plan (CAP) (TCAB-279, 03/13/87) states: "A risk assessment analysis will be made on the 10-inch steam generator blowdown line. This analysis will be similar to that done on the main steam lines in Subcategory Report 90700, Design as Related to Industrial Safety. Appropriate corrective action will be taken if it is determined that there is an unacceptable risk." The evaluation team concurs with this corrective action, also noting that pipe line rerouting or relocation of the personnel station may be appropriate.

Issues	Findings	Corrective Actions
Element 232.19 - BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)
****************** Element 232.20 - Defective Rockwell Valves ******************		
SQN	SQN	SQN
(N/A)	(N/A)	(N/A)
WBN	WBN	WBN
a. Defective Rockwell valves were discovered in a procurement audit.	a. The valves procured under the contracts in question were shop hydrostatically tested to a pressure of 2150 psig, 100 psig less than the pressure required by the Code for ASME III, Class 2 and 3 applications. However, noting the conflicting requirements specified in applicable standards and ASME III code editions, the governing ASME Codes and Standards Committee, in correspondence with TVA, has validated the acceptability of the hydrostatic testing conducted on these valves. Valves procured from Rockwell International under the contracts in question are suitable for use "as is" in TVA Class 2 and 3 nuclear power plant piping systems.	a. No corrective action is required.
BFN	BFN	BFN
(N/A)	(N/A)	(N/A)
BLN	BLN	BLN
(N/A)	(N/A)	(N/A)

ATTACHMENT C

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47BM476 Rev. 1	Piping Bill of Material, Reactor Building Annulus Floor Drains and Emb. Piping
47BM476-1 Rev. 2	Piping Bill of Material, Reactor Building Containment Drains and Emb. Piping

17. TVA, Watts Bar Nuclear Plant Drawings

47W851-1 Rev. 17	Mechanical Flow Diagram, Reactor Building Floor and Equipment Drains
47W476-1 Rev. 9	Annulus Floor Drains and Embedded Piping
47W476-2 Rev. 10	Reactor Building Containment Drains and Embedded Piping
47BM476 Rev. 1	Piping Bill of Material, Reactor Building Annulus Floor Drains and Emb. Piping
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21. TVA Specification No. 1475, "Thermal Insulation Materials for Piping and Equipment Inside the Containment for Sequoyah Nuclear Plant Units 1 and 2 and Thermal Insulation Materials for Piping, Equipment and Reactor Vessel Inside the Containment for Watts Bar Nuclear Plant Units 1 and 2," (undated)

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