50-445

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 17, 1996

Mr. C. Lance Terry
Group Vice President, Nuclear
Texas Utilities Electric Company
Energy Plaza
1601 Bryan Street, 12th Floor
Dallas, TX 75201-3411

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF REACTOR TRIP AT PLANT COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1 (TAC NO. M72403)

Dear Mr. Terry:

Enclosed for your review and comment is a copy of the preliminary Accident Sequence Precursor (ASP) analysis of an operational event which occurred at Comanche Peak Steam Electric Station, Unit 1 on June 11, 1995 (Enclosure 1), and was reported in Licensee Event Report (LER) No. 445/95-003,-004. This analysis was prepared by our contractor at the Oak Ridge National Laboratory (ORAL). The results of this preliminary analysis indicate that this reactor trip may be a precursor for 1995. In assessing operational events, an effort was made to make the ASP models as realistic as possible regarding the specific features and response of a given plant to various accident sequence initiators. We realize that licensees may have additional systems and emergency procedures, or other features at their plants that might affect the analysis. Therefore, we are providing you an opportunity to review and comment on the technical adequacy of the preliminary ASP analysis, including the depiction of plant equipment and equipment capabilities. Upon receipt and evaluation of your comments, we will revise the conditional core damage probability calculations where necessary to consider the specific information you have provided. The object of the review process is to provide as realistic an analysis of the significance of the event as possible.

In order for us to incorporate your comments, perform any required reanalysis, and prepare the final report of our analysis of this event in a timely manner, you are requested to complete your review and to provide any comments within 30 days of receipt of this letter. We have streamlined the ASP Program with the objective of significantly improving the time after an event in which the final precursor analysis of the event is made publicly available. As soon as our final analysis of the event has been completed, we will provide for your information the final precursor analysis of the event and the resolution of your comments. In previous years, licensees have had to wait until publication of the Annual Precursor Report (in some cases, up to 23 months after an event) for the final precursor analysis of an event and the resolution of their comments.

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We have also enclosed several items to facilitate your review. Enclosure 2 contains specific guidance for performing the requested review, identifies the criteria which we will apply to determine whether any credit should be given specific actions in recovering from the event, and describes the specific information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 445/95-003,-004, which documented the event.

Please contact me at (301) 415-2972 if you have any questions regarding this request. This request is covered by the existing ORB clearance number (3150- 0104) for NRC staff follow-up review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

Original signed by
Phillip M. Ray, Acting Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-445

Enclosures: As stated

cc w/encls: See next page

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OFC	PD4-1	PD4-1
NAME	PNoonan	PRay:sp
DATE	6/14/96	6/17/96
COPY	YES/NO	YES/NO

PMRay signed 6/17/96

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in the analysis for the use of licensee-identified additional equipment orspecific actions in recovering from the event, and describes the specific information that you should provide to support such a claim. Enclosure 3 is a copy of LER No. 445/95-003,-004, which documented the event.

Please contact me at (301) 415-2972 if you have any questions regarding this request. This request is covered by the existing ORB clearance number (3150-0104) for NRC staff follow-up review of events documented in LERs. Your response to this request is voluntary and does not constitute a licensing requirement.

Sincerely,

Phillip M. Ray

Phillip M. Ray, Acting Project Manager
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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-445

Enclosures: As stated

cc w/encls: See next page

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Comanche Peak, Units 1 and 2

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LER No. 445/95-003, -004

Event Description: Reactor trip, Auxiliary Feedwater (AFW) pump trip, second AFW pump unavailable

Date of Event: June 11, 1995

Plant: Comanche Peak 1

Event Summary

While at 100% power on June 11, 1995, Comanche Peak 1 experienced a control power supply failure resulting in both main feedwater pumps (MFPs) tripping, and operators subsequently initiating an anticipatory reactor trip. Flow from one of two motor driven auxiliary feedwater pumps (MDAFWP) was unavailable and the turbine driven auxiliary feedwater pump (TDAFWP) started on low-low steam generator level but tripped on overspeed. The conditional core damage probability estimated for this event is 3.2×10^{-5}.

Event Description

While at 100% power on June 11, 1995, Comanche Peak 1 experienced a control power supply failure resulting in both MFPs tripping, and operators subsequently initiating an anticipatory reactor trip. Slave relay testing was under way when a non-safety related inverter transferred from its normal inverter ac power supply to its alternate power supply. The alternate ac power supply was deenergized as required by the test procedure at the time, so associated loads were deenergized. The specific cause of the transfer is not certain but it may have been due to an electrical transient in a static transfer switch control circuit. Loss of the power supply caused a spurious "MFP oil pressure low" signal when auxiliary relays in pump supervisory instrumentation deenergized and actuated. This caused the condensate pumps to trip; loss of the condensate pumps caused both MFPs to trip. Operators then initiated a manual reactor trip in anticipation of an automatic one.

ENCLOSURE 1

The MFP trips caused an auto-actuation of the MDAFWPs. MDAFWP 1-02 (Train B) started and supplied water to steam generators (SGs) 3 and 4 (Fig 1). MDAFWP 1-01 (Train A) was aligned to its test header at the time in and was not immediately available to supply water to the SGs. The TDAFWP started on low-low SG level but tripped on overspeed, caused by a failure of the governor valve to control turbine speed. The governor valve stem was found to be corroded and binding against the valve packing. Operators realigned MDAFWP 1-01 from the test header to its normal configuration and the pump supplied cooling to SGs 1 and 2 within about 8 minutes.

Additional Event-Related Information

The licensee event report (LER) provided additional information concerning the thermal-hydraulic effects of having only one AFW pump available immediately after a plant trip. Plant safety analyses assume for a "Loss of Normal Feedwater Flow" transient that the TDAFWP or both MDAFWPs provide a flow rate of at least 860 gpm to the SGs. During this transient, only one MDAFWP was initially available, providing a reduced flow rate to the SGs. However, the LER indicated that the reduced flow rate was adequate to remove plant decay heat from the SGs because of the early manual trip of the reactor and because initial water levels in the SGs were greater than the assumption used in the FSAR analysis. Because sufficient heat removal capability was available, the thermal expansion of the reactor coolant system inventory did not fill the pressurizer completely.

Modeling Assumptions

This event was modeled as a reactor trip with the TDAFWP failed and flow from MDAFWP 1-01 initially unavailable. Basic event AFW-TDP-FC-1C was set to "TRUE" (failed). (Table 1 provides a description of the basic event names.) Because MDAFWP 1-01 was recovered 8 minutes into the event, the probability for nonrecovery of the overall AFW system (AFW-XHE-NOREC = 0.26) was believed to be conservative (i.e., too high). Therefore, the overall probability of recovering the AFW system was increased by taking credit for recovering the MDAFWP 1-01 within 8 min. Recovery of MDAFWP 1-01, itself, was incorporated into the models using the methodology described in Reference 4. This methodology suggests a nonrecovery probability of 0.1 when "[f]ailure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial stress." Consequently, the nonrecovery probability for MDAFWP 1-01 was incorporated by setting the probability for event AFW-MDP-FC-1A equal to 0.1. Because AFW is required without delay during ATWS sequences, a new event, AFW-MDP-FC-AA, with a nonrecovery probability of 1.0 was substituted for AFW-MDP-FC-1A in the ATWS model.

As main feedwater apparently could not have been recovered without correcting the inverter problem, restarting the condensate system, and restoring a feedpump to service, the feedwater system was assumed not to be recoverable (MFW-XHE-NOREC = "TRUE").

The failures in this event increase the potential significance of failure to trip/ATWS sequences. In order to more accurately model potential reactor trip failures, the reactor trip model was modified by "ANDing" events RPS-REC (recoverable RPS failures) and RPS-XHE-XM-TRIP (operator nonrecovery probability). This was then "ORed" with event RPS-NONREC (nonrecoverable RPS failures). These modifications do not significantly alter the calculated conditional core damage probability for this event, but they provide a more realistic modeling of the event.

The event trees for Comanche Peak assume that conditions requiring a reactor trip will first result in an automatic reactor trip demand and, if the automatic trip fails, a manual reactor trip demand. During this event, once operators recognized that a loss of main feedwater flow had occurred, they initiated a manual reactor trip. Because of the operators quick response, consideration was given to the potential impacts of the early reactor trip on ATWS sequences. The Comanche Peak FSAR indicates that 1 - 1½ min may elapse between a loss of feedwater and an automatic reactor trip. The additional 1 min of response time available to operators during postulated ATWS sequences in this event was not believed to materially affect the event sequences or probabilities and no related model changes were indicated.

Analysis Results

The conditional core damage probability (CCDP) estimated for this event is 3.2×10^{-5}. The dominant core damage sequence (sequence 21 on Fig. 2 and sequence 8 on Fig. 3) involves:

- failure to successfully trip,
- successful control of reactor pressure, and
- failure of AFW.

The second highest core damage sequence (sequence 20 on Fig. 2) involves:

- a successful reactor trip,
- failure of AFW
- failure of MFW, and
- failure of feed-and-bleed cooling.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences are shown in Table 2. Table 3 lists the sequence logic associated with the sequences listed in Table 2. Table 4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table 5.

Acronyms

ac	Alternating Current
AFW	Auxiliary Feedwater
ATWS	Anticpated Transient Without Scram
CCDP	Conditional Core Damage Probability
CST	Condensate Storage Tank
FSAR	Final Safety Analysis Report
LER	Licensee Event Report
MDAFWP	Motor-Driven Auxiliary Feedwater Pump
MFP	Main Feedwater
SG	Steam Generator
SWS	Service Water System
TDAFWP	Turbine-Driven Auxiliary Feedwater Pump

References

1. LER 445/95-003, Rev. 1, "Loss of Both Condensate and Both Feedwater Pumps Due to Failure of Non-Safety Related Inverter Resulted in a Manual Reactor Trip," August 14, 1995.

2. LER 445/95-004, Rev. 1, "Allowed Outage Time was Exceeded on Turbine Driven Auxiliary Feedwater Pump Which Tripped on Overspeed," September 8, 1995.

3. Texas Utilities Generating Company, *Comanche Peak Steam Electric Station Final Safety Analysis Report*.

4. M. B. Sattison, et. al., *Methods Improvements Incorporated into the SAPHIRE ASP Models*, NUREG/CP-0140, Vol. 1, Proceedings of the U.S. Nuclear Regulatory Commission, Twenty-Second Water Reactor Safety Information Meeting, April 1995.

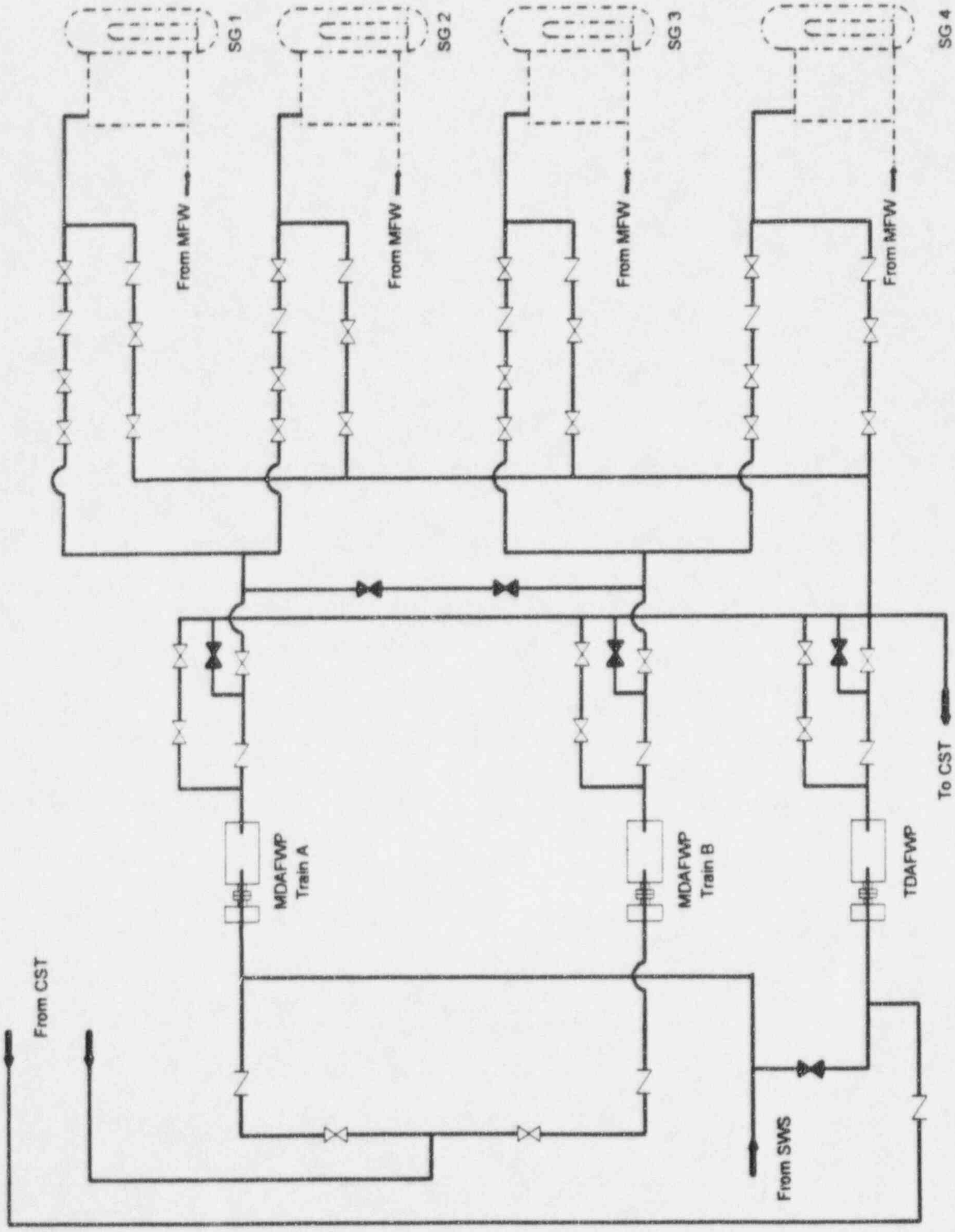

Fig. 1 Auxiliary feedwater system for Comanche Peak

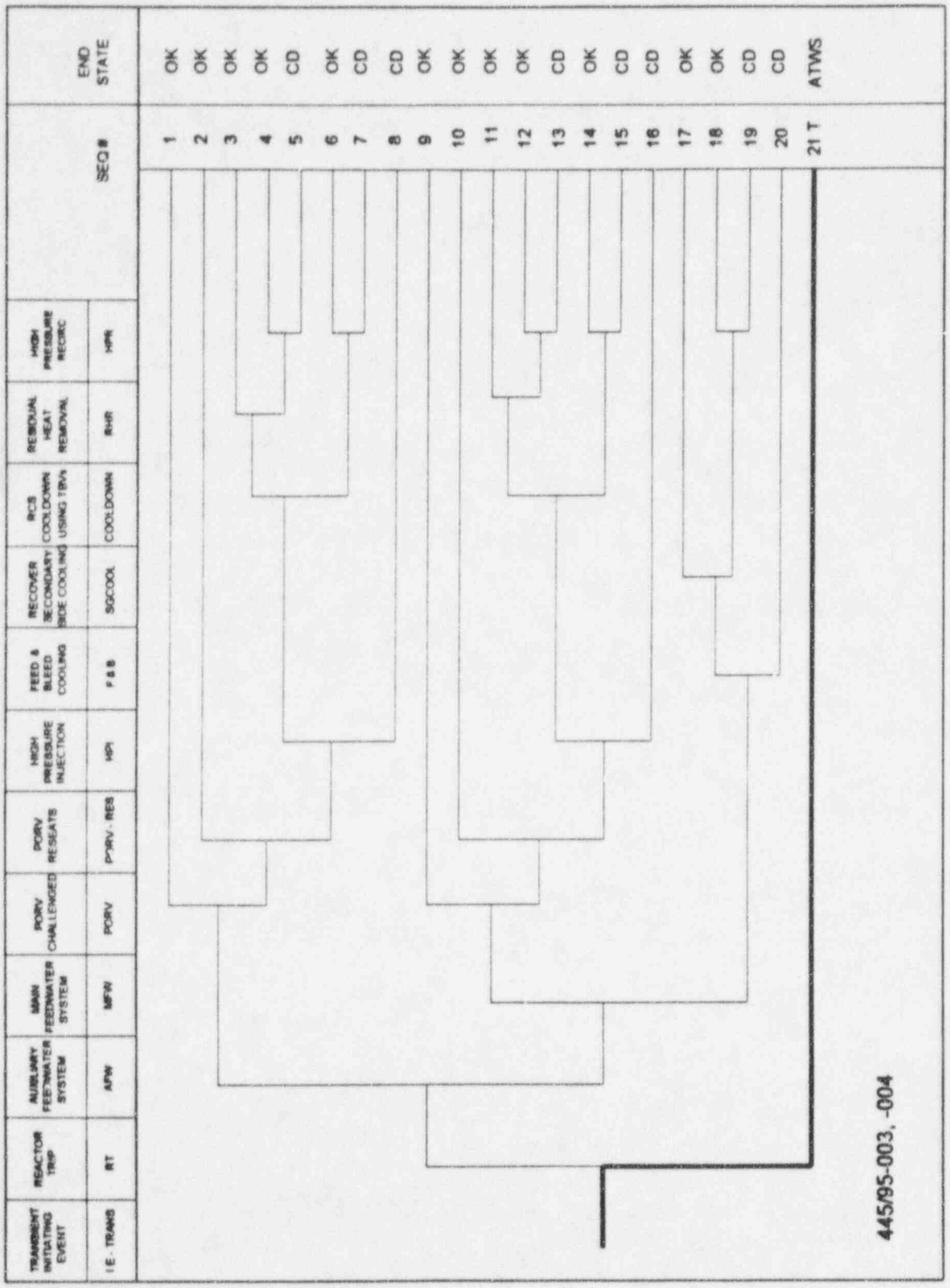

Fig. 2 Dominant core damage sequences for LERs 445/95-003, -004.

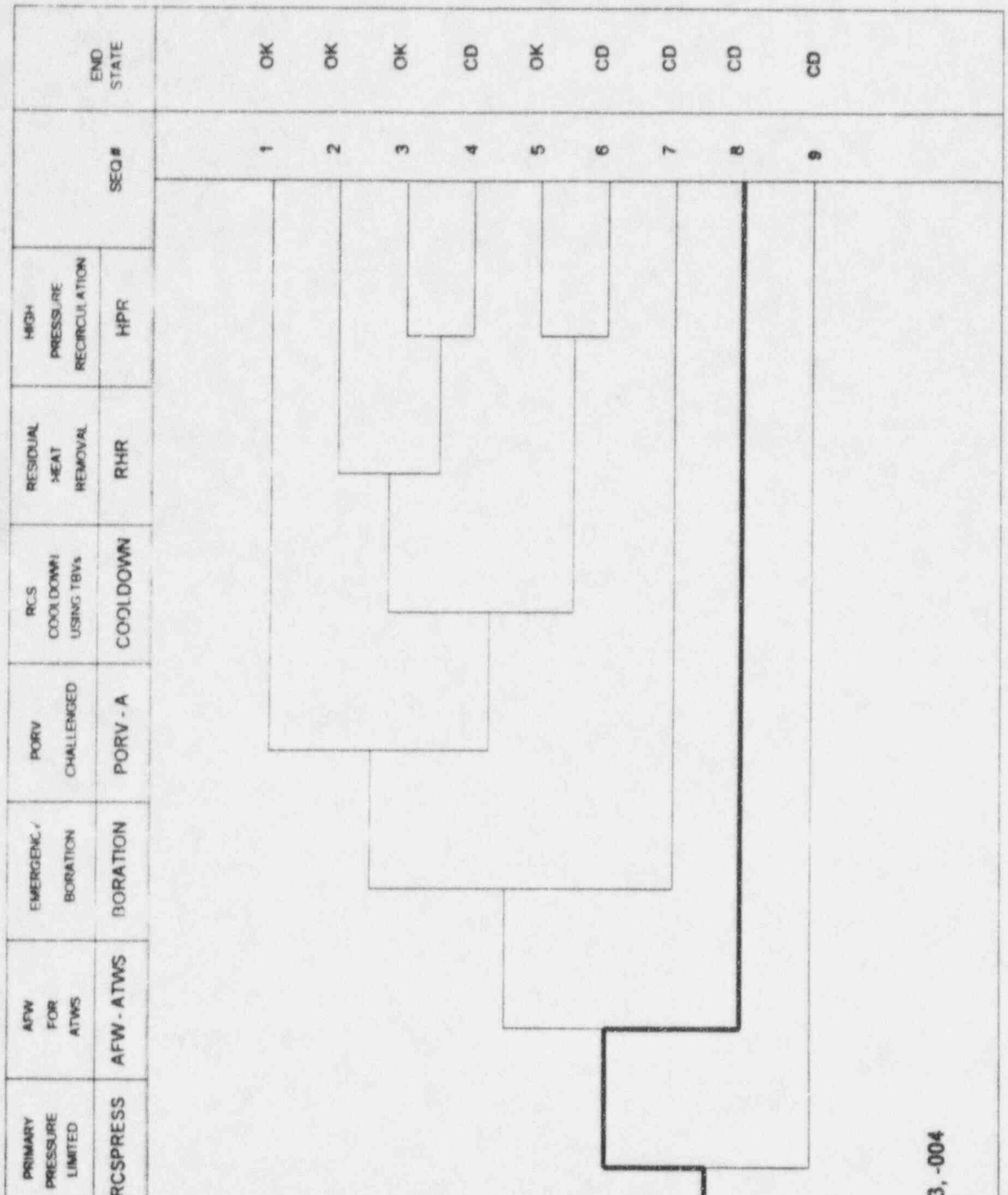

Fig. 3 Anticipated transient without scram (ATWS) event tree for Comanche Peak.

Table 1. Definitions and probabilities for selected basic events for LER 445/95-003, -004

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss of Offsite Power Initiating Event	8.5 E-006	0.0 E+000	IGNORE	No
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.6 E-006	0.0 E+000	IGNORE	No
IE-SLOCA	Small Loss of Coolant Accident Initiating Event	1.0 E-006	0.0 E+000	IGNORE	No
IE-TRANS	Transient Initiating Event	5.3 E-004	1.0 E+000		Yes
AFW-MDP-CF-AB	Common Cause Failure of Motor Driven Pumps	2.1 E-004	2.1 E-004		No
AFW-MDP-FC-AA	Motor Driven Pump A Fails	4.0 E-003	1.0 E+000	TRUE	Yes
AFW-MDP-FC-1A	AFW Motor Driven Pump A Fails	4.0 E-003	1.0 E-001		Yes
AFW-MDP-FC-1B	AFW Motor Driven Pump Fails	4.0 E-003	4.0 E-003		No
AFW-PMP-CF-ALL	AFW Serial Component Common (CCF) to all Trains	2.8 E-004	2.8 E-004		No
AFW-TDP-FC-1C	AFW Turbine Driven Pump Fails	3.2 E-002	1.0 E+000	TRUE	Yes
AFW-XHE-NOREC	Operator Fails to Recover AFW System	2.6 E-001	2.6 E-001		No
AFW-XHE-NREC-ATW	Operator Fails to Recover AFW System During an ATWS	1.0 E+000	1.0 E+000		No
AFW-XHE-XA-SSW	Operator Fails to Align Suction to SSW	1.0 E-003	1.0 E-003		No
HPI-XHE-XM-FB	Operator Fails to Initiate Feed and Bleed Cooling	1.0 E-002	1.0 E-002		No
MFW-SYS-TRIP	Main Feedwater System Trips	1.0 E+000	1.0 E+000		No
MFW-XHE-NOREC	Operator Fails to Recover Main Feedwater	2.6 E-001	1.0 E+000	TRUE	Yes
PPR-SRV-CC-1	PORV 1 Fails to Open on Demand	6.3 E-003	6.3 E-003		No
PPR-SRV-CC-2	PORV 2 Fails to Open on Demand	6.3 E-003	6.3 E-003		No
RPS-NONREC	Non-Recoverable RPS Trip Failures	2.0 E-005	2.0 E-005		No
RPS-REC	Recoverable RPS Failures	4.0 E-005	4.0 E-005		No
RPS-XHE-XM-TRIP	Operator Manual Trip Failure	1.0 E-002	1.0 E-002		No

Table 2. Sequence conditional probabilities for LER 445/95-003, -004

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Percent Contribution
TRANS	21-8	2.0 E-005	63.1
TRANS	20	1.1 E-005	35.1
Total (all sequences)		3.2 E-005	

Table 3. Sequence logic for dominant sequences for LER 445/95-003, -004

Event tree name	Sequence name	Logic
TRANS	21-8	RT, /RCSPRESS, AFW-ATWS
TRANS	20	/RT, AFW, MFW, F&B

Table 4. System names for LER 445/95-003, -004

System name	Logic
AFW	No or Insufficient AFW Flow
AFW-ATWS	No or Insufficient AFW Flow - ATWS
F&B	Failure to Provide Feed and Bleed Cooling
MFW	Failure of the Main Feedwater System
RCSPRESS	Failure to Limit RCS Pressure to <3200 psi
RT	Reactor Fails to Trip During Transient

Table 5. Conditional cut sets for higher probability sequences for LER 445/95-003, -004

Cut set Number	Percent Contribution	Conditional Probability[a]	Cut sets
TRANS Sequence 21-8		2.0 E-005	
1	98.0	2.0 E-005	RPS-NONREC, AFW-XHE-NREC-ATW
2	2.0	4.0 E-007	RPS-REC, RPS-XHE-XM-TRIP, AFW-XHE-NREC-ATW
TRANS Sequence 20		1.1 E-005	
1	22.9	2.6 E-006	AFW-XHE-XA-SSW, AFW-XHE-NOREC, MFW-SYS-TRIP, HPI-XHE-XM-FB
2	14.4	1.6 E-006	AFW-XHE-XA-SSW, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-1
3	14.4	1.6 E-006	AFW-XHE-XA-SSW, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-2
4	9.2	1.0 E-006	AFW-MDP-FC-1A, AFW-MDP-FC-1B, AFW-XHE-NOREC, MFW-SYS-TRIP, HPI-XHE-XM-FB
5	6.4	7.3 E-007	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, HPI-XHE-XM-FB
6	5.8	6.6 E-007	AFW-MDP-FC-1A, AFW-MDP-FC-1B, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-1
7	5.8	6.6 E-007	AFW-MDP-FC-1A, AFW-MDP-FC-1B, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-2
8	4.8	5.5 E-007	AFW-PMP-CF-AB, AFW-XHE-NOREC, MFW-SYS-TRIP, HPI-XHE-XM-FB
9	4.0	4.6 E-007	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-1
10	4.0	4.6 E-007	AFW-PMP-CF-ALL, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-2
11	3.0	3.4 E-007	AFW-MDP-CF-AB, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-1
12	3.0	3.4 E-007	AFW-MDP-CF-AB, AFW-XHE-NOREC, MFW-SYS-TRIP, PPR-SRV-CC-2
Total (all sequences)		**3.2 E-005**	

[a] The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probability of the initiating events are given in Table 1 and begin with the designator "IE". The probabilities for the basic events are also given in Table 1.

GUIDANCE FOR LICENSEE REVIEW OF PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event that occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include actual initiating events, such as a loss of off-site power (LOOP) or loss-of-coolant accident (LOCA), degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences. This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and the licensee event report (LER) for this event.

Modeling Techniques

The models used for the analysis of 1995 and 1996 events were developed by the Idaho National Engineering Laboratory (INEL). The models were developed using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The models are based on linked fault trees. Four types of initiating events are considered: (1) transients, (2) loss-of-coolant accidents (LOCAs), (3) losses of offsite power (LOOPs), and (4) steam generator tube ruptures (PWR only). Fault trees were developed for each top event on the event trees to a supercomponent level of detail. The only suppor` system currently modeled is the electric power system.

The models may be modified to include additional detail for the systems/components of interest for a particular event. This may include additional equipment or mitigation strategies as outlined in the FSAR or IPE. Probabilities are modified to reflect the particular circumstances of the event being analyzed.

Guidance for Peer Review

Comments regarding the analysis should address:

- Does the "Event Description" section accurately describe the event as it occurred?

- Does the "Additional Event-Related Information" section provide accurate additional information concerning the configuration of the plant and the operation of and procedures associated with relevant systems?

- Does the "Modeling Assumptions" section accurately describe the modeling done for the event? Is the modeling of the event appropriate for the events that occurred or that had the potential to occur under the event conditions? This also includes assumptions regarding the likelihood of equipment recovery.

Appendix H of Reference 1 provides examples of comments and responses for previous ASP analyses.

Criteria for Evaluating Comments

Modifications to the event analysis may be made based on the comments that you provide. Specific documentation will be required to consider modifications to the event analysis. References should be made to portions of the LER, AIT, or other event documentation concerning the sequence of events. System and component capabilities should be supported by references to the FSAR, IPE, plant procedures, or analyses. Comments related to operator response times and capabilities should reference plant procedures, the FSAR, the IPE, or applicable operator response models. Assumptions used in determining failure probabilities should be clearly stated.

Criteria for Evaluating Additional Recovery Measures

Additional systems, equipment, or specific recovery actions may be considered for incorporation into the analysis. However, to assess the viability and effectiveness of the equipment and methods, the appropriate documentation must be included in your response. This includes:

- normal or emergency operating procedures.*
- piping and instrumentation diagrams (P&IDs),*
- electrical one-line diagrams,*
- results of thermal-hydraulic analyses, and
- operator training (both procedures and simulator),* etc.

Systems, equipment, or specific recovery actions that were not in place at the time of the event will not be considered. Also, the documentation should address the impact (both positive and negative) of the use of the specific recovery measure on:

- the sequence of events,
- the timing of events,
- the probability of operator error in using the system or equipment, and
- other systems/processes already modeled in the analysis (including operator actions).

For example, Plant A (a PWR) experiences a reactor trip, and during the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW modeling would be patterned after information gathered either from the plant FSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be

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* Revision or practices at the time the event occurred.

mitigated by the use of the standby feedwater system. The mitigation effect for the standby feedwater system would be credited in the analysis provided that the following material was available:

- standby feedwater system characteristics are documented in the FSAR or accounted for in the IPE,
- procedures for using the system during recovery existed at the time of the event,
- the plant operators had been trained in the use of the system prior to the event,
- a clear diagram of the system is available (either in the FSAR, IPE, or supplied by the licensee),
- previous analyses have indicated that there would be sufficient time available to implement the procedure successfully under the circumstances of the event under analysis,
- the effects of using the standby feedwater system on the operation and recovery of systems or procedures that are already included in the event modeling. In this case, use of the standby feedwater system may reduce the likelihood of recovering failed AFW equipment or initiating feed-and-bleed due to time and personnel constraints.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the operational event.

- The specific LER, augmented inspection team (AIT) report, or other pertinent reports.

- A summary of the calculation results. An event tree with the dominant sequence(s) highlighted. Four tables in the analysis indicate: (1) a summary of the relevant basic events, including modifications to the probabilities to reflect the circumstances of the event, (2) the dominant core damage sequences, (3) the system names for the systems cited in the dominant core damage sequences, and (4) cut sets for the dominant core damage sequences.

Schedule

Please refer to the transmittal letter for schedules and procedures for submitting your comments.

References

1. L. N. Vanden Heuvel et al., Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232) Volumes 21 and 22, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory and Science Applications International Corp., December 1995.

TU ELECTRIC

Log # TXX-95217
File # 10200
Ref. # 10CFR.73(a)(2)(iv)

August 14, 1995

C. Lance Terry
Group Vice President, Nuclear

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
ACTUATION OF REACTOR PROTECTION SYSTEM
LICENSEE EVENT REPORT 445/95-003-01

REF: 1) TU Electric letter logged TXX-95196 from C.L. Terry to the NRC dated July 11, 1995

Gentlemen:

Enclosed is Supplement 1 to Licensee Event Report (LER) 95-003-00 for Comanche Peak Steam Electric Station Unit 1, "Loss of Both Condensate and Both Feedwater Pumps Due to Failure of Non-Safety Related Inverter Resulted in a Manual Reactor Trip." The initial LER was submitted on July 11, 1995 (Reference 1).

This supplement provides additional information on the cause of the event, the corrective actions, and the preventive actions.

Sincerely,

C. L. Terry

C. L. Terry

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

GLM/cc

cc: Mr. L. J. Callan, Region IV
Mr. D. F. Kirsch, Region IV
Resident Inspectors, CPSES

ENCLOSURE 3

Enclosure to TXX- 95217

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
COMANCHE PEAK - UNIT 1	05000445	1 OF 7

Title (4)
LOSS OF BOTH CONDENSATE AND BOTH FEEDWATER PUMPS DUE TO FAILURE OF NON-SAFETY RELATED INVERTER RESULTED IN A MANUAL REACTOR TRIP

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
06	11	95	95	- 003 -	01	08	14	95	N/A	

Operating Mode (9): 1
Power Level (10): 100

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)
- [ ] 50.73 (a) (2) (i)
- [ ] 50.73 (a) (2) (ii)
- [ ] 50.73 (a) (2) (iii)
- [x] 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)

License Contact For This LER (12)

Name	Area Code Telephone Number
R. FLORES, SYSTEM ENGINEERING MANAGER	(817) 897-5590

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)

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

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

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

On June 11, 1995, at approximately 12:01 CDT, while performing a slave relay actuation test on Unit 1, a non-safety related inverter transferred from its normal inverter AC power supply to its bypass (alternate) AC power supply, which was deenergized per the slave relay test procedures. This resulted in loss of both condensate pumps. The loss of the condensate pumps resulted in the trip of both Main Feedwater (MFW) pumps. A manual reactor trip of CPSES Unit 1 was initiated due to loss of feedwater to the steam generators. The trip of both MFW pumps initiated an Auxiliary Feedwater actuation signal which started the Motor Driven Auxiliary Feedwater (MDAFW) pumps. Following a LO-LO level signal in Steam Generators 1-01 and 1-02, the Turbine Driven Auxiliary Feedwater (TDAFW) pump started as designed but tripped on overspeed. The details involving the TDAFW pump overspeed will be provided in LER 445/95-004-01.

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

An event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF) including the Reactor Protection System (RPS)(EIIS:(JC)).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On June 11, 1995, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, and operating at 100 percent power.

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

MDAFW pump 1-01 was inoperable due to alignment to its test header as required for the slave relay testing.

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

On June 11, 1995, at approximately 1201 CDT, the CPSES Unit 1 Balance of Plant (BOP) Reactor Operator (utility, licensed) was performing the Train A slave relay test for the K601A relay (EIIS:(RLY)). While performing the test, a non-safety related inverter transferred from its normal inverter AC power supply to its bypass (alternate) AC power supply, which was deenergized per the slave relay test procedure. This resulted in loss of power to auxiliary relays 1-PY/2111 & 2112 which caused a MFW pump (EIIS:(P)(SJ)) low oil pressure signal which tripped both condensate pumps. The loss of the condensate pumps resulted in a trip of both MFW pumps. A manual reactor trip of CPSES Unit 1 was initiated due to the loss of feedwater to the steam generators (EIIS:(SG)(SB)).

The trip of both MFW pumps initiated an Auxiliary Feedwater (EIIS:(BA)) actuation signal for the MDAFW pumps. MDAFW pump 1-02 started and supplied feed to Steam Generator's (SG) 1-03 and 1-04. However, MDAFW pump 1-01 was aligned to its test header as required for the slave relay testing. Following a LO-LO level signal in SG's 1-01 and 1-02 the Turbine Driven Auxiliary Feedwater pump started, but tripped on overspeed. MDAFW pump 1-01 was re-aligned to SG's 1-01 and 1-02 within approximately 8 minutes after the reactor trip. Control room personnel responded in accordance with emergency operating procedures, and the plant was stabilized in Mode 3, Hot Standby.

An event or condition that results in manual or automatic actuation of any ESF, including the RPS, is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At 1312 CDT, on June 11, 1995, the Nuclear Regulatory Commission Operations Center was notified of the event via the Emergency Notification System.

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

Control board (EIIS:(MCBD)(JE) indicators and alarms alerted the Reactor Operator (RO) that there was a loss of feedwater. The RO verified the loss of feed indications and manually tripped the reactor. The BOP RO identified that the TDAFW pump had tripped on overspeed.

II. COMPONENT OR SYSTEM FAILURES

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

The inverter's static transfer switch malfunctioned when the switch transferred to a deenergized power source. The static transfer switch is designed to prevent a transfer to a deenergized power source.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Although the precise cause of the static transfer switch malfunction could not be conclusively determined, TU Electric believes that the malfunction occurred due to inadequate transient protection in the design of the inverter and failure to calibrate the static switch logic sense printed circuit board (PCB) and the analog logic PCB. Electrical transients generated as a result of load shedding of the bypass power source may have defeated the reverse lockout circuitry causing the inverter to transfer to the deenergized bypass power source. Although PCB calibrations were initially performed on the safety related Elgar inverters during Startup, documentation for the calibration of the non-safety related Elgar inverters could not be located. The failure to calibrate the static switch logic sense PCB and the analog logic PCB may have impaired the inverter's normal response to equipment challenges such as transients and may have resulted in the transfer to the deenergized power source.

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

Not applicable - no failures of components with multiple functions have been identified.

D. FAILED COMPONENT INFORMATION

Elgar Corp.
Part Number UPS-103-1-132
118 Vac Non-Safeguard Inverter IV1C2

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The Reactor Protection System (EIIS:(JC)) and Auxiliary Feedwater System (EIIS:(BA)) actuated during the event.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

The failure of the non-safety related inverter did not result in the inoperability of any safety system trains. The failure of the Unit 1 TDAFW pump will be discussed in LER 445/95-004-01.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The actual event resulted in less auxiliary feedwater initially available than is assumed in the analysis of the "Loss of Normal Feedwater" transient presented in FSAR Section 15.2.7. This ANS Condition II event is analyzed to demonstrate the adequacy of the Auxiliary Feedwater System. The relevant event acceptance criterion is that the pressurizer should not completely fill with water, which could potentially lead to a more severe event. In this analysis, 860 gpm of auxiliary feedwater is assumed to be delivered by a combination of the two, half-capacity motor-driven auxiliary feedwater pumps and the single, full capacity turbine driven auxiliary feedwater pump, depending on the assumed single failure.

However, in the actual event, the reduction in the delivered auxiliary feedwater flow was initially offset by the effects of the early manual reactor trip, which occurred when there was more fluid remaining in the steam generators than is assumed in the FSAR analysis. The realignment of the second motor driven auxiliary feedwater pump assured that sufficient heat removal capability was available. Even with the reduced initial supply of auxiliary feedwater, the pressurizer did not completely fill with water. Thus the ANS Condition II event acceptance criterion was not exceeded and the safety and health of the public was unaffected.

IV. CAUSE OF THE EVENT

TU Electric believes that the causes of this event were the malfunction of the inverter's static transfer switch and the design of the protection portion of the condensate system which allowed a loss of power to auxiliary relays 1-PY/2111 & 2112 to cause both condensate pumps to trip on a MFW pump low oil pressure signal.

V. CORRECTIVE ACTIONS

TU Electric's initial corrective actions included repair of the identified inverter deficiencies and successful functional testing of the inverter. A Design Modification has been implemented on Unit 1 to prevent future loss of power to the subject relays from causing the condensate pumps to trip on a false low lube oil pressure resulting in a potential challenge to plant safety systems. A similar Design Modification has also been issued for Unit 2. A trip reduction initiative was completed and plant modifications have been proposed to reduce the probability of inadvertent plant trips.

The static switch logic sense PCB and the analog logic PCB have been calibrated. A review was performed of other inverters which indicated that, with the exception of the 7.5KVA Westinghouse inverters, other non-safety related inverters had also not been calibrated. These inverters will be calibrated upon completion of calibration procedures. Preventive maintenance activities will also be established to maintain calibration of all inverters. A transient analysis is being performed for the inverter involved in this event. Upon completion of this analysis, the need to perform transient analysis on other similar inverters will be determined. Other non-safety related Elgar inverters will be inspected and deficiencies will be corrected where identified.

LICENSEE EVENT REPORT (LER)

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VI. PREVIOUS SIMILAR EVENTS

There has been one previous event that resulted in an RPS actuation due to a safety-related Westinghouse RPS inverter failure (LER 445/90-002-00) and one previous event that resulted in a Technical Specification required shutdown due to a safety-related Westinghouse RPS inverter failure (LER 445/90-041-00). Corrective actions taken to resolve the root causes of the previous events would not have prevented this event.

Log # TXX-95210
File # 10200
Ref. # 10CFR50.73(a)(2)(1)

TU ELECTRIC

September 8, 1995

C. Lance Terry
Group Vice President, Nuclear

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 1
DOCKET NOS. 50-445 AND 50-446
CONDITIONS PROHIBITED BY TECHNICAL SPECIFICATIONS
LICENSEE EVENT REPORT 445/95-004-01

REF: 1) TU Electric Letter logged TXX-95196 from Mr. C. L. Terry to the NRC dated July 11, 1995
2) TU Electric Letter logged TXX-95167 from Mr. C. L. Terry to the NRC dated June 14, 1995 requesting enforcement discretion for testing of Turbine Driven Auxiliary Feedwater Pump at Mode 3
3) NRC letter dated June 15, 1995 from Mr. L. J. Callan to Mr. C. L. Terry granting enforcement discretion (NOED Tracking 95-4-0005)
4) TU Electric Letter logged TXX-95190 from Mr. C. L. Terry to the NRC dated July 14, 1995

Gentlemen:

Enclosed is supplemental Licensee Event Report (LER) 95-004-01 for Comanche Peak Steam Electric Station Unit 1, "Allowed Outage Time was Exceeded for the Turbine Driven Auxiliary Feedwater Pump which Tripped on Overspeed." The initial Licensee Event Report was submitted on July 14, 1995 (Reference 4).

Via Reference 1, TU Electric has submitted its LER-95-003-00 for Comanche Peak Steam Electric Station Unit 1, "Loss of Both Condensate and Both Feedwater Pumps Due To Failure of Non-safety Related Inverter Resulted in a Manual Reactor Trip." During this aforementioned event, the Turbine Driven Auxiliary Feedwater Pump actuated due to Lo-Lo Steam Generator alarms and subsequently tripped on overspeed. Reference 2 requested enforcement discretion to allow CPSES Unit 1 to remain in Mode 3, Hot Standby, while testing on the Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) pump was being performed. The enforcement discretion was granted via reference 3.

The subject Licensee Event Report is being submitted to satisfy the requirements of 10CFR50.73(a)(2)(1)(B), because the time requirements for the action statement were not met (see Reference 2).

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Additionally, on June 21, 1995, a TDAFW pump overspeed trip was experienced on CPSES Unit 2 in an event unrelated to this LER. This Unit 2 event is being submitted on a voluntary basis with both the Unit 1 and Unit 2 events being included in the enclosed LER.

Sincerely,

C. L. Terry

C. L. Terry

By: Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

OB:ob
Enclosure

cc: Mr. L. J. Callan, Region IV
Mr. D. F. Kirsch, Region IV
Resident Inspectors, CPSES

TXX-95210

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)

ALLOWED OUTAGE TIME WAS EXCEEDED ON TURBINE DRIVEN AUXILIARY FEEDWATER PUMP WHICH TRIPPED ON OVERSPEED

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									CPSES-UNIT 2	05000446
06	14	95	95	-- 004 --	01	09	08	95	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	3	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	000		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	X	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. FLORES, SYSTEM ENGINEERING MANAGER	(817) 897-5590

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
				N					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 11, 1995, at approximately 12:01 CDT, while performing a slave relay actuation test on Unit 1, a non-safety related inverter transferred from its normal inverter AC power supply to its bypass (alternate) AC power supply, which was deenergized per the slave relay test procedures. This resulted in loss of both condensate pumps. The loss of the condensate pumps resulted in the trip of both Main Feedwater (MFW) pumps. A manual reactor trip of CPSES Unit 1 was initiated due to a loss of feedwater to the steam generators. The trip of both MFW pumps initiated an Auxiliary Feedwater actuation signal which started the Motor Driven Auxiliary Feedwater (MDAFW) pumps. Following a LO-LO level signal in Steam Generators 1-01 and 1-02, the Unit 1 Turbine Driven Auxiliary Feedwater (TDAFW) pump started as designed but tripped on overspeed.

The Unit 1 TDAFW pump overspeed was caused by a failure of the governor to control turbine speed. The required repairs/replacements were made, the pump has been successfully tested and declared Operable. On June 21, 1995, a TDAFW pump overspeed trip was experienced on CPSES Unit 2 in an event unrelated to the Unit 1 event. TU Electric believes that the event was caused by water in the steam line which resulted from a warm-up run which was performed prior to the trip. The steam traps and the governor valve linkage were inspected, the appropriate equipment was reworked, and the TDAFWP declared operable.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

Unit 1
Any operation or condition prohibited by the plant's Technical Specifications, i.e., the event was considered reportable because the time requirements for the action statement were not met.

Unit 2
The Unit 2 event is being submitted on a voluntary basis, due to recognition of the significance and generic interest of the event .

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

Unit 1
On June 11, 1995, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, and operating at 100 percent power.

Unit 2
On June 21 1995 Comanche Peak Steam Electric Station (CPSES) Unit 2 was in Mode 1, Power Operation, and operating at 100 percent power.

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

Not applicable - no structures, systems or components were inoperable at the start of the event that contributed to the event.

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

Unit 1

On June 11, 1995, at approximately 1201 CDT, the CPSES Unit 1 Balance of Plant (BOP) Reactor Operator (utility, licensed) was performing the Train A slave relay test for the K601A relay (EIIS:(RLY)). While performing the test, a non-safety related inverter transferred from its normal inverter AC power supply to its bypass (alternate) AC power supply, which was deenergized per the slave relay test procedure. This resulted in loss of power to auxiliary relays 1-PY/2111 & 2112 which caused a MFW pump (EIIS:(P)(SJ)) low oil pressure signal which tripped both condensate pumps. The

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

loss of the condensate pumps resulted in a trip of both MFW pumps. A manual reactor trip of CPSES Unit 1 was initiated due to the loss of feedwater to the steam generators (EIIS:(SG)(SB)).

The trip of both MFW pumps initiated an Auxiliary Feedwater (EIIS:(BA)) actuation signal for the MDAFW pumps. MDAFW pump 1-02 started and supplied feed to Steam Generator's (SG) 1-03 and 1-04. MDAFW pump 1-01 was aligned to its test header as required for the slave relay testing. Following a LO-LO level signal in SG's 1-01 and 1-02, the Turbine Driven Auxiliary Feedwater pump started, but tripped on overspeed. MDAFW pump 1-01 was re-aligned to SG's 1-01 and 1-02 within approximately 8 minutes after the reactor trip. Control room personnel responded in accordance with emergency operating procedures, and the plant was stabilized in Mode 3, Hot Standby.

On June 14, 1995, at approximately 11:00 a.m. (CDT), during a teleconference with NRC Region IV Staff, TU Electric requested and was granted a Notice of Enforcement Discretion (NOED). The NOED was requested for additional time necessary to perform repairs and retesting, which would have exceeded the allowed outage time for remaining in Mode 3 and thus would not be in compliance with Technical Specification 3.7.1.2 (refer to NOED Tracking No. 95-4-0005).

Unit 2

On June 16, 1995, at approximately 1:00 p.m. a routine quarterly surveillance on the Unit 2 TDAFWP was performed. Operation's Test Crew (Utility, Licensed/Non Licensed) reported an abnormal noise coming from the pump which could indicate pump cavitation. Additionally, the Condensate Storage Tank (CST) was observed to be at the 67 percent level. Engineering (Utility, Non Licensed) was requested to evaluate the noise. While the evaluation was in progress, the CST was filled to the 85 percent level. A second cold start test was completed at approximately 9:30 p.m. on June 16, 1995, with no abnormal noise heard and the system was declared Operable. However, since these tests were run at different CST levels, and the pump performance was somewhat lower than normal, it was conservatively decided to schedule additional testing for June 21, 1995. The June 21,1995 test was to be a warm start in order to minimize turbine wear. The objective of the June 21, 1995 test was to observe any indication of degraded hydraulic performance (if present), which would indicate cavitation or internal pump wear and to investigate the source of the noise which occurred during the original test. The steam admission bypass valves were opened for four minutes to warmup the system. The pump remained aligned to the steam generators. The Control Room speed controller remained at the maximum speed setting. The bypass valves supplied enough steam to roll the turbine at approximately 3200 rpm and flow to the steam generators. Upon noticing flow to the steam generators, Operations Crew closed the steam generator

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flow control valves thus isolating flow to the generators. The bypass valves were then closed and the warmup terminated. Approximately thirteen (13) minutes after the bypass valves were closed, a quick start was performed. The Unit 2 TDAFWP was aligned in the test mode to recirculate to the CST. When the steam admission valves (2-HV-2452-1 & 2) were opened , the turbine tripped on overspeed. Unusual amounts of water were observed coming from the exhaust stack, the sentinel valve and the governor valve stem packing. The System Engineer (Utility, Non Licensed) who was observing the governor valve, noted no movement of the valve linkage during the trip. This trip occurred after warming up the steam lines. The TDAFWP was declared INOPERABLE.

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

Unit 1

The BOP RO identified that the TDAFW pump had tripped on overspeed.

Unit 2

Failure was discovered during a pump run, which was being performed to acquire pump data.

II. COMPONENT OR SYSTEM FAILURES

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

Not applicable - for Unit 1,this report is being generated due to exceedance of the specified allowed outage time requirement. However, the failure modes, mechanism, and effects of the failed component are better described in the cause of the event and the safety consequences section(s) in this LER. Unit 2 is being submitted on a voluntary basis.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

Not applicable - for Unit 1,this report is being generated due to exceedance of the specified allowed outage time requirement. However, the cause of the component/system failure is described in the cause of the event section of the LER. Unit 2 is being submitted on a voluntary basis.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

Not applicable - for Unit 1 this report is being generated due to exceedance of the specified allowed outage time requirement. However, systems/secondary functions are described in the safety consequences section of this LER. Unit 2 is being submitted on a voluntary basis.

D. FAILED COMPONENT INFORMATION

Manufactured by: Dresser Rand
Part Name : Horizontal Valve Assembly with cam crank-Remote Governor servo linkage

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

Unit 1

No Safety system responses occurred due to this event.

Unit 2

No Safety system responses occurred due to this event.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

The Unit 1 TDAFW pump was inoperable for approximately 4 days 10 hours and 2 minutes. The allowed Technical Specification outage time is 3 days.

The Unit 2 TDAFW system inspections and corrective actions were completed within the allowed outage time i.e. the Unit 2 TDAFW pump was inoperable for approximately 69 hours.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The Auxiliary Feedwater (AFW) System is designed to supply an independent source of water to the steam generators during accident and transient conditions in the event of a loss of main feedwater. The major components of the CPSES AFW System are three essential safety-grade pumps, one turbine-driven pump (TDAFWP) and two motor-driven pumps (MDAFWPs). The AFW supply is provided by the condensate storage tank. The backup supply for the AFW system is the service water system.

The AFW System is designed to accommodate a single failure in any active system component without loss of function. Each of the two MDAFWPs supplies two of the four steam generators. The TDAFWP supplies all four steam generators. The MDAFWP and the TDAFWP are connected together downstream of the AFW valves before the connection to the feedwater bypass line. The MDAFWPs are also cross connected, through normally closed manual valves in series, to allow either MDAFWP to supply any of the steam generators after operator action to open the valves. The two MDAFWPs are provided with one suction connection to the condensate storage tank. The TDAFWP is provided with a separate suction connection to this tank. Steam supply to the TDAFWP is provided from two of four steam generators through separate air-operated valves which fail open on loss of the air supply. Thus, adequate feedwater is assured to at least two steam generators in the event of a high-energy pipe break or other postulated design-basis accident concurrent with a single failure.

The TDAFWP provides a diverse means of assuring feedwater supply to the steam generator independent of all offsite or onsite AC power sources.

The AFW System is required to function after any plant trip described in FSAR Chapter 15. With few exceptions, the initiating event does not affect the capability of the AFW System to perform its intended safety function; therefore, these events are unaffected by the status of the TDAFWP.

The TDAFWP is required to be operable in the analysis of the Feedwater Line Break presented in FSAR Section 15.2.8. In this analysis, one MDAFWP is assumed to be the single failure. The second MDAFWP is assumed to deliver its entire contents to the faulted steam generator, and the TDAFWP is assumed to deliver 430 gpm to the three intact steam generators. (In reality, one would expect the second MDAFWP to deliver somewhat more than half of its capacity to the affected steam generator (an intact steam generator would receive the remaining fluid). This American Nuclear Society (ANS) Condition IV event is assumed to be initiated from full power and is analyzed to ensure that the core remains in a coolable geometry. This condition is satisfied by demonstrating that no voiding occurs in the hot leg.

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The TDAFWP is also assumed to be operable in the analyses of the Loss of Non-emergency Power to the Station Auxiliaries and Loss of Normal Feedwater transients presented in FSAR Sections 15.2.6 and 15.2.7. These events are assumed to be initiated from full power and are analyzed to demonstrate that the AFW system can remove enough heat to prevent the pressurizer from filling to the point where water relief through a safety or relief valve occurs. For these ANS Condition II events, water relief is equated with valve failure to close, thereby allowing the event to progress to a more serious accident. In this analysis, a minimum of 860 gpm is assumed to be provided by any combination of AFW pumps.

In addition, the AFW System would be used to provide a source of AFW following any plant trip. The TDAFWP is also the sole source for AFW following a station blackout.

The intended safety function of the AFW System is to provide adequate AFW to an adequate number of steam generators such that, when considering a single failure, all events are shown to meet their relevant event acceptance criteria.

Event 1, Unit 1 Turbine Driven Auxiliary Feedwater Pump Overspeed Trip

The actual event resulted in less auxiliary feedwater initially available than is assumed in the analysis of the "Loss of Normal Feedwater" transient presented in FSAR Section 15.2.7. This ANS Condition II event is analyzed to demonstrate the adequacy of the Auxiliary Feedwater System. The relevant event acceptance criterion is that the pressurizer should not completely fill with water, which could potentially lead to a more severe event. In this analysis, 860 gpm of auxiliary feedwater is assumed to be delivered by a combination of the two motor-driven auxiliary feedwater pumps and the turbine driven auxiliary feedwater pump, depending on the assumed single failure.

However, in the actual event, the reduction in the delivered auxiliary feedwater flow was initially offset by the effects of the early manual reactor trip, which occurred when there was more fluid remaining in the steam generators than is assumed in the FSAR analysis. The realignment of the second motor driven auxiliary feedwater pump assured that sufficient heat removal capability was available. Even with the reduced initial supply of auxiliary feedwater, the pressurizer did not completely fill with water.

Even if the reactor operator had not tripped the reactor, an automatic reactor trip would have occurred soon after the loss of main feedwater on Steam Generator Lo-Lo level. The introduction of a single train of auxiliary feedwater to two steam generators along with the availability of the steam dumps and/or ARVs, is sufficient to prevent overfilling the pressurizer prior to reactor trip. After the reactor trip, the single train of AFW is sufficient to maintain cooling of the RCS until such time that the operator can realign the second MDAFWP flow to begin an RCS cooldown. Thus, even without operator action in the short term, the ANS Condition II event acceptance criterion was not exceeded and the safety and health of the public was unaffected.

Event 2, Unit 2 Turbine Driven Auxiliary Feedwater Pump Overspeed Trip

Upon the overspeed trip of the TDAFW pump, a 72 hour Tech Spec action statement was entered. Repair activities and testing were completed and the system was returned to OPERABLE status.

Based on this discussion, it is concluded that the event did not adversely affect the safe operation of CPSES Unit 2 or the health and safety of the public.

IV. CAUSE OF THE EVENT

Unit 1

This event was considered reportable because the time requirement for the action statement was not met. An NOED was requested and received before a violation of the Technical Specification occurred.

The Turbine Driven Auxiliary Feedwater Pump(TDAFWP) overspeed trip was caused by a failure of the Governor Valve to control turbine speed. . A Task Team was established by TU Electric management to determine probable causes, the contributing causes and to recommend actions to correct and minimize issues surrounding this event. The findings of the Task Team are stated below:

PROBABLE CAUSES

1) The Governor Valve stem was discovered corroded and was binding with the packing.

2) Investigation following the overspeed trip found the operation of governor valve cam linkage assembly to be binding slightly. This binding may have been sufficient, when combined with stem corrosion to prevent the governor from adequately controlling the TDAFWP.

Unit 2

The Task Team concluded that the Unit 2 TDAFWP overspeed trip was caused by a failure of the governor valve to control turbine speed. The findings of the Task Team are stated below:

PROBABLE CAUSES

TU Electric believes that the event was caused by water in the steam lines which resulted from a warm up run performed 13 minutes earlier. This water restricted movement of the governor which left the governor incapable of controlling speed during this start. Additionally, degraded traps and slight binding in the governor valve cam linkage were potential contributors to the event.

V. CORRECTIVE ACTIONS

On June 14, 1995, at approximately 11:00 a.m. (CDT), during a teleconference with NRC Region IV Staff, TU Electric requested and was granted a Notice of Enforcement Discretion (NOED). The NOED was requested for additional time necessary to perform repairs and retesting, which would have exceeded the allowed outage time for remaining in Mode 3 and thus would not be in compliance with Technical Specification 3.7.1.2 (refer to NOED Tracking No. 95-4-0005). TU Electric was cognizant of the Technical Specification requirements; therefore, no corrective actions for this issue were required.

Unit 1 TDAFWP

Subsequent to initial trouble shooting, the valve stem was changed out to a new inconel stem. During valve reassembly some stickiness was noted in the cam follower assembly. Parts were disassembled, cleaned, inspected and reassembled and freedom of movement was verified. Insulation was removed from selected drain lines so that water levels could be monitored. Water level in the drain pot upstream of the turbine was at the level of the drain line tap each time it was checked with the system shutdown for various lengths of time, indicating the steam traps were functioning properly.

TU Electric has repaired/replaced the defective parts. The TDAFW pump had been successfully tested and was declared Operable on June 15, 1995, at approximately 10:05 p.m.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Unit 2 TDAFWP

The degraded steam traps and the governor valve cam linkage were reworked. Disassembly and replacement of the Unit 2 governor valve stem with inconel was subsequently accomplished. The TDAFW pump was successfully tested and was declared Operable on June 24, 1995, at approximately 4:00 p.m.

Additionally, TU Electric is evaluating the contributing causes and the recommendations as determined by the Task Team in order to implement additional corrective actions if warranted.

VI. PREVIOUS SIMILAR EVENTS

There has been one other previous event which resulted in exceeding of Technical Specification action statement (refer to LER 445/95-001-00). However, the causes for the aforementioned event were significantly different than the subject event. Corrective actions taken to resolve the root causes of the previous event would not have prevented this event.