TU ELECTRIC

Log # TXX-91136
File # 10200
Ref. # 10CFR50.73
10CFR50.73(a)(2)(i)

April 29, 1991

William J. Cahill, Jr.
Executive Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS
LICENSEE EVENT REPORT 91-003-001 (SUPPLEMENTAL REPORT)

Ref: 1) TU Electric Letter (TXX-91053) from William J. Cahill, Jr. to the NRC dated February 25, 1991.
2) TU Electric Letter (TXX-91103) from William J. Cahill, Jr. to the NRC dated March 28, 1991.

Gentlemen:

By reference 1, TU Electric submitted Licensee Event Report (LER) 91-003-00 for Comanche Peak Steam Electric Station Unit 1, "Less than Adequate Procedure Review Leading to the Failure to Fully Satisfy ASME Section XI Testing Requirements." This transmittal noted a supplemental report (LER 91-003-01) would be forthcoming by March 29, 1991. By reference 2, TU Electric indicated this supplement would be provided by April 29, 1991. Please find attached the subject report.

Sincerely,

William J. Cahill, Jr.

By: W G Guldemond
W. G. Guldemond
Manager, Site Licensing

JAA/bm

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)
Mr. J. W. Clifford, NRR

IE22 1/1

Enclosure to TXX-91136

NRC FORM 366 — U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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Title (4) LESS THAN ADEQUATE PROCEDURE REVIEW LEADING TO THE FAILURE TO FULLY SATISFY ASME SECTION XI TESTING REQUIREMENTS

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Numbers
0 1	2 4	9 1	9 1	- 0 0 3 -	0 1	0 4	2 9	9 1	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

Operating Mode (9): 3
Power Level (10): 0 0 0

This report is submitted pursuant to the requirements of 10 CFR §: (Check one or more of the following) (11)

- [ ] 20.402(b)
- [ ] 20.405(a)(1)(i)
- [ ] 20.405(a)(1)(ii)
- [ ] 20.405(a)(1)(iii)
- [ ] 20.405(a)(1)(iv)
- [ ] 20.405(a)(1)(v)
- [ ] 20.405(c)
- [ ] 50.36(c)(1)
- [ ] 50.36(c)(2)
- [X] 50.73(a)(2)(i)
- [ ] 50.73(a)(2)(ii)
- [ ] 50.73(a)(2)(iii)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(b)
- [ ] 73.71(c)
- [ ] Other (Specify in Abstract below and in Text, NRC Form 366A)

Licensee Contact For This LER (12)

Name	Telephone Number
T.A. HOPE SUPERVISOR, COMPLIANCE	Area Code 8 1 7 — 8 9 7 - 6 3 7 0

Complete One Line For Each Component Failure Described in This Report (13)

Cause	System	Component	Manufacturer	Reportable To NPRDS		Cause	System	Component	Manufacturer	Reportable To NPRDS

Supplemental Report Expected (14)

[X] Yes (If yes, complete Expected Submission Date) [ ] No

Expected Submission Date (15): Month | Day | Year

Abstract (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On January 24, 1991, an engineer was performing a review of a Residual Heat Removal system operability test procedure being developed to support licensing of Unit 2. The reviewer noted that the ASME Section XI test requirements specified in the Inservice Testing Plan were not satisfied by the procedure. The reviewer observed a similar discrepancy in the Unit 1 test procedure. Inadequate implementation of the Section XI test requirements constitutes a failure to meet the operability requirements for a Technical Specification Limiting Condition for Operation. The cause of the event was determined to be inadequate technical review during or following document revision. Corrective actions include a request for relief from the testing requirement and continued evaluation.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

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Text (If more space is required, use additional NRC Form 366A's) (17)

I. DESCRIPTION OF THE REPORTABLE EVENT

A. EVENT CLASSIFICATION

Any operation or condition prohibited by the plant's Technical Specifications.

B. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On January 24, 1991, (discovery date), at approximately 1400 CST, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 3, Hot Standby, in preparation for a plant startup.

On January 25, 1991 (reportability date), the plant was in Mode 1, Power Operation, with the reactor at 35 percent of rated thermal power.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems, or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On January 24, 1991, an engineer (contractor, non-licensed) in the technical support organization was performing a review of the test procedure being developed for testing of the Unit 2 Residual Heat Removal (RHR) system (EIIS:(BP)). The reviewer noted that the full flow test required by the Inservice Testing (IST) Plan for the RHR pump discharge check valves (EIIS:(V)(BP)) is not accomplished by performance of the test procedure. Rather, a partial flow test is specified. Because of the similarities between the procedures being developed to support licensing of Unit 2 and the existing Unit 1 procedures, the reviewer compared the Unit 1 RHR system operability test to the IST Plan requirements. A similar discrepancy was discovered.

CPSES Technical Specification 4.0.5(a) requires inservice testing of ASME Code Class 1, 2 and 3 valves in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50.55a(g) except where specific written relief has been granted by the Nuclear Regulatory Commission (NRC). Those inservice testing requirements are accomplished as specified in the *CPSES Unit 1 Inservice Testing Program Plan for Pumps and Valves*. The RHR pump discharge check valves are required by the present IST Plan to be full stroke tested at cold shutdown. Contrary to the plan requirement, the cold shutdown full stroke tests of the RHR pump discharge check valves were not performed within the allowed surveillance interval. Per Surveillance Requirement 4.0.3, upon discovery of this discrepancy, the valves must be successfully tested within 24 hours, or the system (RHR trains A&B) declared inoperable, placing CPSES Unit 1 into the applicable action statement of Specification 3.0.3.

On January 26, 1991, TU Electric contacted the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, to request a Temporary Waiver of Compliance from cold shutdown full stroke testing of CPSES Unit 1 RHR pump discharge check valves. The NRC orally granted a Temporary Waiver of Compliance from Technical Specification requirements until applicable written relief from ASME testing requirements could be granted, or until February 9, 1991, whichever occurred first. On January 26, 1991, TU Electric submitted a formal written request for a Temporary Waiver of Compliance, and on January 28, 1991, the NRC formally granted that request. On January 28, 1991, TU Electric submitted a request for relief from ASME Section XI cold shutdown full stroke testing of RHR pump discharge check valves. On February 8, 1991, the NRC granted an interim approval of the relief request until the September, 1991, refueling outage to allow evaluation of alternative testing flow paths.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR PROCEDURAL ERROR

While performing a review of the Unit 2 RHR system operability test, the reviewing engineer identified a discrepancy between the IST Plan requirements and the test procedure. Further review of the Unit 1 RHR test revealed a similar error. This discovery prompted investigation which led to the conclusion that the RHR pump discharge check valves had not been tested in accordance with ASME Section XI requirements.

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Text (If more space is required, use additional NRC Form 366A's) (17)

II. COMPONENT OR SYSTEM FAILURES

A. FAILED COMPONENT INFORMATION

Not applicable - there were no component failures associated with this event.

B. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

Not applicable - there were no component failures associated with this event.

C. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable - there were no component failures associated with this event.

D. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - there were no component failures associated with this event.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Not applicable - there were no safety system responses associated with this event.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable - there were no safety systems rendered inoperable due to a failure.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The objective of the inservice testing of ASME Code Class 1, 2, and 3 valves is to demonstrate that there is a high probability that the applicable check valves will operate satisfactorily if and when they are called upon to perform their safety related functions. Previous testing includes full flow testing in January, 1990, with the reactor vessel head removed and in conjunction with safety injection flow balancing and full stroke testing of other check valves in the system. Partial stroke testing utilizing flow through the 3/4 inch test header has been performed quarterly, and cold shutdown testing of the RHR pump discharge check valves performed as recently as November, 1990, has documented flows through these check valves of between 3500 and 4000 gpm. These testing activities, along with the low probability of a check valve failure concurrent with an event necessitating the use of this system provides sufficient basis to conclude the event does not result in a threat to the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

ROOT CAUSE NO.1

A root cause of the event has been determined to be less than adequate review resulting in the failure to initiate a required change to a lower tier document upon revision to an upper tier document. Investigation of the events prior to discovery revealed that the Master Surveillance Test List (MSTL) had been revised three times between September 1989 and January 1990. The MSTL is a comprehensive listing of Technical Specification surveillance requirements and the corresponding test procedures satisfying those requirements. Each revision affected these check valves, both the test frequency and the procedure assigned to accomplish the test. In Revision 15 on January 20, 1990, the full stroke test reverted back to a cold shutdown frequency (from Refueling) consistent with the IST Plan, and was assigned to the RHR system operability test procedure. When this MSTL revision occurred, the current test procedure, Revision 1, was not changed to implement the revised MSTL requirement.

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ROOT CAUSE NO. 2

Another root cause has been determined to be less than adequate performance of technical review during the procedure revision process. Subsequent to issuance of the current revision of the MSTL, Revision No. 2 to the Unit 1 RHR system operability test procedure was issued on October 25, 1990. The preparation and review process failed to detect the lack of a full stroke test in the procedure as assigned by the MSTL.

V. CORRECTIVE ACTIONS

IMMEDIATE

Relief from ASME Section XI cold shutdown full stroke testing of RHR pump discharge check valves was requested from and granted by the NRC. A change was made to the IST Plan to reflect this relief. This change brought the existing RHR system operability test into agreement with the IST Plan.

ACTION TO PREVENT RECURRENCE

Reviews are being performed on a sample of MSTL entries to ensure (1) consistency with source documents, (2) adequacy of implementing procedures, (3) adequacy of scheduling activities and frequency, and (4) proper overlap when multiple procedures cover a single activity.

The administrative procedure controlling the MSTL will be revised to require a detailed review by departments affected by changes to the MSTL to ensure that those changes are reflected in the associated test procedures.

The IST procedure development and review process will be strengthened to clearly define responsibilities of reviewers and to establish a philosophy for the level of review required for revised procedures. Training will be developed for procedure writers and technical reviewers describing responsibilities and management expectations.

As discussed in LER 91-007-00, an overall review of the IST program will be performed to identify and correct any additional problems.

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VI. PREVIOUS SIMILAR EVENTS

CPSES Licensee Event Reports (LERs) 90-005-00, 90-010-00, 90-015-00, 90-024-00, 90-026-00, 90-034-00, 90-040-00, and 90-044-00 describe reportable events resulting from failure to perform surveillance activities required by plant Technical Specifications.

VII. ADDITIONAL INFORMATION

During incident evaluation, a review was performed to identify any instances of misapplication of ASME Section XI relief requests. Although the review concluded that the existing relief requests in the CPSES IST Plan are appropriate, several relief requests have been selected for further review. These relief requests all propose a refueling outage frequency for component testing, whereas the Code specifies a quarterly or cold shutdown frequency. The referenced testing falls into two categories: (1) measurement of pump flowrate as required by subsection IWP, and (2) close exercising of check valves as required by subsection IWV. The subject relief request numbers are:

P.10	12.1
5.1	13.1
5.2	15.19
5.4	16.1
6.1	16.2
10.1	

The testing to which the CPSES IST Plan is currently committed is being satisfactorily performed and there exists no operability concern. The impracticality of performing these tests at the Code specified frequency is being reassessed, and any change in test frequency as a result of this reassessment will be addresed prior to startup following the first refueling outage.