DRAFT REV. 0

# CPRRG

# TASK GROUP 3

# REPORT OF

# SAFETY SIGNIFICANCE

February 1987

# CONTENTS

## ATTACHMENTS

# 1.0 EXECUTIVE SUMMARY

## 1.1 Background

By memorandum dated January 21, 1987, the NRC's Executive Director for Operations (EDO) charged the Comanche Peak Report Review Group (CPRRG) with the responsibility for (1) determining whether the current augmented review and inspection effort at Comanche Peak is sufficient to compensate for any identified weaknesses in Region IV's inspection programs, (2) examining issues relating to the processing and disposition of inspection findings of OIA Report 86-10, and (3) examining the safety significance of issues identified in OIA Report 86-10. In addition, the CPRRG was requested to review NRC Form 766, make recommendations regarding its use, and offer the EDO an opinion on the possibility of broader implications involving Region IV. Task Groups were formed to evaluate these issues. This report documents the results of Task Group 3's examination of the safety significance of the 34 issues identified in OIA Report 86-10.

## 1.2 Task Group

Task Group 3 was formed on February 2, 1987, and consists of a senior level NRR manager, a project manager, and senior technical experts in quality assurance, mechanical/structural engineering, auxiliary systems engineering and instrumentation and control systems. Individuals were selected for this effort based on demonstrated skill and ability in either a technical discipline or technical project management. The scope and depth in technical expertise of Task Group 3 provided the capability to address, in detail, each of the 34 issues.

## 1.3 Safety Concerns from other Task Groups

Task Group 3's charter included provisions for incorporating new concerns identified by either Task Group 1 or Task Group 2. Neither Task Group 1 nor Task Group 2 identified additional safety concerns for Task Group 3 to address. Accordingly, the scope of the Task Group 3 effort did not expand beyond the 34 issues identified in the OIA Report.

## 1.4 Methodology

The Task Group organized the 34 issues by technical discipline and evaluated each to determine its safety significance based on the assumption that the condition was as stated in OIA Report 86-10. A worst-case consequence evaluation was performed assuming that the concern existed without recognition or correction. When possible, related individual issues were grouped together and evaluated as a single larger concern. Hardware issues that developed from quality assurance concerns were evaluated to determine the worst-case consequence from both a hardware and a quality assurance perspective.

## 1.5 Conclusions and Recommendations

The majority of the issues identified in the OIA report revolve around questions regarding procedures, corrective action systems, audits, traceability, and documentation. In evaluating the worst case for safety significance, Task Group 3 postulated hardware deficiencies resulting from these administrative problems, although actual hardware deficiencies were never identified by the inspectors. In developing recommendations for followup, Task Group 3 looked at corrective actions broad enough to identify and address both documentation deficiencies as well as any hardware deficiencies that may exist.

Table 1 provides a summary of the conclusions reached by Task Group 3 on the 34 issues evaluated by addressing the safety significance for the worst-case, the adequacy for actions planned by the applicants or taken to date, and recommendations for followup. The Table also cross references each of the 34 issues to the section where it is discussed in this report.

TABLE 1

SUMMARY OF ISSUES RAISED IN OIA REPORT 86-10 APPENDIX MM

| Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|
| Inspection Report 85-07/05 | | | | | |
| 1. Failure to translate design criteria from NSSS vendor into installation specifications, procedures and drawings; and failure to control deviations from the requirements contained in these documents with regard to Unit 2 RPV installation. | Unresolved Item 446/8505-05 | Yes | Westinghouse evaluation performed but not reviewed by NRC-adequacy indeterminate. | 1. Perform a detailed review of Westinghouse's engineering evaluation;<br>2. Perform a visual inspection of the accessible reactor vessel surroundings during or after hot functional test. | 3.1 |
| 2. Failure to maintain tolerance required and failure to report tolerance deviations on a nonconformance report with regard to Unit 2 RV support brackets and shoes. | Unresolved Item 446/8505-06 | Yes | (Same as #1 IR 85-07/05 above) | (Same as #1 IR 85-07/05 above) In addition, examine the traveler procedure and nonconformance procedure to confirm that adequate controls are specified to assure that non-conformances are reported and and properly processed; and, that changes are reflected in the plant-specific design basis and maintained by the applicant or its' designee. | 3.2 |
| 3. Failure to perform audits of surveillances of reactor pressure vessel specifications, procedures and installation. | Unresolved Item 446/8505-07 | None | Adequate - CPRT Program Plan. | Monitor applicants' response to staff concerns raised in SSER 11 regarding their audit-program (CPRT Program Plan ISAP VII.c and VII.a.4). | 3.3 |

| | Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|---|
| 4. | For the CVCS spool piece, failure to maintain traceability of item by applicable specification and grade of material and heat number of heat code. | ~~No finding referenced in report.~~<br>dropped | Yes | None | 1. Reverify traceability of the spool piece (ensure right type of steel has been fitted into the CVCS piping system; and<br>2. Verify that an adequate quality control procedure exists for QA inspectors to witness the transfer of marking for material that would otherwise lose its traceability when cut into smaller sections. | 3.4 |
| 5. | Deferral of hydrostatic test on cold leg test subassembly. | ~~Deleted from report~~<br>dropped. | None | Hydrostatic test encompassing the cold leg test subassembly-adequate. | None | 3.5 |
| 6. | No objective evidence (records) that mixing blades had been inspected quarterly since trucks were placed in service in 1977. | Violation<br>(No response)<br>445/8507-04<br>446/8505-02 | Yes | Applicant to provide statistical data regarding uniformity of concrete -adequacy indeterminate. | 1. Review statistical data regarding uniformity of concrete; and<br>2. Review larger sample of concrete compression tests. | 3.6 |
| | <u>Inspection Report 85-14/11</u> | | | | | |
| 1. | FSAR 17.1.17 does not describe TU Electric records system. | Unresolved Item<br>445/8515-U-04<br>446/8511-U-04 | None | Applicants have concluded that their procedures do not adequately describe the final TU Electric review and the conversion of documents to records. TU Electric is drafting new procedures to correct this deficiency. | Review the revised TU Electric NEO procedures upon their completion and verify the implementation. Pending TG2 recommendations relating to enforcement, verify corrective actions proposed by the applicants. | 3.7 |

| Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|
| 2. QA manual does not address ANSI-N45.2.9 requirements/commitments. | Unresolved Item<br>445/8514-U-05<br>446/8511-U-05 | None | None | See #1 IR 85-14/11 | 3.7 |
| 3. TU Electric failed to have/use procedures to control shipment of original design records for piping to Stone and Webster, NY. | Violation<br>445/8514-V-02<br>446/8511-V-03 | None - May have an impact on alterations later should records be lost. | Procedure was developed, records were retrieved and copied, and personnel were trained to the procedure. | See #1 IR 85-14/11 | 3.7 |
| 4. Original design records shipped in cardboard boxes to Stone & Webster. | Dropped | None | None | See #1 IR 85-14/11 | 3.7 |
| 5. No backup copy of records made for records shipped to Stone & Webster. | Dropped | None | None | See #1 IR 85-14/11 | 3.7 |
| 6. Failure to control and account for QA design records transferred from site to Stone & Webster, NY. TU Electric stated design records shipped without making backup copy because cost too much. Also stated it was company policy to proceed at own risk. | Dropped | None | None | See #1 IR 85-14/11 | 3.7 |
| 7. Site records containment liner and mechanical penetration of Chicago Bridge and Iron shipped to Houston, Texas, in cardboard boxes. | Dropped | None - May have an impact on alterations later should records be lost. | None | See #1 IR 85-14/11 | 3.8 |

| Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|
| 8 No backup copy of records made for records shipped to Chicago Bridge & Iron. | Open Item<br>445/8514-01-01<br>446/8511-0-01 | None | None | See #1 IR 85-14/11 | 3.8 |
| TU Electric failed to inventory records sent to Chicago Bridge & Iron. | Unresolved Item<br>445/8514-U-06<br>446/8511-U-06 | None | None | See #1 IR 85-14/11 | 3.8 |
| 10. TU Electric audited CBI Houston and in the scope of report stated it included Criterion XVI, QA records but did not document the audit of records Violation criterion XVIII. | Dropped | None | Adequate-CPRT Program Plan | Monitor applicants response to staff concerns raised in SSER 11 regarding their audit program (CPRT Program Plan ISAP VII.c VII.a.4) | 3.8 |
| 11. Failure to preclude rain from entering QA intermediate records vault over several years time. | Open Item<br>445/8514-0-02<br>446/8511-0-02 | None | The water leakage was corrected | See #1 IR 85-14/11 | 3.7 |
| 12. Failure to preclude food and coffee pot from QA intermediate records vault. (Fire hazard). | Dropped | None | Coffee pot removed immediately | See #1 IR 85-14/11 | 3.7 |
| 13. Failure to install fire suppression system, drains, and a sloped floor at permanent vault. | Unresolved Item<br>445/8514-U-08<br>446/8511-U-08 | None | None | See #1 IR 85-14/11 | 3.7 |

| Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|
| 14. Plant records stand in folders or binders in open face cabinets at records center. | Unresolved Item<br>445/8514-U-07<br>446/8511-U-07 | None | TU Electric is installing an alarm to detect operation of the sprinkler system and personnel to the potential for flooding. | See #1 IR 85-14/11 | 3.7 |
| 15. Failure to provide temporary or permanent storage facility for records entered into the permanent records center then co-mingled with in-process documents in paper flow group. | Unresolved Item<br>445/8514-U-09<br>446/8511-U-09 | None | TU Electric has placed the documents in fire rated cabinets | See #1 IR 85-14/11 | 3.7 |
| [illegible] not identified at main distribution station (i.e., labels taken off or lost an E-309 electrodes at the main distribution station). | Dropped | Yes | None | Determine if individual electrodes are identified individually according to AWS or MIL-E-22200 recommendations with a "Type Mark," such as "309". If electrodes are not uniquely identified, applicants' should undertake a statistically valid sampling of stainless steel welds to determine whether and to what degree misidentification and misapplication occurred. | 3.9 |
| Inspection Report 85-16/13 | | | | | |
| 1. Failure to develop/implement procedure to demonstrate 50.55(e) deficiencies corrected. | Unresolved<br>445/8516-U-01<br>446/8513-U-01 | None | TU Electric has established a licensing commitment resolution process to track the timely completion of commitments made to the NRC. Further, there is a task force charterd to identify, validate and assure positive closure of CDRs. | 1. Applicants should revise Procedure NEO CS-1 to specify that all items reported under the procedure should first be recorded into an established corrective action system.<br>2. Perform an audit of the current CDR program to verify its adequacy and implementation. | 3.10 |

| | Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|---|
| 2. | Failure to revise implementing procedures containing 50.55(e) reporting. TU Electric failed to revise implementing procedures before corporate NEO Procedure CS-1 was implemented resulting in a conflict with five other procedures. | Unresolved<br>445/8516-U-01<br>445/8513-U-01 | None | None | See #1 IR 85-16/13 | 3.10 |
| 3. | Failure to maintain retrievable 50.55(e) files (i.e., could not produce record in almost a month). | Unresolved<br>445/8516-U-01<br>446/8513-U-01 | None | None | See #1 IR 85-16/13 | 3.10 |
| 4. | Failure to report to NRC actual corrective action taken on 50.55(e)s. | Unresolved<br>445/8526-U-01<br>446/8513-U-01 | None | None | See #1 IR 85-16/13 | 3.10 |
| 5. | TU Electric's 50.55(e) files not auditable. | Unresolved<br>445/8516-U-01<br>446/8513-U-01 | None | None | See #1 IR 85-16/13 | 3.10 |
| | Electric never responded to all aspects of IEB 79-14. | Open Item<br>445/8516-0-03<br>446/8513-0-03 | Yes | Applicants have ongoing programs that address aspects of IEB 79-14. Adequacy is indeterminate without further information. | Provide the applicants with a clear and concise written evaluation of the applicants' actions taken to date and specify additional actions required to close the issue of IEB 79-14. | 3.11/3.1 |

| | Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|---|
| 7. | TU Electric's IEB record files were incomplete (1982 and 1985). | Open Item<br>445/8516-0-05<br>446/8513-0-05 | Yes | Adequate - applicants' committed to perform a procedure and record review to ascertain the adequacy of the IEB Program. | 1. The applicants' review should assure that all Bulletins were received, processed, and corrective actions initiated through established programs.<br>2. Applicant's should revise procedure No. NOE-205 to clearly discuss the interface between the operating organization and the corrective actions systems.<br>3. Evaluate the effectiveness of applicants' program through an audit. | 3.12 |
| 8. | NAMCO switches IEB 79-28 were not properly identified on installation travelers. | Unresolved Item<br>445/8516-U-04<br>446/8513-U-04 | None | Adequate - as installed configuration satisfies the design requirements for the two switches--EQ CAP complete field verification on Class IE equipment with documentation. | 1. Assure applicants understand root cause for traveler mismatch with equipment;<br>2. Assure records system is structured such that equipment records affected by in-plant equipment changes are identified and tracked to completion;<br>3. Determine the cause and implication of the apparent breakdown of the development of travelers for the entire NAMCO switch replacement effort; and<br>4. Monitor the applicants' on-going programs (EQ CAP) to confirm that the correct switches have been installed. | 3.14 |

| Issue | Final Resolution | Safety Significance for Worst Case | Adequacy of Action Planned or Taken | Followup and Recommendations | Report Section |
|---|---|---|---|---|---|
| 9. Deficiency in TU Electric's procedures to handle IEBs. They do not describe how construction management personnel handle IEB's requiring action especially hardware repair, replacement and modification. | Unresolved<br>445/8516-U-02<br>446/8513-U-02 | Yes | See #7 IR 85-16/13 | See #7 IR 85-16/13 | 3.12 |
| 10. No focal point at TU Electric to track IEB actions. | Dropped | Yes | See #7 IR 85-16/13 | See #7 IR 85-16/13 | 3.12 |
| 11. TU Electric's internal letter stated that TU Electric did not identify a nonconformance on IEB 79-14 to NRC. | Dropped | None | None | None | 3.13 |
| 12. Insufficient evidence of successful testing of BISCO fire seals - filing of false report by BISCO - Validity of BISCO seal questioned. | Unresolved<br>445/8516-U-06<br>446/8513-U-06<br>445/8516-U-07<br>446/8516-U-07 | None | Adequate - reinstalling penetration seal AB-790-1022A in accordance with BISCO procedure - verification walkdown of all as-build penetration seals will be accomplished. | Reinspect BISCO seals and certification documentation to confirm final compliance with fire protection requirements (utilize IE's inspection report) | 3.15 |

## 2.0 INTRODUCTION

By memorandum dated January 21, 1987 to J. G. Davis, V. Stello, NRC's Executive Director for Operations, established the Comanche Peak Report Review Group (CPRRG). The purpose of this Group is to review the issues raised by the Office of Inspectors and Auditor (OIA) as a result of an investigation of wrong doing concerning the handling of safety issues by Region IV personnel at Comanche Peak. The CPRRG was assigned the task of reviewing the technical issues identified in OIA Report 86-10 and of determining the following:

- Whether the current augmented review and inspection effort at Comanche Peak is sufficient to compensate for any identified weakness in Region IV's QA inspection programs.
- Whether the issues when identified were appropriately handled as to process and disposition.
- The safety significance of the 34 issues identified in OIA Report 86-10.
- The purpose and significance of NRC Form 766 and to make appropriate recommendations concerning its use.

Finally, without expanding the specific tasks above, the CPRR was to offer the Executive Director for Operations (EDO) any judgement on whether it is likely that there are broader implications in Region IV.

This report assesses the safety significance and adequacy of corrective actions for the 34 issues identified in Attachment MM to OIA Report 86-10.

Section 3 of this report provides Task Group 3's evaluations of the issues: Issues are identified by Inspection Report and numbered and restated as they appear in Attachment MM to OIA Report 86-10. Section 3 includes 16 separate evaluations covering the 34 issues. When possible, related individual issues were grouped together and evaluated as a single larger concern. The format of the individual evaluations included in Section 3 makes those cases where issues were grouped obvious to the reader.

Issues are explained in the sub-sections of Section 3 entitled "Description." These "Description" sections elaborate on the concerns raised by the inspectors, reflecting additional information found through a review of the inspection reports, testimony and other documentation provided to the Task Group. Background information germane to the concerns is provided in the subsections entitled "Discussion."

For each issue the Task Group performed a worst-safety case assessment assuming that the issue exists as stated without recognition or correction. The results of these assessments are provided in the subsections entitled "Safety Significance." When sufficient information was available, the Task Group also performed a more realistic assessment of safety-significance.

For each issue, the Task Group evaluated the adequacy of the actions taken or planned by the applicant to resolve the concern. Where those actions did not, in the Task Group opinion, adequately resolve the concerns, additional actions were recommended. Recommendations include additional actions for both the applicant and NRC staff. The subsections entitled "Followups Actions and Recommendations" summarize this portion of the Task Group's effort. References for each evaluation are provided in the last sub-sections of each evaluation.

## ACKNOWLEDGEMENT

The principal members of the Task Group were supported by several staff members whose contributions permitted the Group to meet its established deadlines. We wish to specifically acknowledge the efforts of Walt Oliu, who provided philosphical and editorial guidance; Pam Shea, who provided typing and overall administrative support; and Ron Lipinski and Martin Hum, who provided technical support in the areas of structural and mechanical engineering.

## 3.0 ISSUES

### 3.1 Issue 1 from Inspection Report 85-05:

Failure to translate design criteria from NSSS vendor into installation specifications, procedures and drawings; and failure to control deviations from the requirements contained in these documents with regard to Unit 2 RPV installation.

#### Description:

During a routine inspection conducted from April 1, 1985 through June 21, 1985 to verify final placement of the Unit 2 reactor pressure vessel (RPV) and internals, NRC inspectors found that neither site-prepared installation drawings nor specifications which implemented the Westinghouse Nuclear Services Division (WNSD)-recommended procedures were available.(1) The concerns raised by the inspectors on this issue include lack of specific installation criteria on centering tolerances, levelness tolerances, and clearances between support brackets and support shoes.

The inspectors believed that the traveler used by the constructor, Brown & Root, was not an adequate specification on which to base installation of the RPV. They believed the original WNSD installation procedure should have been transcribed into a site specification or procedure.

#### Discussion:

During the construction of a nuclear power plant, the manufacturer's engineering design department produces design output documents that translate the design into specifications, procedures, and drawings that are used to actually construct the plant. These design documents must be controlled and any changes to them must be reviewed and approved by the original manufacturer.

The Unit 2 RPV at Comanche Peak was set in place using a WNSD procedure(1) as a guide to develop a document referred to as a traveler. The traveler is used at Comanche Peak to provide detailed instructions to personnel performing the work. It is intended to fulfill the NRC requirement to provide procedures and instructions appropriate to the circumstances, as specified in a Brown & Root procedure.(2) Brown & Root Traveler ME-79-248-5500(3) contained the essential information to install the RPV and showed concurrence signatures from the Westinghouse site representative and from the applicant's quality control staff.

The NRC inspectors reviewed Brown & Root Construction and Operation Traveler ME-79-248-5500.(3) Requirements in WNSD procedures were specified in the traveler. The inspectors found that worksheets attached to the traveler showed the pressure vessel to be centered and leveled within the established tolerances. However, Traveler Operation 19 required verification of a 0.002" to 0"-0.005" clearance between the support bracket and support shoe, after the shim plates were applied. Change 5 in the traveler subsequently approved a change in the allowed clearance by permitting clearances from 0.015 to 0.025 inch. The installation data reflected in attachment 3B of the traveler indicated an as-built clearance of 0.012 to 0.026 inch, a space which exceeds both the original and revised tolerances. The inspectors were concerned about the acceptability of these revised tolerances.

The inspectors also believed that the traveler was neither an adequate means to control the installation of the RPV, as required by 10 CFR 50, Appendix B, nor that the system provided for adequate control of changes to that traveler. The inspectors believed that the Construction Operation Traveler, which was based on the WNSD-recommended installation criteria, was not equivalent to first translating WNSD criteria into a CPSES specification, drawing, procedure, or instruction, and then issuing a traveler based on these documents. The inspectors' concern was that without following this sequence of document development, changes by site construction personnel to the traveler could be made without a formal design change being processed. Basically, the inspectors believed that a traveler was not equivalent to a detailed procedure or instruction.

Safety Significance:

The safety significance of this issue depends on whether the Unit 2 RPV was properly installed in accordance with the WNSD-recommended installation criteria.

There are two elements to this issue. One involves the Unit 2 RPV installation and the other relates to the adequacy of travelers as an effective document to control the quality of installed or erected components.

The Unit 2 RPV installation has been determined to be acceptable by WNSD. In a letter to the TU Electric,(4) WNSD states that variations slightly in excess of the 0.020-inch (cold) clearance requirement have no effect on the design analysis of the RPV support or on the reactor coolant loop system. It is not clear whether WNSD has based their conclusions on past experience or on a more detailed plant-specific engineering evaluation. In response to an NRC request for information,(5) the applicants stated that WNSD has recently indicated an evaluation exists for accepting the tolerances; however, the evaluation has not been received at the jobsite. In addition, WNSD has indicated that the final as-built stress reports (which will incorporate deviations encountered during the construction phase) will be issued after completion of construction.

In the worst case, a larger-than-specification clearance could permit excessive movement of reactor coolant loop components during plant operations or during a seismic event. Since the RPV is supported alternatively by four nozzles, the uneven clearances between support brackets and support shoes could create binding in the supports and thus induce local stresses larger than the design stress in the reactor nozzles, supports, or portions of the reactor coolant system. The worst-case scenario is that this excessive clearance could affect the design analysis (e.g., fatigue life) of the reactor vessel nozzles, supports, and reactor coolant loop system. Consequently, the worst-case scenario would have safety significance because the design life of the affected components could be reduced. However, since the detailed engineering analysis of the as-built condition is not available, the actual safety significance of this change in tolerances cannot be assessed at this time.

With regard to the adequacy of travelers as measures to control activities affecting quality, the operation traveler is designed to meet quality assurance criteria set forth in licensing documents and applicable codes and standards during the construction of CPSES. The fact that the WNSD vessel installation procedure was incorporated directly into the traveler has no safety significance. The major concern should be that the instructions, no matter what form they took, were accurate, clear, and correctly transmitted the design output criteria to the field where the work was performed. The controlling procedure states that when implemented as called for, the operation traveler system fully complies with the QA criteria and that no additional controls are required.

In the case of Traveler ME-79-248-5500, the tolerances and dimensions were taken directly from the WNSD generic document which recommended that the installation instructions be prepared by the applicants and reviewed by WNSD site representatives. The concurrence by the WNSD field engineer represents their acceptance of the traveler as an appropriate instruction and/or drawing for installation of the reactor vessel. In addition, the controlling procedure for travelers required that all changes, regardless of their significance, must be documented on the traveler, denote the operations affected, describe and identify the authority for making the required change, and include QA representative approval.

Followup Actions and Recommendations:

Without a review of WNSD's engineering evaluation, the adequacy of the applicants' corrective action cannot be assessed. Therefore, the Task Group recommends that a detailed review of WNSD's engineering acceptance of the as-built tolerances be conducted when the report becomes available to ensure that the excessive clearance noted does not have any affect on the design analysis of the reactor vessel supports or reactor coolant loop system. In addition, a visual inspection should be made of the accessible reactor vessel surroundings, especially the reactor vessel nozzles and supports areas, for any sign of distress, construction debris, or concrete cracking, either during or immediately after hot functional testing.

References:

1. Westinghouse Electric Corporation, Westinghouse Nuclear Service Division, "Procedure for Setting of Major NSSS Components," dated February 13, 1979.

2. Brown & Root, Inc. Procedure MCP-1, Installation of Mechanical Equipment and No. CP-CPM-6.3, Revision 6, Preparation, Approval and Control of Operation Travelers.

3. Brown & Root Construction and Operation Traveler, ME-79-248-5500, "Reactor Vessel Installation," dated April 10, 1979.

4. Letter from R. S. Howard (Westinghouse) to J. T. Merritt, Jr. (TU Electric), dated January 10, 1986.

5. Letter from W. G. Council (TU Electric) to U. S. Nuclear Regulatory Commission, Attn: Document Control Desk, dated February 10, 1987.

## 3.2 Issue 2 from Inspection Report 85-05:

Failure to maintain tolerance required and failure to report tolerance deviations on a nonconformance report with regard to Unit 2 RV support brackets and shoes.

### Description:

The NRC inspectors believed that the control of changes in traveler ME-79-248-5500,(1) in accordance with the traveler control procedure CP-CPM-6.3, Revision 5,(2) was not adequate and that a nonconformance report should have been prepared on the issue (See Section 3.1) of tolerances for Unit 2 reactor pressure vessel (RPV) clearances between support brackets and support shoes.

### Discussion:

The RPV for Unit 2 was installed using the constructor's (Brown & Root) traveler ME-79-248-5500, the pertient pages of which are appended to the end of this section. (Additional details can be found in Section 3.1.) Step 19 of the traveler states, "Verify cold clearance of .020" to 0"-.005" for each side shim. See attachment 3B also." Attachment 3B is a table of the measurements taken during installation of the RPV. Several of the measurements exceed the allowable tolerance.

The original tolerances specified on Operation Traveler ME-79-248-5500, Revision 0, were extracted from Westinghouse Nuclear Services Division's (WNSD) generic procedures. The operation traveler's installation tolerances were subsequently revised to reflect site-specific field conditions. WNSD's concurrence with the revised tolerances was documented by signature on the operation traveler. In addition, Westinghouse Water Reactors Division provided a letter, WPT-8148 dated January 10, 1986,(3) accepting the Unit 2 RPV shim installation tolerances.

With regard to the quality assurance aspects of the issue, the applicant requested and received approval for a change to the tolerances from the WNSD. This approval was recorded on the traveler as Revision 5, which states, in part, "During the verification of clearances it was revealed that the clearances at 64 (degrees) left and right shims, 158 (degrees) right shim only, 247 (degrees) right shim only, and 338 (degrees) left shim only did not meet the tolerance specified in op. #19."

The column on the traveler adjacent to the operation number is titled "Dept." and for operation 19 has the abbreviation "QCV." This means that the quality control staff verified the activity. Under those circumstances, the QC staff should have acknowledged that the operation deviated from the requirement. This deviation should have resulted in a nonconformance report being written against the activity. By way of acknowledging this fact, the applicants recognized that an engineering disposition was necessary and contacted WNSD to obtain it (in TU Electric's letter CPPA-48113, as indicated in their response to the Task Group 3's request for information.

### Safety Significance:

The safety significance of this issue depends on whether the RPV was properly installed in accordance with (Brown & Root Construction and Operation Traveler ME-79-248-5500," Reactor Vessel Installation."(1) As discussed in Section 3.1, the worst-case condition may have some safety significance.

As noted in Section 3.1, the Brown & Root Construction and Operation Traveler No. ME-79-248-5500 was issued, controlled, and changed in accordance with Brown & Root Procedure CP-CPM-6.3. In addition, this traveler was reviewed and accepted by the WNSD site representatives, as were changes to it. Revision 5 to the traveler, steps 19 and 19A, clearly states that if specified clearances cannot be met, then the WNSD site representative will determine specified the acceptability of the as-built clearance. In attachment 3B to the traveler, the WNSD representative reviewed and documented their acceptance of the clearance changes, thus satisfying all steps of the traveler. However, the practice of processing deviations outside of the nonconformance program subverts the intent of the program. The nonconformance program performs several functions, including systematic processing of deviations, controlled reviews and disposition by the engineering staff, and program trending for management review.

### Followup Actions and Recommendations:

The Task Group concludes that the followup actions and recommendations discussed in Section 3.1 adequately address the hardware aspects of this issue.

The Task Group recommends that the traveler procedure and nonconformance review and approval procedure be examined as to whether adequate controls are specified to assure that nonconformances are reported and properly processed. The Task Group further recommends that changes, such as those affecting the RPV, be documented in the plant-specific design bases and maintained by the applicant or an applicant designee.

### References:

1. Brown & Root Construction and Operation Traveler, ME-79-248-5500, "Reactor Vessel Installation," dated April 10, 1979.

2. Brown & Root, Inc., Procedure CP-CPM-6,3, Revision 5, "Preparation, Approval, and Control of Operation Travelers," dated December 13, 1978.

3. Letter from R. S. Howard (Westinghouse) to J. T. Merritt, Jr. (TU Electric),
dated January 10, 1986.

4. Letter from TU Electric to NRC, February 13, 1987.

## CONSTRUCTION OPERATION TRAVELER CONTINUATION

TRAVELER NO. ME-79-248-5500

ACTIVITY DESCRIPTION Reactor Vessel Installation

PAGE 5 OF 8 QB
conts. on page 6 [illegible]

PREPARED BY [illegible] DATE 3/20/79
REVIEWED BY R. A. [illegible] DATE 4-10-79
ANI REVIEW N/A DATE

| OP. NO. | DEPT. | OPERATION | QA/QC ENG | ANI |
|---|---|---|---|---|
| 14 | M/W<br>QCW | Take gauge readings of the spaces on either side of the vessel support pads and record on Attachment # 3 . Also recorded cavity temperature. 88°F | R. J.<br>7-30-79 | |
| 15 | M/W<br>QCW | Machine the side shims to the thickness recorded in Operation # 14 . minus .020" for cold clearance. Tolerance is + 0, - .001" temperature ± 5°F. 85°F | R. J.<br>7-30-79 | |
| 16 | M/W<br>QCW | Coat the surfaces of the side shims which will be in contact with the vessel support pads with "molykote type Z" dry film lubricant. Buff to improve) finish. | R. J.<br>7-30-79 | |
| 17 | M/W | Install the side shim in the support shoes. Raise the vessel to a height necessary for top installation of the shims. Bolt the shim keepers onto the support shoes. | | |
| 18 | M/W | Lower the vessel and check levelness and vessel axis for orientation with containment axis. Record and obtain approvals on Attachment # 4 . | | |
| △5 19 | M/W<br>QCW | Verify cold clearance of .020" to 0" - .005" for each side shim. SEE Attachment 3B ALSO | R. J.<br>7-26-79 | |
| 20 | QA | QA review of operation traveler. R. J. II 10/2/79 | Per. Rev 5F @ Insp. | |
| Rev △ 1<br>R Bratt (Eng) 4/20/79<br>R J G II 5-2-79 | M/W | Assemble RPV support column/shoe for each support and temporarily install in place so as to set the 3" support column base plate to the proper elevation which ultimately | | |

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## CONSTRUCTION OPERATION TRAVELER CONTINUATION

TRAVELER NO ME 79 - 248 - 5500

ACTIVITY DESCRIPTION Reactor Vessel Installation

PAGE 7 OF 8

PREPARED BY C. B. DATE 7/26/79

REVIEWED BY C. W. DATE 7-26-79

ANI REVIEW ______ DATE ______

| OP. NO. | DEPT. | OPERATION | QA/QC ENG | ANI |
|---|---|---|---|---|
| 5<br>19 | MW<br>QCV | Change the operation to read:<br>" Verify cold clearance of .015" to .025" for each side shim". | R. J.<br>7.26.79 | |
| 5<br>19A | MW | During the verification of clearances it was revealed that the clearances at 67° Left and Right shims, 158° Right shim only, 247° Right shim only and 338° Left shim only did not meet the tolerance specified in op #19.<br>Follow the procedure specified below:<br>Take new gauge readings for the locations specified above on attachment # 3A. Also, record the clearances for the remaining shims (which were within the tolerance and checked upon the acceptance of Levelness and orientation of the Reactor Vessel) in the space provided below. Remachine the shims for the locations with new gauge readings (att # 3A) and reinstall. Verify that the clearances at these shims are within the tolerance of op. # 19 and also that the at the remaining shims are within ±.003" with tol. clearances documented below.<br>If these tolerances may not be met then Westinghouse representative will determine the acceptibility of the clearances. Document below the clearance any out of tolerances & actions CB 7/26/79<br>SEE Attachment 3B.<br>Cont. on<br>( Next Page) | | |

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## 3.3 Issue 3 from Inspection Report 85-05:

Failure to perform audits or surveillance of reactor pressure vessel specifications, procedures, and installation.

### Description:

Based on their inspection, NRC inspectors concluded that the failure to perform audits or surveillance of the Unit 2 reactor pressure vessel (RPV) installation, specifications, hardware placement, or as-built records, was a violation of Appendix B, Criterion XVIII, "Audits."

### Discussion:

The applicant is required to establish a quality assurance program that will function during the construction and operation phases of the plant that complies with 10 CFR 50, Appendix B criteria. One requirement of the quality assurance program is that it audit the quality assurance function itself. The quality assurance program, although designed to assure product quality, is also a management mechanism for checking the production process and for verifying that the process is functioning properly. One method of gathering the information to make this determination is an audit.

The audit program should be planned to cover those aspects of the quality and production process that are key indicators of the overall process. This function is accomplished through the development of an audit plan. Such audits do not (because they cannot) examine every characteristic of all elements of the process. The selective audit information gathered should then be reviewed by management to determine if corrective actions are warranted.(1)

During a routine inspection, the NRC inspector requested the applicant's records of QA audits or surveillances for the Unit 2 RPV installation. The applicant did not provide any documentation of an audit or surveillance which evaluated specified placement criteria, placement procedures, hardware placement, or as-built records.

### Safety Significance:

The failure to perform an audit of the Unit 2 RPV installation may not be significant by itself. The issue must be viewed in the context of the overall audit plan to determine if the plan is comprehensive. The audit is intended to sample the quality assurance program and verify that it is working, whereas a quality control activity is directly related to determining product acceptability.

There is no direct equipment safety significance resulting from failure to audit the installation of the RPV. The quality of the installation was monitored by quality control personnel and documented on the traveler. However, there may be a broader concern if the audit plan was deficient or management was not reviewing the audit results and taking appropriate corrective action.

If the audit plan was deficient, there would be inadequate monitoring of the overall QA program, and deficient areas would remain undetected. Because of this, management would not be informed about and thus unable to implement needed corrective actions.

This problem has been identified in NUREG-0797, Supplement No. 11, Appendix O, Section 3.2.11, and Appendix P, Section 4.7.

The Comanche Peak Technical Reviews Team (TRT) found that the TU Electric audit group consisted of only four auditors during the 1981-1982 period and that TU Electric management was not sufficiently committed to quality assurance. The TRT cited failure to perform management assessments and overview of the effectiveness of the quality assurance program. Further, they determined that all aspects of safety-related activities were not audited, and that there was procedural implementation inadequacies, questionable auditor qualifications, incomplete assessments of the QA program on an annual basis, and inadequate corrective actions to prevent a recurrence of deficiencies.

The applicants have responded to the issues raised by the TRT and others by forming the Comanche Peak Response Team which will address these issues.(1)

Recommendations and Followup:

The followup of this issue will be addressed by the Comanche Peak Response Team (CPRT) as stated in NUREG-0797, Supplement 13, Abstract, "The NRC staff concludes that the CPRT Program Plan provides an overall structure for addressing all existing and any future issues which may be identified from further evaluations..."

The CPRT has issued the Comanche Peak Response Team Program Plan and Issue - Specific Action Plans (ISAP) which contain a section which addresses quality of construction and QA/QC adequacy program plans(3). The basis of this plan is reinspection of hardware to establish that there are no undetected or uncorrected deficiencies. Further, a section exists which specifically addresses applicant audit programs and auditor qualifications.(4)

References:

1. Quality Control Handbook, J. M. Juran, Third Edition, PP 21-10 to 21-13.

2. Letter from E. H. Johnson (NRC) to Texas Utilities Electric Company, Attn: W. G. Counsil, dated February 3, 1986.

3. Comanche Peak Response Team Action Plan, ISAP VII.c, Revision 1, "Construction Reinspection/Documentation Review Plan," dated January 24, 1986.

4. Comanche Peak Response Team Action Plan, ISAP VII.a.4, Revision 1, "Audit Program and Auditor Qualification," dated January 24, 1986.

### 3.4 Issue 4 from Inspection Reports 85-07/05:

For the CVCS spool piece, failure to maintain traceability of item by applicable specification and grade of material and heat number or heat code.

Description:

Based on his inspection, the NRC inspector concluded that the failure to mark the chemical volume control system (CVCS) spool piece 3Q1 (DWG No. BRP-CS-2-RB-76) with material specification and grade, heat number, or heat code, was a violation of 10 CFR 50, Appendix B, Criterion VIII, "Identification and Control of Materials, Parts, and Components," and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1974 Article NA 3766.6.

Discussion:

Bulk piping material is manufactured with identifying markings located intermitently along its length. The markings, specified by the American Standard for Testing of Materials (ASTM), include information about the type of material and the heat number. These markings permit the user to trace the product back to the certified material test report, which attests to the chemical and physical characteristics of the piping. The piping material is manufactured in lengths ranging up to 30 feet. These lengths are cut and assembled into piping systems at the piping fabricators or at the construction site. Relatively short pieces of piping are referred to as "spool pieces."

When spool pieces are cut from bulk piping in a way that produces a section without the manufacturer's marking, then the user must transfer the marking or a unique code that permits it to be traced back to the certified material test report. These cutting operations are witnessed by quality control inspectors who can attest to the proper transfer of the marking.

Article NA 3766.6 of ASME Code Section III, 1974 requires that the piping to be identified be marked with the applicable specification and grade of material, heat number, or heat code of the material, and any additional marking required by Section III, to facilitate traceability of the reports of the results of all tests and examinations performed on the material. Alternatively, the Code permits a marking symbol or code to be used which identifies the piping with the material's certification and requires that such a symbol or code shall be explained in the certificate.

The NRC inspector claimed that the spool piece in question had been marked with the spool piece number (3Q1) and the Brown & Root (B&R) drawing number but that he could not find the material specification number and type, heat code, or other means of traceability. If it can be verified that the spool piece number provides a unique identification marking and that B&R

drawing number number (as-built sketch) provides traceability to a tabulation of materials which contains all information required by the Code, then the spool piece markings have satisfied all the traceability requirements of the Code.

Safety Significance:

This issue has no safety significance if the material identification can be positively established and the integrity of the spool piece field welds can be ascertained; otherwise, this spool piece should be replaced.

With regard to the safety significance of the worst-case scenario, it depends on whether this spool piece is made of austenitic stainless steel or a carbon steel. If it is carbon steel and the welder did not identify the mistake during welding, then the issue is safety significant. For this case, the spool piece will develop a leak or break rather quickly, either during preoperation testing or early in the plant life, because the borated reactor coolant water is known for aggressively corroding carbon steel rapidly.(3)

There are three potential consequences from failure of the spool piece. If it is located in the CVCS safety-related piping system (i.e., the safety injection system) and inside the reactor coolant system (RCS) pressure boundry, its failure would have safety significance. If it is located in the safety-related portion of the CVCS, but outside the RCS pressure boundry, its failure could result in a total loss of RCS makeup and safety injection. If the spool piece is located in nonsafety-related piping of the CVCS, the consequences of a break would be minimal.

Usually, the CVCS is isolated during a LOCA, except for the centrifugal charging pumps and the piping in the safety injection path. However, this problem can easily be detected by a simple check using a magnet to make sure that the spool piece is not made of carbon steel (austenetic stainless steel is nonmagnetic).

If instead of carbon steel, the wrong type of austenitic stainless steel piping was used, these degradations will not happen because stainless steel is highly resistant to corrosion in the primary coolant environment.

In summary, although the worst-case scenario would have safety significance, the problem can easily be alleviated by verifying that a carbon steel spool piece was not used in the system. The FSAR, Table 9.3-7 indicates that the piece in question is made of austenitic stainless steel.

Followup Actions and Recommendations:

There are no corrective actions planned by the applicant with regard to this issue. The Task Group recommends that the traceability of this spool piece be reverified, if not already done, so as to ensure that the right type of steel has been fitted into the CVCS piping system.

Furthermore, it should be verified that an adequate quality control procedure exists for quality control inspectors to witness the transfer of markings on piping that would otherwise lose its traceability when cut into smaller sections.

References:

1. Letter from E. H. Johnson (NRC) to Texas Utilities Electric Company, Attn: W. G. Counsil, dated February 3, 1986.

2. Comanche Peak Steam Electric Station, Final Safety Analysis Report, Section 9.3.4.1.1.7, page 9.3-33.

3. Czajkowski, C. J. NUREG/CR-2827, "Boric Acid Corrosion of Ferritic Reactor Components," July 1982.

## 3.5 Issue 5 from Inspection Reports 85-07/05:

Deferral of hydrostatic test on cold leg test subassembly.

### Description:

Based on their inspection, NRC inspectors concluded that according to their interpretation of the ASME Code, Section III,(1) the cold leg of the reactor coolant system should have been hydrostatically tested in the vendor shop prior to shipment to the site. They also maintained that the NPP-1 Form should not have been signed and that the NPT stamp should not have been applied.

### Discussion:

Title 10 of the Code of Federal Regulations, Section 50.55a, requires that reactor coolant pressure boundary systems be designed, fabricated and installed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III.

Paragraph NB-6221 of ASME Code, Section III, requires that completed components and appurtenances, with certain exceptions, be subjected to a hydrostatic test at a pressure not less than 1.25 times the system design pressure prior to installation in the system. This paragraph also permits substitution of a system hydrostatic test for a component hydrostatic test provided, if required, that: (1) the component can be repaired by welding as a result of the system hydrostatic test, (2) the component repair can be postweld heat treated and nondestructively examined subsequent to the system hydrostatic test, and (3) the component is subjected to a minimum required system hydrostatic test following the completion of repair and examination. In addition, paragraphs NA-1210 and NA-1232 of the Code clearly specify that piping subassemblies are sections of a piping system and, therefore, are not components. Since piping subassemblies are not defined as components, the requirements of NB-6221 and NA-8230, which governs the application of the appropriate code symbol only after the hydrostatic test, do not apply to the cold leg piping subassembly.

### Safety Significance:

This issue was raised because of an incorrect interpretation of the hydrostatic test requirements of the ASME Code, Section III; it therefore has no safety significance. The worst case scenario would occur if this piping subassembly was not hydrostatically tested, was subsequently installed in the piping system, and during the system hydrostatic test, ruptured due to overstress. Therefore, the worst case has no safety significance.

### Followup Actions and Recommendations:

The NRC inspection program includes a procedure, IE Inspection Procedure 70462, "Reactor Coolant System Hydrostatic Test Witnessing," which requires the NRC inspector to witness this test. Therefore, it is extremely unlikely the piping would not be tested. In view of this requirement, recommendations are unnecessary.

Reference:

1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Subsections NA and NB, 1974 Edition, July 1, 1974.

## 3.6 Issue 6 from Inspection Reports 85-07/05:

No objective evidence (records) that mixer blades had been inspected quarterly since they were placed in service in 1977.

Description:

During inspections of Comanche Peak Steam Electric Station (CPSES) conducted from April 1, 1985 through June 21, 1985, NRC inspectors identified a lack of records indicating that mixing blades in concrete mixers had been periodically inspected.(1)

Discussion:

In the FSAR the applicant committed to American Concrete Institute (ACI) Standard ACI-304-73.(2) ACI-304, Section 4.2, states that "Mixers should be properly maintained to prevent mortar or dry material leakage and inner mixer surfaces should be kept clean and worn blades replaced." It does not define the frequency or the extent of the inspections in order to achieve "proper maintenance." The same document, in Section 4.4, further states that "The performance of mixers is usually determined by a series of uniformity tests made on samples taken from two to three locations within the concrete batch being mixed for a given time period." It also states that various tests, such as air content, slump, etc., are used to check mixer performance. B&R Procedure No. 35-1195-CCP-10,(3) Section 3.9.4, states that "Mixer blades shall be replaced when they have lost 10 percent of their original height," and that the "concrete superintendent or his representative shall perform a quarterly check to ensure performance." This requirement pertains to both, Section 3.9, "Central Mixing", and to Section 3.11, "Delivery of Concrete in Truck Mixer to Point of Placement," of Procedure No. 35-1195-CCP-10 by reference in Section 3.11.6. When the NRC inspector asked for evidence that the blades had been checked for wear on a quarterly basis as required by the procedure, he was told that such records could not be produced.

Safety Significance:

Neither B&R Procedure 35-1195-CCP-10(3) nor ACI 304-73(2) require documentation for the inspection of mixer blades. The B&R procedure is more specific on this subject and stipulates that checks of the condition of the mixer, including the blades, should be made on a quarterly basis. It would, therefore, be prudent on the part of the contractor to keep some kind of records to verify that such maintenance inspections took place.

Inspection of mixer blades is one of the many precautionary measures taken during construction in order to ascertain that the concrete quality will meet job specifications. The ultimate proof of concrete adequacy results from the compression tests performed on concrete cylinder samples taken from each pour. In the worst case, if the mixer blades were worn out to the extent that the quality of the concrete could be affected, it would be evidenced by failure in compression tests and lack of uniformity of the concrete compressive strength. Failure to meet the job specifications would constitute the "worst-case" scenario. This condition would have some safety signficance if concrete below specified strength were used in the containment structure or in concrete supports for safety-related components.

Although ACI-304 recommends that mixer blades be maintained, it emphasizes the fact that mixer performance is evident in its final product, i.e., the quality and consistency of concrete. Tests for air content, slump, unit weight of air-free mortar, compressive strength, water content, etc., are the most common ways to check mixer performance. Following the philosophy of the ACI, the Task Group requested that the applicant submit the following additional information:

1. Three sample cylinder test records spaced so that one is from the beginning, one in the middle, and one at the end of the period considered.

2. The records of the minimum compressive concrete strength for the period in question.

3. A record of statistical distribution of the concrete test results as far as available for the period considered.

By letter dated February 10, 1987,(4) the applicant provided the information to the Task Group consisting of the records of compression tests on concrete cylinders on July 29, 1977, August 19, 1977, February 5, 1980, and May 18, 1984. In all cases the 28-day compressive strength of the concrete exceeded the specified design strength of 4000 psi.

In the same letter the applicant provided uniformity test results dated July 9, 1976. The results for weight per cubic foot, air content (volume percent of concrete), coarse aggregate content, and the average compressive strength at seven days, are all within the maximum permissible differences, thus complying with project requirements. During a telephone conference with the applicant's representatives on February 11, 1987, the Task Group was informed that the preparation of information related to the coefficient of variation of concrete mix requires more time and will be provided in the future.

Followup Actions and Recommendations:

It is not entirely possible to assess the quality of concrete used at the plant with complete confidence on the basis of the few test samples reviewed by the Task Group. This matter should be investigated in more detail by reviewing test results available at the site.

The statistical data regarding uniformity of concrete, when submitted by the applicant, should be reviewed by the Task Group in order to gain additional confidence with regard to the quality of concrete used at the plant. Examination of this information, together with the evidence that periodic compression tests of concrete samples met the job specifications, will constitute sufficient evidence that the mixer blades did or did not affect quality of concrete, and, therefore, safety of the plant.

References:

1. Letter from E. H. Johnson (NRC) to W. G. Counsil (TUEC), dated February 3, 1986, Appendix B, NRC Inspection Report 50-445/85-07 and 50-446/85-05.

2. ACI Standard "Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete," ACI-304-73.

3. Brown & Root Procedure No. 35-1195-CCP-10, "Concrete Batch Plant Operations," Rev. 5, dated December 4, 1978.

4. Letter from W. G. Counsil (TUEC) to U.S. Nuclear Regulatory Commission, dated February 10, 1987.

## 3.7 Issues 1-9 and 11-15 of Inspection Report 85-14/11:

Issues 1 through 9 and 11 through 15 all relate to storage, transmission and handling of quality records. These issues were dealt with collectively because the consequences of worst-case events have a common result.

1. FSAR 17.1.17 does not describe TU Electric records system.

2. QA manual does not address ANSI- N 45.2.9 requirements/ commitments.

3. TU Electric failed to have/use procedures to control shipment of original design records for piping to Stone & Webster, NY.

4. Original design records shipped in cardboard boxes to Stone & Webster.

5. No backup copy of records made for records shipped to Stone & Webster.

6. Failure to control and account for design records transferred from site to Stone & Webster, NY. TU Electric stated design record shipped without making backup copy because cost too much. Also stated it was company policy to proceed at own risk.

7. Site records of Chicago Bridge & Iron shipped to Houston, Texas in cardboard boxes.

8. No backup copy of records made for records shipped t Chicago Bridge & Iron.

9. TU Electric failed to inventory records sent to Chicago Bridge & Iron.

11. Failure to preclude rain from entering QA interim records vault over several years time.

12. Failure to preclude food and coffee pot from QA interim record vaults (fire hazard).

13. Failure to install fire suppression systems, drains, and a sloped floor at permanent vault.

14. Plant records stand in folders or binders in open face cabinets at records center.

15. Failure to provide temporary or permanent storage facility for records entered into the permanent records center then co-mingled with in-process documents in paper flow group.

Description:

1. The Final Safety Analysis Report (FSAR), Chapter 17.1.17, does not adequately describe the quality records system for Comanche Peak construction activities. The FSAR does not reflect the level of detail prescribed by the American National Standards Institute (ANSI) N 45.2.9. (1,2)

2. The Texas Utilities Electric Company (TU Electric) Corporate Quality Assurance (QA) Program Manual and QA Plan do not address ANSI N 45.2.9 in all aspects. (1,2)

3. TU Electric shipped original quality records to Stone & Webster, Inc. without an implementing a procedure to control the process. (1,2)

4. Original design records were shipped to Stone & Webster, Inc., in cardboard boxes. These boxes did not afford the proper protection specified for records of this type. (1)

5. The records shipped to Stone & Webster did not have duplicate copies retained onsite. (1)

6. The records sent to Stone & Webster were not inventoried nor documented such that accountability could be maintained.

7. Original quality records were shipped to Chicago Bridge & Iron (CB&I) in Houston, Texas in cardboard boxes. This is similar to item 4 above. (1)

8. The records shipped to CB&I did not have duplicate copies retained onsite. This is similar to item 5 above. (1)

9. The records shipped to CB&I were not inventoried nor documented such that accountability could be maintained. This is similar to item 6 above. (1)

11. There was evidence of rain leaking through a ventilation duct in the interim record vault creating a hazard for stored records. (1,2)

12. Food stuffs and a coffee pot were found in the interim record vault creating a potential rodent and fire hazard. (1,2)

13. There is no installed fire suppression system in the permanent records storage vault. In addition, there is a 2-inch fire hose available to fight fires but there are no floor drains and the floor is not sloped to provide runoff. This condition creates a potential for added water damage to the records should the fire hose be used. (1,2)

14. The TU Electric records center, one of two permanent records storage areas, has records stored in folders and binders in openfaced cabinets. The records center is protected by a deluge water sprinkler system for fire. In the event the sprinkler is actuated, the records will be damaged or destroyed. (1,2)

15. Permanent plant records were withdrawn from record vaults to facilitate work flow. These records were stored temporarily in the Paper Flow Group trailers in fire-rated and nonfire-rated cabinets. (1,2)

Discussion:

Title 10 CFR 50.71 and Appendix B require that records be generated and maintained to confirm that certain activities have been satisfactorily accomplished. These records pertain to plant activities, such as quality control and quality assurance functions, to plant design processes, such as design reports or design verifications, and to the results of equipment performance tests, such as preoperational and plant startup testing. These records provide proof that the facility was designed and constructed as described in the FSAR and document the basis for licensing the plant. Some records are designated as "lifetime or permanent" records while others will be retired after construction is completed and plant operation begins. The permanent records are maintained to support safe plant operation, plant modifications, accident analysis, and in-service examination.

The foregoing issues all deal with the proper storage and retention of these kinds of records. The applicant has committed to store records in accordance with ANSI N 45.2.9, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants." The jurisdiction of this standard begins when a file is completed and is designated a quality record.

Safety Significance:

The "worst case" result of any of the above issues is that important design, construction and operations records would be lost, destroyed or missing. Most records can be recreated by reinspection of accessible equipment, reconstruction of design documents, or replacement of the record or equipment. The condition of inaccessible equipment can also be determined by engineering analysis or more rigorous testing.

The near-term impact of missing records is a delay in completing document packages for system turnover to establish operational readiness for plant preoperational and startup testing. The long-term effects could be that operating plant activities such as plant modifications, in-service examinations and testing, event analysis, and systems analysis would be hampered by incomplete information.

Followup Actions and Recommendation:

Issues 4 through 6 and 12 were not transmitted as violations to the applicant, and do not require a formal response for corrective actions. Item 3, transmission of records without a procedure, was responded to by the applicant.

The response indicated that a procedure was developed, records were retrieved and copied, and personnel were trained to the procedure. On February 13, 1987, the applicants, in responding to issues 1, 2, 7, 8, 9, 13, 14, and 15, provided the following additional information:

Issue 1 - The FSAR does not describe the TU Electric records system. The applicants have concluded that their procedures do not adequately describe the final TU Electric review and the conversion of documents to records. TU Electric is drafting new procedures to correct this deficiency.

Issue 2 - QA manual does not address ANSI N 45.2.9 requirements. The applicant believes the QA manuals and procedures are adequate.

Issues 7, 8 and 9 - Demonstrate that the CB&I records controls were implemented. The applicant essentially believes the CB&I procedure was adequate and correctly implemented. No further action is warranted.

Issue 11 - Failure to preclude rain from entering QA interim records vault. The water leakage was corrected.

Issue 13 - Failure to install a fire suppression system, drains, and a sloped floor in the permanent records vault. TU Electric's position is that a fire suppression system is neither needed nor required. Their survey of the floor has demonstrated that it is sloped.

Issue 14 - Plant records stand in folders or binders in open faced cabinets at records center. TU Electric is installing an alarm to detect operation of the sprinkler system and alert personnel to the potential for flooding. They consider records storage in open faced cabinets is acceptable.

Issue 15 - Permanent plant records were withdrawn from record vaults to facilitate work flow. These records were stored temporarily in the Paper Flow Group trailers in fire-rated and nonfire-rated cabinets. The applicants maintain that documents in the Paper Flow Group are in-process and not subject to ANSI N 45.2.9. However, TU Electric has placed the documents in fire-rated cabinets.

The Task Group recommends that the NRC staff review the revised TU Electric NEO procedures upon their completion and verify the implementation. Pending Task Group 2 recommendations relating to applicable enforcement, the NRC staff should verify corrective actions proposed by the applicants.

References:

1. OIA Report of Investigation 86-10, Allegation of Misconduct by Region IV Management with Respect to the Comanche Peak Steam Electric Station.

2. NRC Inspection Report 50-445/85-14 and 50-446/85-11.

## 3.8 Issue 10 of Inspection Reports 85-14/11:

TU Electric audited CBI Houston and in the scope of the report stated it included Criterion XVI, QA records but did not document the audit of records Violation Criterion XVIII.

### Description:

Texas Utilities Electric Company (TUEC) documented an audit of the quality assurance (QA) program of the Chicago Bridge & Iron, Inc. (CB&I), the contractor responsible for constructing the Unit No. 2 containment building liner, and failed to document the details of the record's portion of the audit, although this portion was listed in the scope of the audit.(1)

### Discussion:

TUGCO is required by 10 CFR 50, Appendix A, Criterion I, and Appendix B, to establish a quality assurance program. One requirement of the quality assurance program is that TUEC be responsible for auditing the work of its contractors, including the contractor's QA records. The major purposes of such an audit are to provide management with information regarding the effectiveness of the production process, to identify non-conforming or deficient items, and to monitor the QA program itself. Based on the audit findings, management evaluates such conditions and initiates corrective actions where warranted.

### Safety Significance:

There is no direct equipment safety significance for not performing an audit. The worst case outcome is a programmatic breakdown of the audit process.

The issue is that the applicant failed to document one element of an audit, quality assurance records. The central question then becomes, was this audit performed?

The audit record establishes that audits were performed of other areas listed in the scope of the audit.(2) If the audit of QA records was not performed, then at a minimum, the result could be an inadequate record's program. For a detailed discussion of the safety significance relating to audits, see Section 3.3 of this report, and for the record's program aspects, see Section 3.7. If the audit of records was performed but not documented, then the effect on construction of the liner is of no consequence.

Recommendations and Followup Actions:

See the Recommendations and Followup Actions in Section 3.3 of this report.

References:

1. Texas Utilities Generating Company letter to Chicago Bridge & Iron, Inc., dated May 7, 1985, QXX-2381; subject: TUGCO QA Audit Report, QA Audit File: TCB-6
2. Draft Report 50-445/85-14 and 50-446/58-11, undated, CPRRG-17.

### 3.9 Issue 16 from Inspection Reports 85-14/11:

Weld rod not identified at main distribution station (i.e., labels taken off or lost on E-309 electrodes at the main distribution station).

Description:

Based on inspections conducted, NRC inspectors concluded that there was a failure to maintain the material classification, size, and heat lot number markings on several containers of Sandvick welding rods located in the main storage areas and that this failure was contrary to paragraph 3.2.1 of Brown & Root (B&R) Procedure CP-CPM-6.9, Appendix B,(1) and was also a violation of 10 CFR 50, Appendix B, Criterion V.

Discussion:

Weld rods used in safety-related applications are normally stored in the main distribution station and then transferred in marked shipping cartons to three rod issue stations: rod houses 2, 3, and 4. Loose labels observed in the main distribution station were also observed in rod house 4 during a routine NRC inspection. Even so, the inspector noted that the material was identifiable because of marking on the storage bin and shipping cartons. However, loss of identify was possible when the rods were removed from storage. Rods are issued for use one lot at a time. They are taken from their cartons only when needed to replenish the stock in the drying oven or when issued for use.

The remainder of a lot not issued is put in the oven. (A heated stationary or energized portable oven was used in the field to keep coated electrodes dry.) B&R Procedure CP-CPM-6.9, Appendix B(1) has requirements for controlling the identification of weld filler material that is removed from its original container.

Safety Significance:

The coated weld rods in question (E309 coated electrodes) are usually used for welding dissimilar metals (e.g., carbon steel to stainless steel). E308 coated electrodes, on the other hand, are employed in welding stainless steel to stainless steel. Both electrodes are similar in color, thickness of coating, and length.(2) They differ from other common coated electrodes on site, such as E7018 or E8018 (which are used to join carbon steel to carbon steel) in color, thickness of coating, and length. Welders would recognize the difference immediately upon striking an arc if E7018 or E8018 was substituted for E309 because the arc characteristics are so different. However, welders may have difficulty in distinguishing the unmarked E308 electrodes from E309 electrodes.

In the worst case scenario, a welder could use an E7018 or E8018 carbon steel electrode in a stainless steel system. Assuming that the weld deposit was adequate and that the weld passed the required visual, nondestructive, and hydrostatic testing, the weld could be put into service in the reactor coolant system. The carbon steel would corrode very quickly because of the aggressive corrosive effects of borated waters(3). The corrosion would ultimately lead to a weld failure and to a loss of coolant accident if the weld was in the reactor coolant pressure boundary. If an E308 electrode is substituted for an E309 electrode, the finished weld practically has the same quality as that made from an E309 electrode; therefore, such a substitution would have no safety significance.

## Followup Actions and Recommendations:

The Task Group recommends that the NRC staff determine if individual electrodes are identified individually according to AWS or MIL-E-22200 recommendations with a "Type Mark," such as "309". If each electrode is uniquely identified, the probability for misuse becomes very low, especially when coupled with the arc characteristics and physical appearance of these electrodes. If the electrodes are not uniquely identified, then the applicant should undertake a statistically valid sampling of stainless steel welds to determine whether and to what degree misidentification and misapplication occurred.

## References:

1. Brown & Root, Inc., Procedure CP-CPM-6.9B, Revision 2, "Weld Filler Material Control," dated September 21, 1984.

2. NUREG-0797, Supplement 10, Safety Evaluation Report, pg. N-79, April 1985.

3. Czajkowski, C. J., NUREG/CR-2827, "Boric Acid Corrosion of Ferritic Reactor Components," July 1982.

### 3.10 Issues 1 through 5 from Inspection Reports 85-16/13:

1. Failure to develop/implement procedure to demonstrate 50.55(e) deficiencies corrected.

2. Failure to revise implementing procedures containing 50.55(e) reporting. TU Electric failed to revise implementing procedures before corporate NEO Procedure CS-1 was implemented resulting in conflict with five other procedures.

3. Failure to maintain retrievable 50.55(e) files (i.e., could not produce record in almost a month).

4. Failure to report to NRC actual corrective action taken on 50.55(e)s.

5. TU Electric's 50.55(e) files not auditable.

Description:

1. TU Electric's procedure to process Construction Deficiency Reports (CDR) failed to require file information which would give evidence of issue closure.

2. TU Electric failed to revise subtier implementing procedures before corporate NEO Procedure CS-1 was issued, resulting in conflict with five other procedures.

3. TU Electric failed to maintain CDR files that were retrievable.

4. TU Electric failed to report to NRC the corrective actions actually taken and changes to commitments.

5. TU Electric CDR files were not auditable with respect to corrective actions.

Discussion:

The reporting requirements under 10 CFR 50.55(e), Construction Deficiency Reports (CDR), were instituted to provide NRC with prompt notification of significant construction deficiencies. They are to give NRC timely information on which to base an evaluation of the potential safety consequences of the deficiency and determine if further regulatory action is required(1). CDRs are normally identified by the applicants' quality assurance program through nonconformance reports, design deficiency reports, vendor 10 CFR 21 reports, or other similar systems.

Any breakdown in the CDR reporting and tracking system would affect the notification, evaluation and final closure of construction deficiencies as it relates to NRC. NRC requires that selected construction deficiency reports be closed through inspections(2). If detailed tracking files are not maintained, closure becomes more difficult; however, the primary corrective action tracking document for the identified deficiency would be the original quality assurance report.

The procedure identified in item 1 does not require certain information to be retained in the applicant's tracking file which would permit the inspector to readily determine if the item had been properly closed. This makes the file unauditable for the inspector unless there are cross references to the corrective actions programs.

The failure to revise subtier procedures, item 2, results in nonuniformity in the processing of CDRs, but does not necessarily affect reporting to the NRC. This also affects the relationship between the NRC and the applicant, and internal processing within the applicant's organization.

The failure to maintain CDR files that were retrievable, item 3, stems from the inspector's inability to cross reference between the CDR files and corrective action program files. This is similar to items 1 and 5 in that files that are not retrievable are also not auditable.

The failure to report corrective actions actually taken and any changes to commitments, item 4, directly affects NRC/applicant communications. Not receiving this information affects NRC's ability to perform a meaningful evaluation and reach any decisions to take further regulatory action.

Safety Significance:

The issues at TU Electric that were identified by the inspector all relate to reporting procedures between the NRC and the applicant. Based on the fact that CPRs were written and filed, there is no indication in the inspector's report that the identification mechanism for CDRs was deficient. Therefore, the Task Group assumes that the sources of input to the process were functioning satisfactorily. Under these circumstances deficient equipment or controlling systems were apparently being corrected through other established mechanisms, such as the nonconformance corrective action process prescribed by 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions. Thus, if these reporting mechanisms are functioning, there is no safety significance relative to the plant equipment.

Followup Actions and Recommendations:

The applicant's response of February 13, 1987, states that CDR activities are currently controlled by TU Electric Procedure NEO CS-1, "Evaluation of and Reporting of Items/Events under 10 CFR 21 and 10 CFR 50.55(e)." TU Electric has established a licensing commitment resolution process to track the timely completion of commitments made to the NRC. Also, there is a task force chartered to identify, validate and assure positive closure of CDRs.

It was noted by the Task Group that TU Electric Procedure NEO CS-1, "Evaluation of and Reporting of Items/Events Under 10 CFR 21 and 10 CFR 50.55(e)," does not specify that all items reported under the procedure should be first recorded in the established corrective action systems. The procedure states that inputs can be received from any source. Where the source is other than an established quality tracking system, it is possible that a reported deficiency would not be properly processed under a formal corrective action system.

The Task Group recommends that an audit be performed by the NRC staff of the current CDR program to verify its adequacy and implementation.

References:

1. 10 CFR 50.55(e) Statement of Considerations, 37 FR 6459.

2. Inspection and Enforcement Manual, Inspection Procedure 92700, 8/13/84

3. Inspection and Enforcement Manual, Interpretations 10 CFR 50.55(e), 4/1/80

## 3.11 Issue 6 from Inspection Report 85-16/13:

TU Electric never responded to all aspects of IEB 79-14.

### Description

During several NRC inspections of Comanche Peak Steam Electric Station (CPSES) from November 1 through 30, 1985, inspectors focused on the applicant's response to IE Bulletin (IEB) 79-14, "Seismic Analysis for As-Built Safety-Related Piping System," was reviewed. Specifically, the issues in IEB 79-14 was evaluated and closed for nonconformances because paragraph 4 was not satisfied by TU Electric in 1983. The NRC inspectors indicated that the closure was premature since Stone & Webster was then analyzing Unit 1 seismic analysis against as-built drawings, an issue directly related to this Bulletin. In addition, the inspectors found that the same analysis had been completed for Unit 2.

TU Electric stated in response that the IEB 79-14 file would be reopened and a supplemental report would be submitted upon completion of the ongoing engineering work relevant to this Bulletin. The status of IEB 79-14 status is still considered open by NRC (445/8516-0-03, 446/8513-0-03).(1)

### Discussion:

This issue cannot be resolved by comparing IEB 79-14 with the applicant's formal responses (2,3,4) and with recent activities by Stone & Webster Engineering Company.(5) Actions taken by the applicant and the staff to address this issue must be considered in the context of Commission practice with regard to evaluation of all responses by utilities to the Bulletin. The discussions in OIA Report 86-10 do not indicate whether the applicant's "As-Built Verification Program"(4) and Comanche Peak Project Procedures(5) change the status of this issue.

NRC issued IEB 79-14, "Seismic Analyses For As-Built Safety-Related Piping Systems" on July 2, 1979 and Revision 1 on July 18, 1979. Supplements 1 and 2 to the Bulletin were issued on August 15 and September 7, 1979, respectively. Since TU Electric responses were not based on the original Bulletin, the pertinent document is Revision 1.

IEB 79-14 requested that all power reactor facility licensees and construction permit holders verify, unless previously verified to an equivalent degree within the last 12 months, that their seismic analysis applies to the actual configuration of safety-related piping systems. The Bulletin was issued because inspections of safety-related piping systems addressed in IEB 79-02, 79-04, and 79-07, and show-cause orders for four nuclear power facilities, revealed some as-built deviations from design documents used for input to seismic design analyses. These deviations were significant enough to have an adverse effect on the validity of the seismic analyses. The Bulletin required licensees of operating facilities to perform walk-down inspections of safety-related piping systems, make comparisons to seismic analysis input, identify nonconformances and evaluate their effect on system operability, and either make hardware modifications or reanalyze the as-built configuration to validate their seismic analyses. Holders of construction permits were required to inspect and report on safety-related piping systems for compatibility of seismic analyses with as-built configurations.

The Bulletin specified that all inspections were to be completed and the results reported to NRC within 120 days. IE provided specific written guidance to the Regions for immediately implementing IE inspections and for reviewing applicant followup actions and written responses to IEB 79-14.(6) Although these instructions focused on operating facilities, the Bulletin was equally applicable to plants under construction. As the implementation of this bulletin progressed, a substantial number of questions were raised by IE inspectors, licensees, and industry representatives.(7,8) In response, NRC issued two supplements to the Bulletin.

Licensee reports were initially evaluated using a task group approach involving personnel from IE headquarters, NRR, and the Regional offices.(6,9) A contractor was subsequently retained to assist the staff in reviewing responses to the Bulletin. As of June 1985, the bulletin was closed out for only 48 of the 124 facilities under review.(10)

The applicant's initial response (October 25, 1979) stated that they were finalizing a procedure to address the Bulletin and requested a waiver from the 120 day reporting requirement because construction was not yet sufficiently complete to support a system inspection program.(2) The applicant subsequently provided (December 3, 1982) a formal response to IEB 79-14 that defined the scope of the "As-Built Verification Program."(3) The safety class, size, and type of piping systems included in the program were defined. The applicant specifically stated that documentation in the form of piping and support construction drawings and support location isometrics would be field verified by site QA personnel (emphasis added).(3) The applicant implemented this program throughout the construction of Units 1 and 2 in response to the Bulletin.(4) The applicant also commissioned Stone & Webster Engineering Corporation to perform a stress requalification of code piping and piping supports.

Safety Significance:

The purpose of IEB 79-14 was to ensure that the seismic analysis input information agrees with actual construction details at this facility. The Bulletin requires that specific nonconformances be resolved by either making changes to the system, such that it conforms to the design, or by correcting the seismic analysis to demonstrate the acceptability of the as-built configuration criteria.

The issue from OIA Report 86-10 is that the applicant never responded to all aspects of IEB 79-14.

In the worst-case scenario CPSES receives its operating license with as-built pipe support systems that are not consistent with the plant's seismic analysis. Under these circumstances, the safety significance of the worst-case scenario could be enormous in that the plant might not be able to achieve a safe shutdown condition following a seismic event. Based on the limited information available, the Task Group reached the following conclusions:

(1) When IEB 79-14 was issued on July 2, 1979, the design and construction of CPSES was not sufficiently advanced to make a detailed response to the Bulletin meaningful.

(2) The program outlined by letter TXX-3597(3) may not have addressed all aspects of the bulletin. However, it did explain the applicant's intentions. The applicant implemented and followed the "As-Built Piping Verification Program" (Instruction CP-EI-4.5-1) with the expressed purpose of addressing IEB 79-14.(4)

(3) Reopening the issue of IEB 79-14 in Inspection Report 50-445/85-16; 50-446/85-13 was appropriate considering the major analytical activities and plant modifications at CPSES.

(4) The applicant has on-going programs to address the concerns identified in IEB 79-14.(5)

(5) The safety significance of the worst case could be enormous.

(6) At this time, there is no safety-significant issue since IEB 79-14 will be closed before licensing.

Followup Actions and Recommendations:

Based on the historical record of IEB 79-14, the actual regulatory requirements and acceptance criteria are not obvious.(11) The Task Group discussed this issue with a member of the Comanche Peak Technical Review Team, who concluded that TU Electric has essentially fulfilled their commitment to IEB 79-14 and considers the issue closed. However, based on a Task Group review of the records, the Task Group finds that this issue has yet to be fully resolved and thus that this issue is still open. The Task Group recommends that the NRC staff provide the applicant with a clear and concise written evaluation of the applicants' actions taken to date and specify additional actions required to close this issue. This written position should be consistent with the Commission policy that was applied to other licensees and applicants that successfully completed the requirements of the Bulletin.

References:

1. Letter to Texas Utilities Generating Company from E. H. Johnson (NRC), dated April 4, 1986.

2. Letter to NRC Region IV from R. J. Gary (TUEC), dated October 25, 1979 (TXX-3062).

3. Letter to NRC Region IV from R. J. Gary (TUEC), dated December 3, 1982 (TXX-3597).

4. Letter to NRC Region IV from W. G. Counsil (TUEC), dated April 3, 1986 (TXX-4729).

5. Letter to NRR from W. G. Counsil (by J. W. Beck, TUEC), dated September 19, 1986 (TXX-5034).

6. Memorandum to Regional offices from J. E. Bryan (NRC), dated August 6, 1979, Subject: TI 2515/29 - Inspection Requirements For IE Bulletin 79-14.

7. Memorandum to Regional office from N. C. Moseley (NRC), dated September 6, 1979, Subject: Supplement 2 to IE Bulletin 79-14.

8. Memorandum to N. C. Moseley (NRC) to D. G. Eisenhut (NRC), dated August 28, 1979, Subject: Recommendation Concerning Inspection and Implementation Requirements of IE Bulletin 79-14.

9. Memorandum to D. G. Eisenhut (NRC) to E. L. Jordan (NRC), dated September 11, 1979, Subject: Evaluation of Responses to IE Bulletin 79-14.

10. Letter to R. L. Baer (NRC) from R. A. Lofy (Parameter, Inc.), dated June 28, 1985, related to NRC Contract 05-82-249.

11. Memorandum to Regional offices from E. L. Jordan, dated February 7, 1986, Subject: IE Bulletin 79-02 and 79-14 Status Reports.

## 3.12 Issues 7, 9 and 10 of Inspection Reports 85-16/13:

7. TU Electric's IEB record files were incomplete (1982 and 1985).

9. Deficiency in TU Electric's procedures to handle IEBs. They do not describe how construction management personnel handle IEB requiring action especially hardware repair, replacement and modification.

10. No focal point at TU Electric to track IEB actions.

Description:

7. TU Electric Inspection and Enforcement Bulletin (IEB) record files were incomplete. The bulletin files were decentralized rather than located in the QA records center. Further, the engineering evaluations were retained by the individual engineers rather than being centrally filed.

9. There were deficiencies in the procedure to process IEBs. (Note: The record is not clear as to what the inspector meant by deficient. The Task Group assumed that the deficiencies resulted from the lack of a central coordination position for IEBs and the perceived file deficiencies.)

10. No central coordination function at TU Electric to track IEB actions.

Discussion:

The NRC issues Inspection and Enforcement Bulletins (IEB) to operating reactor facilities and those under construction to transmit information or to request action or information regarding matters of safety, safeguards or environmental significance.(1) The safety information transmitted may identify generic equipment or design deficiencies. Licensees and applicants are expected to determine the applicability of the Bulletin to their plant and initiate appropriate corrective actions. IEBs usually require licensees and applicants to respond with such information as applicability to the plant, equipment affected, operability status of systems, corrective actions initiated, and schedules for completion.

The NRC inspection program requires that licensee and applicant implementing programs for IEB-requested actions be inspected to assure that appropriate actions have been taken.(2) The program requires that all documents in the response to NRC be reviewed and a determination made that the response was proper. It also requires that onsite sampling inspections be made to verify that equipment changes were made as reported.

Based on the range of issues they encompass, IEBs can be a mechanism for identifying nonconforming conditions. Consequently, licensees and applicants must make provisions for receiving, evaluating, initiating corrective actions, verifying corrective actions, and reporting results to their management and to NRC.

The Companche Peak applicant has a procedure for processing correspondence, including IEB s, from NRC; Nuclear Operations Engineering Manual, Licensing, Procedure No. NOE-205. This procedure assigns responsibilities and discusses the processing and the retention of records.

Safety Significance:

Assuming the worst case to be the breakdown in the processing of IEBs, the impact on safety would be significant. If the NRC issued a Bulletin that affected the facility equipment or operating procedures and it was not incorporated into the plant, the facility could operate with an inherent undetected/uncorrected defect.

A Bulletin, by definition, is only issued when safety concerns have been identified and when NRC believes there is a threat to the public safety. Bulletins can affect operating and construction functions and a thorough and comprehensive control program must be established to evaluate, track and resolve Bulletin issues.

Followup Actions and Recommendations:

Applicant Procedure NOE-205, dated October 7, 1985, generally describes the process for controlling Bulletins through their receipt,logging in, review, plan development, response, and closeout. Paragraph 4.2.15 of the procedure specifies that documents which provide source information for the Bulletin response be included in the document package. If the procedure is implemented, it should provide an adequate system for processing Bulletins. However, the procedure does not clearly discuss the relationship between the operating organization and the corrective actions systems. For TU Electric internal auditors, TU Electric management, and NRC to be able to determine if appropriate actions have been taken, TU Electric must clarify the relationship between the corrective action system and the operating organization.

The applicant submitted a response to unresolved Item 445/85-16-U-02 and 446/85-13-U-02 in the correspondence to NRC dated February 13, 1987. In this letter, the applicant stated that Bulletins requiring licensee responses are processed by the Operations Support Section under procedure ECE-AD-18. All other correspondence is routed to the Industrial Operating Experience Record Coordinator for evaluation per procedures NOS-103.

The applicant has committed to perform a procedure and record review to ascertain the adequacy of the IEB program. The review should assure that all Bulletins were received and processed and that corrective actions were initiated through the above established programs. It is the Task Group's recommendation that the NRC staff evaluate the effectiveness of this program through an audit.

References:

1. NRC Inspection and Enforcement Manual, Chapter 0702, NRC Office of Inspection and Enforcement Bulletins and Information Notices, 2/21/86.

2. NRC Inspection and Enforcement Manual, Chapter 92703, IE Bulletin, Confirmatory Action Letter and Generic Letter Followup, 2/14/86

## 3.13 Issue 11 from Inspection Report 85-16/13:

Description:

All reporting requirements of Inspection and Enforcement Bulletin 79-14 were not met.

Discussion:

Inspection and Enforcement Bulletin (IEB) 79-14 was issued because NRC had identifed several operating facilities in which the as-built configuration of the piping systems did not agree with the seismic analysis design inputs. (See Sections 3.11 and 3.12 for more details.) The Bulletin requested several actions of licensees and applicants, including that they identify and report to NRC on the status of nonconforming conditions noted during inspections that would cause safety systems to be inoperable under certain seismic events.

Safety Significance:

This issue has no safety significance based on the fact that facilities under construction were not intended to report nonconformances to the NRC in compliance with IEB 79-14. The Task Group discussed the Bulletin and its scope with a cognizant Inspection and Enforcement staff member who confirmed that the intent of reporting nonconformances was to assure that operating plants took appropriate corrective actions.

Followup Actions and Recommendations:

This item was initially discussed as an unresolved item in the draft inspection report and subsequently dropped from the final inspection report. The issue was never formally transmitted to the applicant; thus, no corrective actions would have been initiated. No further actions are warranted concerning this matter.

### 3.14 Issue 8 from Inspection Reports 85-16/13:

NAMCO switches IEB 79-28 were not properly identified on installation travelers.

#### Description:

During routine NRC inspections of Comanche Peak Steam Electric Station (CPSES) in November 1985, inspectors identified inconsistencies between certain NAMCO switch model numbers identified on the installation instructions (travelers) and the model numbers on the installed hardware in the plant.(1) Inspectors noted deficiencies in documentation and a delay associated with the filing of documents in the master data base and QA vault. In addition, OIA raised a concern related to the adequacy of the hardware when documenting the results of its review of this issue.(2)

#### Discussion:

On December 7, 1979, NRC issued IEB 79-28, "Possible Malfunction of NAMCO Model EA 180 Limit Switches at Elevated Temperatures." The purpose of the Bulletin was to alert the industry to a deficiency in certain manufactured lots of NAMCO EA 180 limit switches. NAMCO CONTROLS, the switch manufacturer, determined that the switch top cover gasket emitted a resin vapor at temperatures above 175°F. This vapor could condense into deposits on the normally open contacts, possibly causing a switch malfunction.

By letter dated March 24, 1980, from R. J. Gary (TUGCO) to Karl V. Seyfrit (NRC), the applicant for CPSES (hereafter the applicant) responded to IEB 79-28. In its response, the applicant stated that 14 EA-180 NAMCO switches required replacement of the top cover gasket, that none of the switches had been put in service or exposed to ambient temperatures of more than 175°F, and that the replacement gaskets were being ordered from NAMCO and would be installed by June 30, 1980. Subsequently, by letter dated July 30, 1981, from R. J. Gary to Karl V. Seyfrit, the applicant revised its earlier response to IEB 79-28, stating that due to difficulty resolving environmental qualification concerns, all NAMCO switches within the scope of IEB 79-28 would be replaced prior to plant operation.

During routine NRC inspections of CPSES in November 1985, the inspectors focused on the applicant's actions in response to IEB 79-28, which apparently evolved into a broader review of other NAMCO switches not addressed by the Bulletin. During the November 1985 inspection, inspectors identified inconsistencies between certain NAMCO switch model numbers identified on the installation instructions (travelers) and the model numbers on the installed hardware. Specifically, two NAMCO switches on residual heat removal (RHR) system valves 1-HCV-606 and 1-FCV-618 were identified on travelers EE 82-1415-5801 and EE 83-0373-5801 as EA 180-32302 and EA 170-31302, respectively. The switches actually installed were stamped EA 180-31302 and EA 180-31302. It should be noted that the Inspection

Reports (85-13/16) identified these switches as being within the scope of IEB 79-28,(1) however, this is inconsist with the applicant's statement that these particular switches were replaced as part of the environmental qualification upgrade and not a part of the fourteen switch replacement effort initiated in response to the Bulletin.(5)

Safety Significance:

In assessing the safety significance of the November 1985 findings, Task Group 3 assumed that the wrong switches were installed and that this situation existed without correction or recognition. Even based on this assumption, the Task Group determined that this issue is not safety significant. In performing its worst-case assessment, the Task Group reviewed the CPSES Final Safety Analysis Report (FSAR), information provided by the switch manufacturer (NAMCO CONTROLS), and TU Electric, and discussed the issue with their representatives.

As shown on Figure 5.4-6 of the CPSES FSAR, (3) both valves 1-HCV-606 and 1-FCV-618 are located in the A train of the Unit 1 RHR system outside the containment building. Valve 1-FCV-618 is an air diaphragm-operated butterfly valve utilized to control bypass flow around the A RHR heat exchanger. Valve 1-HVC-606 is an air diaphragm-operated butterfly valve utilized to control discharge flow from the A RHR heat exchanger. The NAMCO switches identified by the inspectors provide valve position indication readouts in the emergency response facility and control room.

In assessing the safety significance of an error involving the installation of these switches, Task Group 3 evaluated the consequences of installation of switch model EA 170-31302 on RHR valve 1-FCV-618 in lieu of switch model number EA 180-31302. The Task Group learned that the switches differ in their construction and capability to function in a hostile environment.(4) Although both EA 180 and EA 170 series switches are designed to function in a high radiation field, switches in the EA 180 series have been designed to function in a containment building following a design bases accident under such harsh environmental conditions as elevated temperatures and pressures. In the worst case scenario, a switch designed to function in a mild environment could have been installed on RHR valve 1-FCV-618. Since this valve is located outside the containment building in a mild environment, the Task Group concludes that such an error would not be safety significant.

In assessing the safety significance of an error involving the installation of switch EA 180-32302 on valve 1-HVC-606, rather than switch EA 180-31302, the Task Group determined that the worst case would result in a malfunctioning valve position indicator. The Task Group learned that the switches differ in their internal return spring configuration and direction of rotation. The EA 180-31302 switches are set up at the factory to operate in the clockwise direction and the EA 180-32302 switches are set up at the factory to operate in the counter clockwise direction.(4) Installing a switch with the wrong direction of rotation would result in a false indication of valve position in the control room. Although a false indication could be misleading to the plant operators, other information is available to the control room operators about system status, including RHR discharge to the reactor coolant system cold leg temperature recorder (TR-612), and RHR discharge to the reactor coolant system cold leg flow indicator (FI-618), both of which can be read at the control board. Thus, the Task Group concludes that such an error is not safety significant.(3) Moreover, an installation error that results in a malfunctioning valve position indicator should be detected and corrected as part of the pre-operational and startup test program. As a minimum, prior to declaring the RHR system valves operable, testing prescribed by Section XI of the ASME Code must be completed. One such test involves stroking each valve and verifying the operability of the valve position indicator.

Followup Actions and Recommendations:

As discussed in the OIA Report, according to the inspectors, the applicant provided NRC with two new travelers to demonstrate the acceptability of the as-built configuration. In addition, Region IV management provided OIA copies of an applicant nonconformance report and four travelers that demonstrated the acceptability of the as-built configurations.(2)

In response to an NRC request for information,(6 & 7) the applicant clarified the evolutionary nature of the NAMCO switch replacement program.(5) During the replacement program for environmentally unqualified switches, the applicant replaced the existing type and model with equivalent type and model limit switches. The last change to the switch on valve 1-HCV-618 was controlled by traveler EE 83-1851-5801 and completed in August, 1983. A review of this traveler by the applicant indicates that switch EA 180-32302 was replaced with EA 180-31302. The last change to the switch on valve 1-HCV-606 was controlled by traveler EE 83-0459-5801 and completed in June, 1983. A review of this traveler by the applicant indicates that switch EA 180-32302 was replaced with EA 180-31302. Based on a review of the appropriate records and testing performed, the applicant has confirmed that the as-installed configuration satisfies the design requirements.

With regard to the discrepancy between the travelers and the installed hardware identified by the inspector(1), the Task Group has concluded that it results from a delay in updating the official file. Changes made in mid-1983 had not yet been entered into the permanent plant record in November 1985. The inspector initially obtained the NAMCO switch traveler information from the permanent plant record vault.(8) None of the information available to the Task Group clarifies why outdated records were being stored nor what assurances exist that the records would ever have been updated. If the switch records were in the permanent plant record vault, this indicates that installation had been completed and possibly that a portion of the system had been turned over to the plant.

Once plant records are completed, any changes to the equipment must be identified in the records. It is the Task Group's recommendation that the closeout of the unresolved item should assure that the applicant understands the root cause of the traveler mismatch with the equipment. They should also assure that the records system is structured such that equipment records affected by in-plant equipment changes are identified and tracked to completion. The records system should assure that the changes are captured, that duplicate packages do not result, and that permanent records, such as equipment qualification files, are updated.

In reviewing the travelers associated with the NAMCO switch replacement for valves 1-FCV-618 and 1-HCV-606, the Task Group found at least three travelers that directed the installation of one or more switches designed to operate in a particular direction of rotation in locations where the opposite direction of rotation was required. Travelers EE 82-1415-5801, EE 83-0447-7802, and EE 83-0373-5801 directed the installation of switches with one direction of rotation that were later replaced by a switch with the opposite direction of rotation via travelers EE 83-0459-5801 and EE 83-1851-5801. Because traveler EE 82-1415-5801 directed the removal of switches with the wrong direction of rotation, there is evidence that other travelers that proceeded EE 82-1415-5801 were in error. Although the differences were identified and corrected, the circumstances raised a concern relative to the development of travelers for the entire NAMCO switch replacement effort. It is the Task Group's recommendation that the closeout of this unresolved item attempt to determine the cause of this apparent breakdown and its implications. The applicant has confirmed that all switches affected by IEB 79-28 were identified and replaced based on programs in place for procurement and documentation review, on installation instructions utilized, on QA inspection records, and on walkdowns. As a part of the current Corrective Action Program (CAP) in the equipment qualification area, a complete field verification of Class 1E equipment and documentation verification will be performed by the applicant. The purpose of this program is to identify and resolve all as-built environmental qualification discrepancies. It is the Task Group's recommendation that the closeout of this unresolved item include monitoring of the applicant's ongoing programs (including the CAP) to confirm that the correct switches have been installed.

References:

1. Letter to Texas Utilities Generating Company from E. H. Johnson (NRC) dated April 4, 1986 transmitting Inspection Reports 85-13 and 85-16.

2. OIA Report 86-10, Attachment MM, Technical Review of Issues Contained in Comanche Peak Inspection Reports.

3. Comanche Peak Steam Electric Station Final Safety Analysis Report, Units 1 and 2. Docket Numbers 50-445 and 50-446.

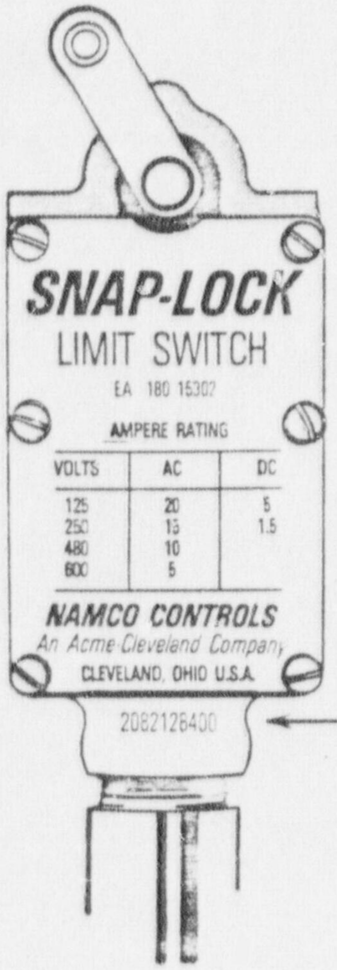
4. NAMCO Limit Switches and Quick Connectors for Nuclear Environment, Series EA 180-302/602-Rev. and EA-170-302/602 Rev., 3M/5-85.

5. Letter from W. G. Counsil (TU Electric) to the NRC dated February 9, 1987.

6. Letter from V. S. Noonan (NRC) to W. G. Counsil (TU Electric) dated February 1, 1987.

7. Letter from V. S. Noonan (NRC) to W. G. Counsil (TU Electric) dated February 6, 1987.

8. Memorandum from R. D. Martin (NRC) to J. G. Davis (NRC) dated January 20, 1987.



Switch Model Number

Conduit Number: Week/Year of Manufacture (1st four digits)

NAMCO Limit Switch

3.15 Issue 12 from Inspectsion Report 85-16/13:

Insufficient evidence of successful testing of BISCO fire seals - filing of false report by BISCO - Validity of BISCO seal questioned.

Description:

In Comanche Peak Inspection Report 85-16/13(1), NRC inspectors identified an unresolved item with regard to the qualification of eight Brand Industrial Services, Inc. (BISCO) fire-rated electrical penetration seals (Nos. AB-790-174-1022A, EC-854-150A-1018A and B, EC-854-151A-2003A and -2004A, EC-854-151B-2025A and -2026A and TB-803-010A-1008A). Specifically, the inspectors could not verify from available documentation that the actual seal installation met the design requirements specified in the Comanche Peak Final Safety Analysis Report (FSAR)(2). The inspectors found that documentation pertaining to the testing of the BISCO seals in accordance with the testing standards of ASTM E-119 and IEEE 634, as specified in the FSAR, could not support the 3-hour rating certification statement provided by the applicant.

Discussion:

FSAR Section 9.5 states that a minimum 3-hour fire-resistant barrier shall be provided that separates each cable spreading room from other plant areas and that separates redundant safety divisions. This FSAR criteria is in accordance with the requirements of Appendix R to 10 CFR Part 50 and the guidelines of Standard Review Plan, Section 9.5.1. The inspection report indicates that the electrical penetration seals in question were intended to provide a full 3-hour fire rated barrier between redundant trains of safe shutdown system cabling in the identified plant areas. If so rated, the seals would prevent a fire in one room from spreading through the wall penetration to the adjacent room containing redundant safe shutdown cabling for a mimimum of three hours and thereby preclude a loss of safe shutdown functions. A typical electrical conduit fire-resistant penetration seal configuration is shown in the accompanying figure.

Safety Significance:

In the worst case, an unqualified penetration seal could not adequately prevent the spread of fire between adjacent areas containing redundant safe shutdown cabling. Such a fire could damage redundant cabling designed to achieve post-fire safe shutdown. However, by letter dated February 9, 1987(4), the applicant indicated that of the eight penetration seals of concern, six (EC-854-150A-1018A and B, EC-854-151A-2003A and -2004A, and EC-854-151-2025A and 2026A) are installed with 1-hour fire-rated barriers, and the remaining two (TB-803-010A-1008A and AB-790-174-1022A) are installed with 3-hour fire-rated boundaries. For the six 1-hour seals, no further concern exists since available documentation confirmed seal qualification in excess of a 1-hour rating. Should a fire breach the penetrations of the remaining two boundaries, redundant trains of safe shutdown equipment/cabling would not be affected since in one case (TB-803-010A-1008A), no safe shutdown equipment is in the fire area itself, and in the other (AB-790-174-1022A), the adjoining area contains only Unit 2 safe shutdown equipment. The impact on Unit 2 has not yet been assessed by the licensee, but the Unit 2 safe shutdown capability would be ensured by proper protection. Thus, even if the seals in question were to fail, the post-fire safe shutdown capability of the plant would not be lost. It should be further noted that installed automatic fire suppression systems in the areas of concern would provide additional protection against the spread of fire beyond that provided by the seals, and that fire detectors would alert plant operators to take any necessary manual actions to fight the fire or initiate plant shutdown. Applicant calculations for the areas of concern also indicate combustible loadings below the rating of the barriers.

Followup Actions and Recommendations:

In the February 9, 1987 letter, the applicant indicated that compliance with licensing requirements for the seals will be achieved by reinstalling penetration seal AB-790-174-1022A in accordance with the appropriate BISCO procedure prior to fuel load. No further action will be taken on penetration seal TB-803-010A-1008A because no safe shutdown equipment would be affected by a fire in the area. For the six remaining seals in the 1-hour fire-rated barriers, no further action is planned since the documentation concern extended only to the 3-hour fire rated seals. Finally, the applicant indicated that a verification walkdown of all as-built penetration seals will be performed. The Task Group concurs with the applicant's corrective actions and concludes that compliance with fire protection requirements will be achieved.

Since BISCO seals are used in a number of nuclear power plants, the above inspection finding was treated as a generic concern by Region IV and was referred to the Office of Inspection and Enforcement, Vendor Branch, for assistance in resolution. The Vendor Branch reviewed the test documentation concern and inspected at other plants and the BISCO facility. The results of their review will be issued shortly in the form of an inspection report. The report will provide general clarifying information on fire barrier penetration seal testing and will provide the necessary backup information to assist inspectors when they review seal certification documentation in the future. Region IV should utilize this report when reinspecting the BISCO seals and certification documentation at Comanche Peak to close out the unresolved inspection item and to confirm final compliance with fire protection requirements.

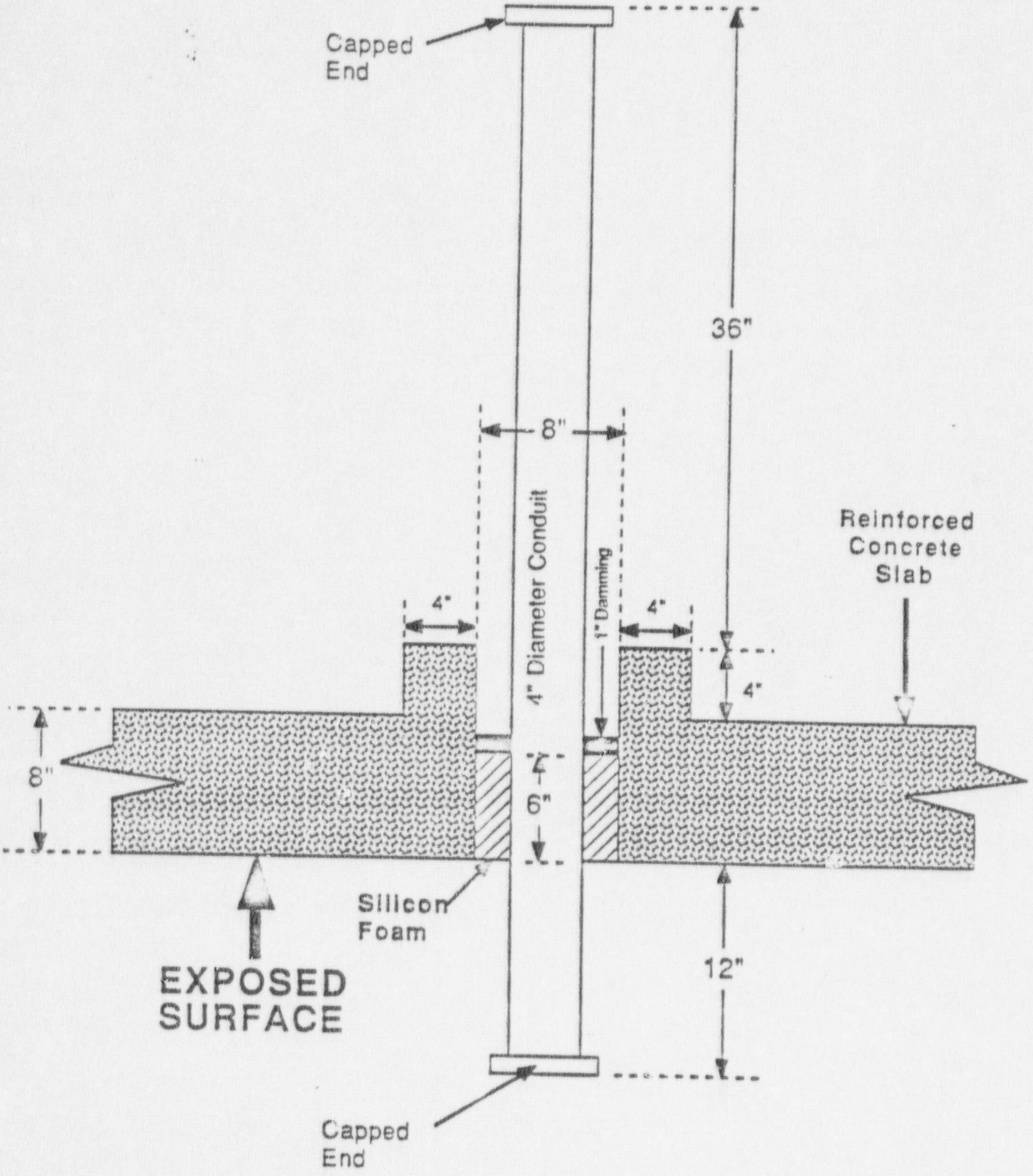
rences:

.. Letter to Texas Utilities Generating Company from E.H. Johnson, Director, Division of Reactor Safety and Projects, NRC Region IV, dated April 4, 1986.

2. Comanche Peak Steam Electric Station, Units 1 and 2 Final Safety Analysis Report, Docket Nos. 50-445 and 50-446.

3. Diagram of Typical Electrical Penetration Fire Resistant Seal.

4. Letter to the NRC from W.G. Counsil, Texas Utilities Electric Company, dated February 9, 1987.



TYPICAL ELECTRIC CONDUIT FIRE-RESISTANT PENETRATION SEAL
(As supplied by several vendors)

85-14/85-11

ATCH 20