November 18, 1990

CITIZENS FOR FAIR UTILITY REGULATION
7600 Anglin
Fort Worth, Texas 76140

The Executive Director of Operations
The United States Nuclear Regulatory Commission
Washington, D.C. 20555

Request to institute a proceeding or for such other action as may be proper under 10 C.F.R., subpart B S2.206

Pursuant to 10 C.F.R., subpart B, S2.206, Citizens for Fair Utility Regulation (hereinafter referred to as CFUR) files this petition with the Executive Director of Operations for the United States Nuclear Regulatory Commission requesting the Director to institute a proceeding to determine if the operating license issued to Texas Utilities Electric Company for the Comanche Peak Steam Electric Station should be revoked for the reasons outlined below.

CFUR, a Texas based consumer and environmental organization, represents persons who live and work in the vicinity of the Comanche Peak Steam Electric Station (CPSES). CFUR has monitored the construction and operation of the CPSES and has filed numerous petitions with the NRC and the federal courts regarding issues of safety at the plant during its construction period.

On October 16, 1989, CFUR filed with the Commission a request for a stay of fuel load and the issuance of a low power license for the Comanche Peak plant, based on a series of outstanding safety issues which CFUR believed had not been resolved in ways whereby the applicant TU Electric could guarantee, with reasonable assurance, to the NRC staff that the plant could operate without endangering the health and safety of the public. (CFUR Request for Stay, October 16, 1989, as Attachment A.)

On October 19, 1989, the Commission denied CFUR's stay request but ordered the NRC staff to address the issues of safety raised by CFUR before the issuance of the low power license. (Commission Order, October 19, 1989, Attachment B.)

One of the key issues of safety raised by CFUR involved the multiple failures of a number of Borg-Warner check valves in the auxiliary feedwater system (AFW) and the service water system which occurred in four separate events during hot functional testing in April and May of 1989. (Augmented Inspection Team Report, 50-445/89-30, 50-446/89-30, July 10, 1989, Attachment C.)

The July 10 report clearly identifies and outlines the series of valve failures and the damage these failed valves caused to the piping system during the events in April and May. The report also clearly identifies what could have occurred had the reactor been loaded with nuclear fuel, that is, that radiation could have escaped into the environment. CFUR will not summarize those events here, but will refer the director to the attached report of July 10.

On October 27, the NRC issued a follow-up report to 89-30 which said to TU, "your evaluations...lack thoroughness and depth, and your corrective actions were ineffective and untimely." (Attachment D, page 1.)

There were serious safety issues raised by the failures of these particular check valves. But an even more serious issue raised at the time related to TU Electric management failures and the number of precursor events involving these Borg-Warner valves since 1983, which TU failed to take seriously and/or to correct adequately. <u>Had the valves been corrected, or replaced, the failures in April and May would not have occurred.</u>

The series of events and the inspections following the events, resulted in a Notice of Violation and Imposition of Civil Penalties ($30,000) on January 25, 1990. Three violations were cited. It is important to note that TU had no preoperational testing program for these valves, and no tests were conducted following the valve failures in 1983 and 1985. (Aforementioned notice, Attachment E.)

A large number of reports have been issued by the NRC, TU Electric and Kalsi Engineering, Inc. (refer to Kalsi Report to TU, November 30, 1989) detailing the history of the failed valves, corrective actions to be taken, and analyses of TU's corrective action program. Other NRC Inspection and Enforcement reports have been issued since licensing which show that the valves continue to fail. CFUR will refer to some of those reports and will summarize our concerns about some of those reports in the body of this petition. However, because of the large volume of information which has been generated, CFUR asks the director to review all of the reports in full and in sequence--<u>including any reports to which CFUR may not have had access</u>--in order to understand fully the pervasive pattern of breakdown in TU's corrective action program as it relates to these valves.

The most significant report to date was issued on January 12, 1990 by the NRC following an inspection by Nuclear Reactor Regulation inspectors during the week of September 11-14, 1989, of the Borg-Warner facility (BW/IP International, Inc.) at Vernon, California. The January 12 inspection was a direct result of the failed Borg-Warner valves at Comanche Peak. (January 12 letter from the NRC to BW/IP International, and Notice of Violation, EA-89-244, Attachment F.)

NRC Region IV did not receive this report until October 16, 1990. Therefore, CFUR had no way of knowing about a report <u>critical to the licensing decision--produced before the licensing decision--until almost 10 months after the licensing decision.</u>

The existence of the report itself; the serious safety issues it raises about the valves installed at Comanche Peak in Units 1 and 2;

the fact that the report was in existence less than a month before CPSES was licensed and may not have been known to the NRC's Region IV at that time, raises serious questions about the integrity of the licensing process and the safety of the plant itself.

Serious questions are also raised about the competence and integrity of TU officials and their commitment to the safe operation of a nuclear facility. Further, CFUR is concerned that the Commission, in making the critical decision to issue a full power license to TU Electric to operate Comanche Peak, may not have known about the January 12 report, the serious questions of safety it raised, not to mention additional questions regarding TU's commitment to follow the law. Further, CFUR believes that Thomas Murley, the Director of Nuclear Reactor Regulation, knew of the report and its findings prior to the issuance of a low power license to TU Electric in February, 1990, and chose to ignore it. Director Murley's office conducted the inspection of the Borg-Warner plant.

Certainly CFUR believes that the intent of the Commission's order to the NRC staff to address the issues of safety raised by CFUR was not met. In a Janauary 30 letter to CFUR from James E. Lyons, Chairman, Allegation Review Committee, Comanche Peak Project Division, with an attached report on the resolutions of those issues, including the failed check valves, no mention is made of the January 12 report and Notice of Violation against Borg-Warner. (January 30 letter to Mrs. Betty Brink, Board Member, CFUR, and attached enclosure, Subject Allegation OSP-A-0089, Attachment G.)

In that report, Mr. Lyons, on behalf of the NRC staff and with assurances from TU Electric writes, on page 4,

> The NRC staff has concluded that the applicant's corrective action program to reset and control the bonnet elevation of Borg-Warner check valves will effectively prevent the previously observed phenomenon where the valve disk jammed under the seat ring...(T)he applicant's commitment to conduct functional backflow test and/or radiographic examination for each valve will provide reasonable assurance that all Borg-Warner check valves are capable of performing their design function.

and, further down the same page,

> An extensive engineering analysis was performed to demonstrate the acceptability of the swing arms (in the service water system) which were not replaced. That analysis is now under review and the NRC will <u>ensure that the check valves operate properly prior to making a decision on a Unit 1 fuel load license.</u> (Emphasis added.)

However, Mr. Lyons, the NRC staff, and TU Electric were aware on January 30 of the findings of the January 12 report which cast doubt on any quick resolution of the Borg-Warner valve problems. A copy of the report had been sent to TU.

In the cover letter of the January 12 report, Brian K. Grimes, Director, Division of Reactor Inspection Safeguards, Office of Nuclear Reactor Regulation, writes,

> TU Electric informed the NRC of a broken cast swing arm, a critical component internal to the swing check valve, and several other swing arms which failed...metallurgical tests. These valves were installed in several <u>key safety-related systems at CPSES and raise concerns over the improper use of commercial grade nonpressure boundary items in safety-related applications.</u> (Emphasis added.)

Brian Grimes' letter continues:

> (T)he implementation of your quality assurance program failed to meet certain NRC requirements. The most significant problem was the failure of BW/IP to adequately qualify suppliers of internal parts...which were subsequently installed in safety-related valves and pumps furnished to the nuclear industry. In one example BW/IP had no documentation to support the use and qualification, since <u>1985</u>, of ACME Castings, Inc., as a supplier of cast valve internals, including swing arms, which have been installed in swing check valves used in nuclear safety-related applications. ACME's quality program had been found unacceptable in 1985 by BW/IP; however, they were retained and utilized as an approved vendor without a documented basis.

And, incredibly, the cover letter states,

> <u>A recent order for replacement swing arms for the CPSES was supplied by ACME.</u> (Emphasis added.)

The letter continues,

> (C)ontrary to BW/IP procedures, BW/IP failed to perform implementation audits for suppliers holding a current Certificate of Authorization issued by the American Society of Mechanical Engineers (ASME).

One of those companies was Atlas Foundry & Machine Co., from which BW/IP ordered replacement swing arms for CPSES. The letter notes, however,

> (L)icensees and their subcontractors are responsible for ensuring that the supplier is effectively implementing the approved QA program as discussed in NRC Information Notice 86-21, issued March 31, 1986. (Emphasis added.)

TU, ultimately, was responsible, according to the law.

Page 2 of the cover letter states,

> The inspectors also identified that BW/IP performed an inadequate review for suitability of 8 commercial grade replacement swing arms for safety-related use at CPSES. BW/IP's verification was inadequate with respect to verifying the mechanical and chemical properties of the swing arm material. (And) the results of BW/IP's visual and dimensional inspection were not documented.

The inspection resulted in a Severity Level III Violation because "a Part 21 report by BW/IP or notification of a significant deviation to NRC licensees would have been required if BW/IP had adequately performed the required evaluation. This violation is of significant regulatory concern." (Emphasis added.)

A copy of the letter and the report was forwarded to TU and ASME.

In light of the promises that were being made by TU to the NRC prior to licensing regarding the corrective actions TU would take, the existence of the January 12 report raises troubling questions that the Director and, ultimately, the Commission must address. For example, on October 19, 1989, a month after the inspection of the Borg-Warner facility, but before the report was officially published, TU Electric's Executive Vice-President, William Cahill, in a briefing before the Commission and with the NRC staff present, assured the Commissioners that TU would correct the check valve failures prior to licensing. From the transcript, page 21,

> MR. CAHILL:...As you are aware, during hot functional testing, deficiencies were identified related to check valve backflow and out of sequence performance of a step in a test. TU Electric, as well as the NRC, conducted extensive evaluation to determine the causes and corrective action to resolve these deficiencies.
> (Slide) We are implementing the corrective actions and post modification testing which assure us that these check valves function as designed. (Emphasis added.) (Attachment H).

However, the final report from the NRC regarding the results of the inspection of Borg-Warner was not out, much less had there been time for Borg-Warner to respond to the charges the NRC raised. Neither Mr.

Cahill nor the NRC staff which was present, including Thomas E. Murley of the NRR whose office had performed the inspection at Borg-Warner, could know what was going to be required to assure that the seriously deficient check valves would perform their design function. No one challenged Mr. Cahill, nor did anyone who knew about the problems at Borg-Warner alert the Commission that an inspection at Borg-Warner had been conducted in September and a report was forthcoming.

Following the July 10 report, the NRC objected to most of TU's initial plans to correct the check valve problem before fuel load. (These objections are contained in the attached October 27 report.) For example, TU Electric stated to the NRC in an August 18, 1989, report (Attachment I) that it would use ultrasonic inspections to verify that no plastic deformation had occurred in areas where the piping code allowable stress was exceeded due to excessive heat and pressure. This condition occurred when the check valves failed in April and May of 1989, releasing excessively hot water (500 degrees F) into pipes not designed to withstand such heat.

In a reply dated September 14, 1989, the staff states,

> Without base line thickness measurements (which did not exist) taken prior to the event, ultrasonic inspections cannot establish whether plastic deformation occurred. Therefore, there is <u>no basis</u> for your conclusion that the piping stresses due to this event were in the elastic range. (Emphasis added.)

Yet, in NRC I & E Report 50-445-90-03, 50-446/90-03, published on February 16, 1990, a week after the license was issued, the staff allowed TU to rely on ultrasonic and radiographic inspections without the necessary base line thickness measurements,

> Subsequently, TU Electric performed radiographic and ultrasonic inspections of the areas in the piping...and verified that no damage had been incurred during the events...

The February 16 report states that several of the check valves "continued to leak."

However, CFUR believes that the allowable corrective actions, quoted below, remain questionable.

> Approximately 13 Borg-Warner check valves in the auxiliary feedwater system have excessive body to bonnet external leakage. Valves were disassembled, honed to remove scratches in the body throat and provide better sealing surfaces and reassembled...several of these check valves continue to leak and are scheduled to be "hot

torqued" in Mode 3...TU anticipates that the extra pressure will seal the valves. (Pages 7-15, February 16 Report Attachment J).

Even before CFUR was aware of the questions raised in the January 12 report, we believed that the honing of sealing surfaces and subsequent leakage indicated that either the bonnet or the body had been warped. Without precision machining the valves will probably continue to leak and, in fact, have done so. The "hot torqued" solution for sealing a leaking valve is unsatisfactory since the procedure may cause the valve to change in configuration when it cools, and leakage could again occur.

Further CFUR concerns regarding the valves relate to the on-going disassembly of the valves in attempts to correct the leaking problems. On July 10, the NRC staff noted that disassembly and reassembly may have played a part in the problems during hot functional testing.

On page 10 of the February 16 report (enclosed), the NRC closes out Open Item 445/8973-O-08, which had been carried over from the July 10 report. This open item referred to the steam generator water flowing in the reverse direction through the feedwater isolation bypass valves (FIBV) and in the forward direction through the preheater bypass valves to the AFW piping. TU apparently convinced the NRC of their "intent to isolate the feedwater isolation bypass valves during startup and shut down conditions to preclude...similar backflow events in the future." The fix was to require the FIBV downstream manual isolation valves to "remain closed" whenever the AFW system was in use to feed the steam generators.

However, between April 24 and May 1, 1990, three months after licensing, four incidents with backleakage occurred in the systems described above, again causing excessive temperatures as a result of "backleakage across the seat of BW/IP 4" pressure seal check valves which serve to isolate the AFW system from the main feedwater system." (Letter TXX90188 from TU to the NRC, May 18, 1990, Attachment K.)

(These series of events are discussed more fully later in this petition.)

In reports issued on October 30 (50-445/89-73 Attachment L) and December 21, 1989 (50-445/89-84 Attachment M) the inspectors determined that there is "no documentation" to support TU's revision of a root cause analysis regarding a failed Borg-Warner valve in 1985. <u>Had TU followed up on that failed valve, the inspectors determined, there would have been no failed valves in 1989.</u>

At this point CFUP would emphasize that in many of the cited reports relating to Borg-Warner and TU Electric, a "failure to document" is a consistently prevailing theme. NRC regulations and the NRC regulatory scheme insist on easily retrievable documentation for reasons that are obvious and correct to this petitioner. Not only is a "paper trail" needed to help prevent an accident or to mitigate an accident in progress, but the NRC also needs assurance that the licensee is committed to following the law, committed to quality, and understands the catastrophic consequences that could result from its failure to do so.

When the licensee fails to document, or cannot produce documentation of, its contacts with its vendors, as in this case, regarding a failed safety system, and then that vendor cannot produce documentation to support its continued use of an "unacceptable" company such as ACME, which supplied parts to the failed safety system, then the whole system of regulatory laws breaks down. The intent of the Atomic Energy Act to protect the health and safety of the public by requiring strict adherence to the regulations, is made a mockery.

In the case of TU Electric and its reliance on Borg-Warner, the proof is in the pudding. The check valves continue to fail and have never been able to perform their design function. The first failures were found by the NRC to have occurred in 1983, and those failures continue to this day. No corrective actions have been taken that were adequate or more than "short term" solutions.

By the time TU Electric received its license to operate, it had already purchased replacement parts for the failed Borg-Warner check valves in safety-related systems from a company, ACME Inc., found "unacceptable" by TU's vendor, Borg-Warner. On January 5, 1990, two of the check valves which had been repaired "continued to hang up", making them potentially "inoperable." (See January 5, 1990 Daily Report).

To further compound the errors, TU has taken replacement parts from Borg-Warner check valves installed in Unit 2 for replacement in Unit 1 even though these valve internals are from the same type of valves which have failed! In a letter of May 18, 1990, TU tells the NRC that "The internals of eight BW/IP check valves from Unit 2 will be...modified for the installation into Unit 1...(to be completed) during the next cold shutdown period of sufficient duration." (Page 3, Letter from TU to the NRC, May 18, 1990, TXX-90188, Attachment N.)

The letter referred to above was in response to NRC staff requests concerning continuing problems with Borg-Warner valves identified between April 24 and May 1, 1990. There were four incidents during that period, almost three months after licensing (see page 7 of this petition):

> 1.) Overheating of AFW piping; 2.) seat leakage across feedwater preheater bypass valves; 3.) sticking feedwater isolation valves; and 4.) a decrease in FWIV body temperature below the specified 90 degrees Fahrenheit setpoint with the valve pressurized. (Id., page 1)

The letter identifies conditions on April 24 and 25, 1990, in which the AFW system piping reached a temperature of 165 degrees Fahrenheit (in excess of the design temperature of 140 degrees F.) The condition stemmed from backleakage across the seat of BW/IP 4" pressure seal check valves which serve to isolate the AFW system from the main feedwater system. Preheated feedwater was flowing through the open feedwater preheater bypass valves back through leaking AFW check valves, (Id. page 1.)

On April 28, AFW line temperatures increased even though the feedwater preheater bypass valves were closed. AFW check valve leakage was causing leakage past the valves. On April 30, following the shutdown of the Number 2 AFW motor driven pump, which was run to attempt to reduce the leakage on one of the leaking AFW check valves, the line temperatures increased to 235 degrees Fahrenheit with the the FPBV's closed. (Id. page 3.)

On April 27, operations personnel could not open the four feedwater isolation valves due to binding caused by differential thermal expansion. The use of a hydraulic lifting device was used to help the operator lift the valve discs off their seats. (Id. page 4.)

Yet, on April 27, William Cahill again assures the NRC that TU would vent the upstream side of check valves as necessary to seat the check valves more tightly, allowing piping temperatures to stabilize, and that "all BW/IP check valves will perform their intended safety function." (April 27 letter TXX90172 from Cahill to the NRC, page 2, Attachment O.)

In reviewing these latest reports, CFUR would have the Director note that there is no clear indication of how this venting was accomplished or what ultimately resulted. Moreover, venting is <u>upstream</u> of the check valve. If the check valve will not close with low differential pressure, venting should result in higher differential pressure and the check valve should close tightly. On page 2 of TXX90188, the statement is made that "Because upstream valves were not leaking, pressure equalized across the auxiliary feedwater check valves. This allowed the valve disc to open slightly permitting backflow." CFUR would ask, "Where did the water go?" Since the stop valve was closed, there should have been no backflow. CFUR believes that these valves are still jammed open just as they were during the hot functional testing of over a year ago. If the check valves were tight, opening the upstream stop valve would provide the pressure difference necessary to guarantee the valves' integrity.

Note that in the NRC's letter to CFUR, the NRC states that "applicant's corrective action program to reset and control bonnet elevation of BW check valves will effectively prevent jamming of the disc below the valve seat." January 30, 1989 letter to CFUR attached.)

The most notable departure from the August 18, 1989 letter from TU to the NRC concerning the same equipment is the short-term solution to the problem represented by "upstream venting of the check valves in order to faciliate more positive seating of the valves." This seems to be an extraordinary solution to CFUR, since the August 18 report states on page 8 that, "The April 23 and May 5 events were of no immediate safety significance because there was no fuel in the reactor and Unit 1 was not radioactive. (Emphasis added.)

The fixes now proposed raise the potential for radioactive contamination. Almost one year to the day after the April and May events of 1989, with fuel loaded and Unit 1 radioactive, there were still auxiliary feedwater leaks, feedwater isolation valves that must be opened by hydraulic lifts, leaking check valves, and now upstream venting in the fond hope that venting will work (a "short term" solution.) TU Management indecision still appears to dominate.

On May 16, the NRC staff wrote in a summary of a meeting with TU regarding the continuing problems with Borg-Warner, that TU was again proposing long-term solutions including more modifications of the existing valves, replacement of some valves, or modification of the existing AFW system. (Letter, May 16, Attachment P.) In that meeting the NRC staff raised concerns about the hydraulic lifts, saying that damage could occur to the valves from excessive lifting forces.

By July 27, TU's William Cahill had committed to replace swing arms during the first refueling outage. He writes that 24 BW/IP check valves have been replaced with investment cast swing arms. The question must be answered as to whether these replacements were from the unqualified suppliers to Borg-Warner, such as ACME, Inc. We know from the January 12 report that 8 swing arm replacements were ordered by Borg-Warner for CPSES from ACME. (July 27 letter, Attachment Q.)

The failure of the Borg-Warner check valves, contrary to TU assurances and the NRC staff's acceptance of TU's promises, has not been resolved. Solutions have only been proposed; TU has made commitments to corrective action reactively not proactively. Even more, TU has in some cases made commitments cynically simply to expedite the licensing. Finally, all proposed solutions must be suspect if they rely on Borg-Warner in any way.

Certainly the existence of the January 12 report and TU's reliance on Borg-Warner over the years for guidance and for replacement parts raises questions of profound significance regarding the safe operation of the Comanche Peak facility and the competence of TU management.

TU has made misleading statements to the NRC staff that the valves would be corrected and performing their design function "prior to licensing." TU has assured the Commission of the same thing. Even TU's attorney, Mr. George Edgar, in responding to CFUR's appeal to the U. S. Supreme Court, told the court on August 13, 1990, that "The problems with the check valves were corrected by TU Electric and inspected by the NRC." (Page 10, Brief for Respondent Texas Utilities Electric Company in Opposition, Attachment R.) This is simply untrue.

Incredibly, TU continues to assure the NRC that these valves will perform their design function. CFUR could ask the obvious question, "When?" The plant has been licensed since February, and the check valves continue to leak.

In closing, I would like to quote from Mr. Chris Grimes, head of the NRC Office of Special Projects, during a meeting on December 7, 1989, with CFUR in response to the Commissions's Order of October, 1989. Mr. Grimes was asked, in regard to the check valve failures, if there was ever a point in time when the NRC would say to a utility "we will give you no more time to get it right."

After some hesitation, Mr. Grimes replied, "To my knowledge, there is usually only two paths. One is enforcement and the other is issuance of an order to show cause why a license might not be revoked. Those have normally followed the issuance of enforcement actions that are severity level one or two. That is, they are matters where they made mistakes so bad that they have actually put public health and safety at risk. <u>They normally only get that opportunity after the license is issued." (sic!)</u> (Transcript, page 56-57, Attachment S.)

We believe TU now has "earned" that opportunity. For all of the above reasons, CFUR prays that the Director will institute a proceeding and require TU to show cause why its license to operate the Comanche Peak Steam Electric Station should not be revoked.

Respectfully submitted,

Betty Brink

Betty Brink,
Board Member, CFUR
7600 Anglin Drive
Fort Worth, Texas 76140
817-478-6372

Enclosures: 19

CC: Nuclear Regulatory Commission, Office of Inspector General
LLoyd Bentsen, U.S. Senate
John Glenn, U.S. Senate
Pete Geren, U.S. House of Representatives
Edward J. Markey, U.S. House of Representatives
John Breaux, U.S. Senate
Nuclear Information and Resource Service
Union of Concerned Scientists
Public Citizen
Selected media
(No Attachments)

Attachment A

RICHARD LEE GRIFFIN
ATTORNEY & COUNSELOR AT LAW
600 NORTH MAIN
FORT WORTH, TEXAS 76106
(817) 870-1401

October 15, 1989

The Honorable Samuel J. Chilk
Secretary, Nuclear Regulatory Commission
United States Nuclear Regulatory Commission
Washington, D.C. 20555

RE: Request for Stay filed by Citizens for Fair Utility Regulation in CLI-88-12.

Dear Mr. Chilk:

Enclosed you will find an original and four copies of the document referred to above. Please file this with the Commission and bring it to their attention.

I am serving the parties to this proceeding as indicated in the certificate of service.

Sincerely yours,

Richard Lee Griffin

Richard Lee Griffin

~~8910180146~~

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the Matter of	§	
	§	
TEXAS UTILITIES ELECTRIC COMPANY, et al.	§	Docket Nos. 50-445-OL 50-446-OL
	§	
(Comanche Peak Steam Electric Station, Units 1 & 2)	§	Docket No. 50-445-CPA
	§	

REQUEST FOR STAY

CITIZENS FOR FAIR UTILITY REGULATION

Richard Lee Griffin
Counsel for Citizens
For Fair Utility Regulation

October 15, 1989

8910180146

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

In the Matter of	§	
TEXAS UTILITIES ELECTRIC COMPANY, et al.	§	Docket Nos. 50-445-OL 50-446-OL
(Comanche Peak Steam Electric Station, Units 1 & 2)	§	Docket No. 50-445-CPA

REQUEST FOR STAY
CITIZENS FOR FAIR UTILITY REGULATION

Citizens for Fair Utility Regulation, hereinafter referred to as CFUR, filed a petition for review in the United States Court of Appeals for the Fifth Circuit, seeking review of the Nuclear Regulatory Commission order CLI-88-12 denying CFUR's petition to intervene in this proceeding. All briefs have been filed in the court of appeals, and the record will be filed on or before October 24, 1989. The case will not be submitted for the court's consideration until the record is filed.

TU Electric, the applicant, has announced its intention to request in the immediate future, a license authorizing fuel loading and low power testing. If a decision directing the issuance or amendment of an operating license is made, it is effective immediately upon issuance, and the Director of Nuclear Reactor Regulation is commanded by regulation to issue the [illegible] or amendment within ten days. 10 C.F.R. § 2.764 (a) and [illegible] Licenses to load fuel and to operate up to five percent of [illegible] power are specifically excluded from automatic review by the Commission and are immediately effective. 10 C.F.R. § 2.764 (f). Other licensing decisions are considered stayed for thirty days pending review of the initial decision by the Commission; fuel loading and low power testing decisions are not. 10 C.F.R. § 2.764 (f)(2)(iii). However, the Commission retains the authority to order that a fuel loading and low power license not be immediately effective. 10 C.F.R. § 2.764 (a).

A stay may not ordinarily be requested from the court of appeals unless it is first requested from the agency. Fed. R. App. P. 18. The stay provisions of 10 C.F.R. Part 2 apply to motions by parties or to Commission review on its own motion. 10 C.F.R. §§ 2.788 and 2.764 (f)(2). The Commission's denial of CFUR's petition to intervene has left CFUR a nonparty for such purposes. However, considering the policy underlying Fed. R. App. P. 18, and considering the Commission's authority to deny immediate effectiveness of initial licensing decisions, 10 C.F.R. § 2.764 (a), CFUR requests the Commission to entertain this request for a stay. Specifically, CFUR requests the Commission to stay the otherwise immediate effectiveness of an initial decision to grant a fuel loading and low power license in this proceeding, and to stay the issuance of such a license by the Director of Nuclear Reactor Regulation. CFUR requests such a stay pending the resolution by the court of appeals of CFUR's petition for review. Should the Commission deny a stay pending final order of the court of appeals, CFUR requests a temporary stay for a reasonable time within which to apply to the court of appeals for a stay under Fed. R. Civ. P. 18.

The regulation governing stays directs the Commission to consider whether the moving party has made a strong showing that it is likely to prevail on the merits; whether the moving party will be irreparably injured unless a stay is granted; whether other parties would be harmed if a stay were granted; and where the public interest lies. 10 C.F.R. § 2.788 (e). The same factors are used by the courts to determine whether or not to grant a stay. See Virginia Petroleum Jobbers Ass'n v. FPC, 259 F.2d 921, 925 (D.C. Cir. 1958).

1. Is the moving party likely to prevail?

It should be noted from the outset that this question does not imply that the moving party must show with mathematical logic that its chances of winning the appeal are better than fifty percent. If the movant were required in every case to show that the appeal would probably be successful, the rule would not require that application first be made to the agency whose order is under review. The agency has already decided the merits. The requirements of Fed. R. App. P. 18 make sense only if in appropriate cases the other three factors can justify a stay by the very agency that issued the order, without having to persuade the agency to change its decision. See Ruiz v. Estelle, 650 F.2d 555, 565 (5th Cir. 1981).

The probability of success on appeal is but one factor, and can be understood best not as a mathematical prediction, but as a question of whether the status quo should be maintained pending a decision on the merits. In other words, the Commission need not be persuaded that it erred, but may exercise its discretion to grant a stay if it finds that the appeal presents a serious legal question and the facts tend to show that the status quo should be maintained in the interim. See Washington Metropolitan Area Transit Commission v. Holiday Tours, Inc., 559 F.2d 841, 843 (D.C. Cir. 1977). This latter consideration can be determined by an analysis of the remaining three factors--harm to the moving party, harm to opposing parties, and the effect on the public interest.

With this in mind, CFUR will not reargue its petition to intervene or its briefs to the court of appeals. However, the Commission should consider the serious legal questions raised in the appeal. CFUR believes that it has shown that the Commission misapplied the standards of 10 C.F.R. § 2.714, which govern intervention. More specifically, CFUR challenges the application by the Commission of commission precedent and judicial precedent in determining what constitutes good cause for late filing of a petition to intervene. The briefs filed by CFUR in the court of appeals challenge a mechanical application of this formulation: "Long-standing and well-settled Commission precedent clearly holds that one party may not demonstrate 'good cause' for late intervention by attempting to substitute itself for another party which has withdrawn from the proceeding." CLI-88-12, pp. 4, 5. The application of that formula has become, *sub silentio*, an absolute rule that no intervention is allowed if one intervenor has withdrawn from the proceeding, regardless of the reason for the withdrawal. This is a serious legal question.

Furthermore, this case presents a unique question: will an applicant for a license be allowed to secure the dismissal of adjudicatory hearings, the withdrawal of an intervenor, and the silence of witnesses by paying large sums of money to the intervenor and the witnesses? CFUR in its petition to intervene could only argue this question by analogy to one settlement agreement it had--that between Mr. Macktal and Brown & Root. Since then another Comanche Peak settlement, between Mr. Polizzi and Gibbs & Hill, has come to light and was declared by the Secretary of Labor to be void as against public policy insofar as it restricted the flow of information about safety and regulatory matters known by Mr. Polizzi. Polizzi v. Gibbs & Hill, Inc., 87-ERA-38 (July 18, 1989).

CFUR has been told by parties to the agreement that the settlements with the whistleblower witnesses were conditioned on the withdrawal of CASE. This is very significant, and it is a new development in licensing proceeding practice.

Marshall Gilmore, a director of CASE whose wife was also a board member, represented Charles Atchison, a whistleblower, in his claims of retaliation by TU Electric in violation of the Energy Reorganization Act of 1974. Anthony Roisman and Billie Garde, attorneys for CASE, also represented individual whistleblowers in similar claims.

The attorneys for CASE and members of its board had a significant economic interest in settling the whistleblower claims. TU Electric conditioned the settlement of the individual claims on the dismissal of the hearings and the withdrawal of CASE. Under these circumstances continuation of the intervention would be very expensive for CASE's lawyers. When CASE withdrew and the hearings were dismissed, some of the whistleblower claims were settled. Mr. Roisman, Ms. Garde, and Mr. Gilmore received $1.5 million. As far as CFUR knows, the individual settlement agreements have not been reviewed by the NRC, and have not been made public.

It appears the settlement was not based on a resolution of safety issues; this is not the kind of settlement the NRC should allow. The combination of the unavailability to this date of the settlement agreements, the approval of the settlement by the presiding officer without examining the individual settlement agreements, and the conflict of interests created for CASE lawyers by TU Electric's offer to settle the individual claims only if CASE withdrew as an intervenor, raises a serious question of law: should the Commission consider these meretricious reasons for the withdrawal of CASE as an intervenor in determining whether CFUR has shown good cause for filing its petition to intervene late?

2. <u>Will irreparable injury occur if the stay is not granted?</u>

Before addressing this item, CFUR respectfully requests the Commission to reevaluate that part of its decision in <u>Public Service Company of New Hampshire</u>, (Seabrook Station, Units 1 and 2), CLI-89-8, 29 NRC 399 (1989), having to do with irreparable harm. <u>Id</u>., 409-412. First of all, that opinion states the untenable position that granting a low power license cannot, as a matter of law, cause irreparable harm. The opinion buttresses this extreme statement by incorrectly stating that a court of appeals reached the same conclusion in <u>Cuomo v. NRC</u>, 772 F.2d

972, 976 (D.C. Cir. 1985). The court in Cuomo stated: "Probability of success is inversely proportional to the degree of irreparable injury evidenced. A stay may be granted with either a high probability of success and some injury, or vice versa." Id., at 976. Two of the reasons found against the movants in Cuomo have no bearing here--a claim that the appealable issues would be moot if a stay were not granted, and a claim that the National Environmental Protection Act presumptively justified a stay. The Cuomo issue germane to CFUR's request is whether irradiation of the reactor and related risks can constitute irreparable harm. Far from saying these risks could never amount to irreparable harm in low level testing, the court in Cuomo weighed the allegations and found them wanting. Id.

If fuel is loaded in the Comanche Peak facility and low power generation of electricity is allowed, a threshold will have been crossed, from which we can never return regardless of the final outcome of the resolution of the safety issues still critical to this plant's safe operation. Nuclear fissioning will have occured, and nuclear waste will have been generated. The interior of the plant will be contaminated in a way that will change its character forever.

CFUR represents people whose health, safety, and livelihood will be harmed if there is an accident at the nuclear facility. Some members live within three miles of the plant, and the railroad line that would carry fuel into and nuclear waste out of the plant runs across the land on which they reside. An accident can occur during low power operation and the consequences would be severe to those near the plant. While the NRC may argue, with some justification, that large scale contamination cannot occur over a widespread area (into the Dallas-Fort Worth areas for example) from an accident during low power operation, that is simply not true for those in the immediate vicinity of the plant.

Further, if CFUR prevails and a license is denied, then the contamination of this plant with radioactive materials will make the plant unsuitable for use as a coal or gas fired plant. Plant workers will be exposed unnecessarily to radiation as the plant is cleaned up; the environment will be exposed to radioactivity it otherwise would be free of; waste will have been generated; and parts of the plant will be contaminated to such a degree that there will have to be removal of those parts to a safe burial site, which does not now exist. Where nuclear waste must remain on site, an accident can occur in an on-site waste storage area as well as in the reactor area, and the consequences can be more severe, according to a February 5, 1987 report titled "Beyond Design-Basis Accidents in Spent Fuel Pools (Generic Issue 82)," prepared for the NRC by the Brookhaven National Laboratory.

Recent developments are directly pertinent to safety problems. Check valves failed during testing in April and May, 1989. The failure was critical and, had the plant been operating with nuclear fuel, radioactive water would have travelled through pipes outside the containment vessel. Also, thousands of counterfeit bolts have been used throughout the plant during a ten year period. With respect to the check valves, an NRC report of July 10, 1989, said TU management's response to the issue was inadequate. The bolt issue is under investigation by the NRC Office of Inspector General and has not been resolved.

In June 1989, Shannon Phillips, a retired NRC inspector and former resident inspector at Comanche Peak, wrote a memorandum to the Commission stating that TU had misled the Commission about construction problems at Comanche Peak. He reported that TU exerted pressure on top NRC management to downgrade his findings in a 1988 inspection report that dealt with repairs made in 1988 to over 7,400 feet of service water piping in the piping system which provides cooling water to the plant's reactor systems. Phillips' memo included an internal TU memo which Phillips said showed a pattern of shoddy inspection techniques by TU.

On October 4, 1989, a group of NRC staff inspectors who had worked at CPSES for the past year informed the Commission that the pending SALP report "...is neither accurate nor complete...." They said factual information had been deliberately withheld, and the utility should receive a below average rating on its past year's performance, rather than a rating that it had met expectations. The group of inspectors stated that the plant is at least six months away from fuel loading.

In _State of Ohio ex rel., Celebrezze v. N.R.C._, 812 F.2d 288, 290 (6th Cir. 1987), the court of appeals said: "Though in this case the likelihood of a nuclear accident is concededly small, the potential severity is enormous." _Id._, 291. (In _Celebrezze_ a petition to intervene in licensing proceedings was denied, and the court of appeals stayed the issuance of a full power license pending review.

The harm to CFUR and its members is clear. The history of construction blunders and coverups at the plant between 1974 and 1986 are well known to the Commission. The facts set out above bring that history right up to this date, and make the safety of low power licensing extremely doubtful.

3. Will granting a stay harm other parties?

Harm to others is tested by substantiality, likelihood of occurrence, and adequacy of proof. Cuomo, supra, at 977. In measuring harm to others, "...mere economic loss does not constitute irreparable injury." Celebrezze, supra, at 291. It is clear from these cases that this factor weighs in CFUR's favor.

4. Where does the public interest lie?

It is probable that all parties to this case will claim the mantle of public interest. See Cuomo, supra, at 988. However, CFUR urges the Commission to adopt the view found in Celebrezze: "Though there is more than one public interest involved here, the most crucial concern is public safety." Id., at 292.

Conclusion

CFUR has adequately demonstrated the need for a stay, and requests the Commission to grant one.

Respectfully submitted,

Richard Lee Griffin

Richard Lee Griffin
Attorney for CFUR

CERTIFICATE OF SERVICE

I hereby certify that on this the 16th day of October, 1989, a true and correct copy of the foregoing "Request for Stay" was served upon the following named counsel: by personal delivery to Janice Moore, and by facsimile transmission to Thomas Schmutz and Dirk Snell, followed by first class United States mail, postage prepaid.

Janice E. Moore, Esquire
Office of the General Counsel
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Thomas A. Schmutz, Esquire
Newman & Holtzinger
Suite 1000
1615 L Street N.W.
Washington, D.C. 20036

Dirk D. Snell, Esquire
U.S. Department of Justice
P.O. Box 23795
L'Enfant Plaza Station
Washington, D.C. 20026

Richard Lee Griffin

Richard Lee Griffin

ADDENDA

8910180146

Declaration of Betty Brink

"On or about October 1, 1989, I was contacted by Dobie Hatley, a former Brown & Root employee at CPSES, who subsequently became a whistleblower witness for CASE in 1984. During the conversation, Mrs. Hatley told me that she was one of the whistleblower witnesses who received a portion of the $5.5 million. She said that she received about $450,000 and she understood that only seven of the whistleblowers, who had been or were scheduled to be witnesses for CASE, received settlements. Each of those seven had cases against TU or its contractors pending before the Department of Labor and were all represented by either Billie Garde, Anthony Roisman or Marshall Gilmore, Mrs. Hatley said. She said the three attorneys together received $1.5 million of the $5.5 million settlement money.

"On or about July 7, 1988, I spoke by telephone to Marshall Gilmore, attorney for CASE, board member of CASE, and attorney for CASE witness, Charles Atchison, who said that the two settlements, the one with CASE and the one with the whistleblower witnesses, were tied together and that both were tied to CASE's agreement to withdraw from the licensing hearings as an intervenor. At that time Mr. Gilmore did not tell me the amounts of money involved or the number of the whistleblowers who would benefit, but he did say that the concern for compensation for the whistleblowers was a major factor in CASE's agreement to settle. If they did not withdraw, Mr. Gilmore said, the whistleblowers would receive no monies.

"That same week I spoke to Billie Garde, attorney for CASE and some of the CASE witnesses, who told me the same thing, that is, that the two settlements were tied to the withdrawal of CASE as an intervenor and the closing of the proceedings."

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 15, 1989.

Betty Brink

Betty Brink

<u>Declaration of Lon Burnam</u>

"During the first week of July 1988, on or about July 6th or 7th, in separate telephone conversations, I spoke with both Billie Garde and Marshall Gilmore. In their individual attempts to persuade me of the necessity of the CASE settlement with TU, both insisted on confidentiality and both asserted that the only way that TU would settle with the whistleblowers is if CASE would settle and withdraw as an intervenor. Both said that the agreement had many provisions that would allow CASE to monitor safety concerns at the plant for a five year period, and both maintained that they felt CASE had no other option."

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 15, 1989.

Lon Burnam

Lon Burnam

Attachment B

~~INSERT #1~~

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'89 OCT 19 A11:58

COMMISSIONERS:

Kenneth M. Carr, Chairman
Thomas M. Roberts
Kenneth C. Rogers
James R. Curtiss

SERVED OCT 19 1989

In the Matter of TEXAS UTITLTIES ELECTRIC COMPANY, et al. (Comanche Peak Steam Electric Station, Units 1 and 2)	Docket Nos. 50-445-OL 50-445-CPA 50-446-OL

ORDER

This matter is before the Commission on a motion by the Citizens for Fair Utility Regulation ("CFUR"), asking that the Commission stay the issuance of a low-power license that it anticipates will be issued to Texas Utilities Electric Company ("TU Electric") in the near future, allowing it to operate Unit 1 of the Comanche Peak facility. For the reasons stated below, we summarily deny the request.

In its motion, CFUR asks that the Commission stay issuance of the anticipated low-power license pending judicial resolution of its petition before the U.S. Court of Appeals for the Fifth Circuit. CFUR's petition seeks review of our denial of CFUR's petition for late intervention in the Comanche Peak licensing proceeding. See Texas Utilities Electric Co. (Comanche Peak Steam Electric Station, Units 1 and 2), CLI-88-12, 28 NRC 605 (1988), as modified by Texas Utilities Electric Co. (Comanche Peak Steam Electric Station), CLI-89-06, 29 NRC 348 (1989). See Citizens for

8910250067

Fair Utility Regulation v. NRC, Case Nos. 89-4124 and 89-4310 (5th Cir. filed Feb. 16, 1989).

However, in this particular case, we believe that the Commission is not the appropriate body to determine this request. The Commission's stay procedures are primarily intended for use in staying the effectiveness of orders of the Atomic Safety and Licensing Board, the Atomic Safety and Licensing Appeal Board, or the Staff pending further internal review within the Commission. Here, the Commission itself has issued a final order denying CFUR's petition for late intervention. Thus, the Court of Appeals is the appropriate body to determine whether preliminary relief should be granted in a judicial proceeding to review a Commission order. Therefore, we deny the requested stay pending judicial review of the Commission's orders.

However, the Commission is the proper forum for requests for action based upon public health and safety concerns. If low-power operation of Comanche Peak presented an undue risk to public health and safety, we would not permit such operation, regardless of whether CFUR had petitioned for review of our order denying late intervention. In its pleading, CFUR asserts that there are possible safety hazards associated with the low-power operation of Comanche Peak. See Stay Motion at 7-8. CFUR also raises several specific technical concerns. See Stay Motion at 8-9. We hereby refer these matters to the Staff for appropriate resolution in accordance with the Commission's procedures for handling allegations. The Staff should also consider CFUR's allegations concerning the settlement agreement entered in the OL and CPA proceedings on July 13, 1988 and determine whether these allegations present any safety concerns that the staff has not previously considered. See Motion for Stay at 4-6.

We instruct the Staff to address CFUR's safety concerns prior to issuing the low-power license. Therefore, we see no need for the Commission to consider a stay of any anticipated low-power license at this time. According to our best information, TU Electric will not be ready to ask for a low-power license before November 9, 1989, more than three weeks hence.

For the foregoing reasons, the request for a stay of the anticipated low-power license is denied.

It is so ORDERED

For the Commission*

SAMUEL J. CHILK
Secretary of the Commission

Dated at Rockville, Maryland
this 19th day of October, 1989

---

*Commissioner Rogers was unavailable to participate on this order.

Attachment C

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

JUL 10 1989

In Reply Refer To:
Dockets: 50-445/89-30
50-446/89-30

Mr. W. J. Cahill, Jr.
Executive Vice President
TU Electric
400 North Olive Street, Lock Box 81
Dallas, Texas 75201

Dear Mr. Cahill:

This refers to the inspection conducted by Mr. H. Livermore and other members of the Augmented Inspection Team during the period May 15 through June 16, 1989, concerning the check valve failures which allowed backflow through the auxiliary feedwater system during hot functional testing of Unit 1 at the Comanche Peak Steam Electric Station. The team's findings as described in this report were presented to you and other members of your staff at the conclusion of the inspection.

The enclosed copy of our AIT inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

As a result of this inspection, the AIT has identified a number of weaknesses in your procedures for evaluating and correcting equipment failures and malfunctions, and weaknesses in your organizational communications. Further, while your subsequent assessment of the check valve failures has been comprehensive, the AIT has identified a number of recommendations which should be addressed in your corrective action efforts. Accordingly, we request that you submit a report summarizing the lessons learned from these events and the corrective actions you plan to take, concurrently addressing the weaknesses and recommendations identified by the AIT. This report should also distinguish between those actions which need to be completed before the plant is ready to load fuel and the longer-term programmatic enhancements. Please notify us, within two weeks following your receipt of this letter, of your schedule for the submittal of such a report.

8907180459

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC Public Document Room.

Should you have any further questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

R F Warnick

R. F. Warnick, Assistant Director
for Inspection Programs
Comanche Peak Project Division
Office of Nuclear Reactor Regulation

Enclosure:
Inspection Report 50-445/89-30; 50-446/89-30

cc w/enclosure:
See next page

cc w/enclosure:
Roger D. Walker
Manager, Nuclear Licensing
TU Electric
Skyway Tower
400 North Olive Street, L.B. 81
Dallas, TX 75201

Juanita Ellis
President - CASE
1426 South Polk Street
Dallas, TX 75224

Susan M. Theisen
Assistant Attorney General
Environmental Protection Division
P.O. Box 12548, Capitol Station
Austin, TX 78711-1548

GDS Associates, Inc.
1850 Parkway Place, Suite 720
Marietta, GA 30067-8237

Lanny A. Sinkin
Christic Institute
1324 N. Capitol Street
Washington, DC 20002

Ms. Billie Pirner Garde, Esq.
Garde Law Office
104 East Wisconsin Avenue
Appleton, WI 54911

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

William A. Burchette, Esq.
Counsel for Tex-La Electric
Cooperative of Texas
Heron, Burchette, Ruckert & Rothwell
1025 Thomas Jefferson St., NW
Washington, DC 20007

TU Electric
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, Maryland 20814

E. F. Ottney
P. O. Box 1777
Glen Rose, Texas 76043

Joseph F. Fulbright
Fulbright & Jaworski
1301 McKinney Street
Houston, Texas 77010

George A. Parker, Chairman
Public Utility Committee
Senior Citizens Alliance of
Tarrant County, Inc.
6048 Wonder Drive
Fort Worth, Texas 76133

Jack R. Newman, Esq.
Newman & Holtzinger, P.C.
Suite 1000
1615 L. Street N.W.
Washington, D.C. 20036

TABLE OF CONTENTS

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

NRC AIT Inspection Report: 50-445/89-30 Permits: CPPR-126
50-446/89-30 CPPR-127

Dockets: 50-445 Category: A2
50-446

Construction Permit
Expiration Dates:
Unit 1: August 1, 1991
Unit 2: August 1, 1992

Applicant: TU Electric
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Facility Name: Comanche Peak Steam Electric Station (CPSES), Units 1 & 2

Inspection At: Comanche Peak Site, Glen Rose, Texas

Inspection Conducted: May 15 through June 16, 1989

Team Leader: H H Livermore 7-7-9
H. H. Livermore, Lead Senior Inspector Date

Team Members: S. D. Bitter, Resident Inspector, Operations
E. N. Fields, Electrical Engineer, NRR
R. M. Latta, Resident Inspector (Electrical), NRR
M. Malloy, Project Manager, NRR
J. N. Rajan, Mechanical Engineer, NRR
W. Richins, NRC Consultant (Parameter)
M. F. Runyan, Resident Inspector (Civil/Structural), NRR
P. Stanish, NRC Consultant (Parameter)

8907180460

Executive Summary

On April 23, 1989, a misalignment of the turbine driven auxiliary feedwater pump discharge valves during hot functional testing (HFT) in combination with multiple failures of Borg-Warner check valves induced a backflow of high temperature water from the steam generators through auxiliary feedwater (AFW) piping to the condensate storage tank. The backflow event occurred with the reactor coolant system at normal operating temperature and pressure (NOT/NOP, 557°F and 2235 psig) and lasted approximately 20 minutes. The resultant excessive heat caused paint on the AFW piping to discolor, blister, and flake although no visible piping damage was evident. Available AFW temperature indicators were off-scale during this event.

On May 5, 1989, while still at NOP/NOT, valves in the AFW system were again misaligned allowing an even more pronounced intrusion of high temperature water into the AFW system. During this event, backflow occurred intermittently for approximately two hours. Additional paint was discolored and blistered on the AFW piping and one pipe support was damaged by thermal expansion.

Leak testing and radiographic examination performed subsequent to these events identified that at least 10 Borg-Warner pressure seal swing check valves (3 and 4 inch) in the AFW supply lines and miniflow lines were stuck open. After approximately six weeks of investigations, the applicant determined the root cause to be improper adjustments of the vertical elevation of the bonnet-disc assembly combined with possible excessive axial play in the disc-arm assembly. The improper adjustments were primarily the result of inadequate installation instructions in the Borg-Warner O&M manual. The applicant's corrective action included a valve-specific bonnet elevation adjustment (for pressure seal bonnet check valves) and a verification that the axial play component is within a specified envelope (for both pressure seal and bolted bonnet check valves). All Borg-Warner check valves located in Unit 1 and Common areas will be physically examined/adjusted and retested for reverse flow prevention capability.

The applicant evaluated the piping and containment penetrations for possible damage. Several areas in the piping were apparently stressed beyond ASME Code allowables. No unacceptable conditions were identified for the penetrations.

There were three precursor events. A similar Borg-Warner check valve failure was identified in 1985 at Comanche Peak but not thoroughly addressed by the applicant. Subsequently, three Borg-Warner check valves in the turbine driven AFW supply lines to the steam generators were found to be leaking on April 5, 1989, prior to HFT. Proper evaluation and resolution of the leakage found on April 5, 1989, might have prevented the high temperature water intrusions on April 23 and May 5, 1989. In addition, a Borg-Warner check valve in an AFW miniflow line was found to be leaking on April 19, 1989, and was repaired prior to the April 23, 1989, event. The applicant initially concluded that the failure of this valve was an isolated event. There exists extensive and well documented industry experience with faulty Borg-Warner check valves.

The AIT determined that a lack of aggressiveness by operations management to thoroughly follow-up on the valve failures identified on April 5 and April 19, 1989, inadequate communications between operations personnel, and lack of adequate manpower for operating valves during the HFT contributed significantly to the AFW events. While the problem resolution effort by the applicant was protracted (approximately 6 weeks), the results were thorough and represent a basic commitment to corrective action.

1.0 General Background Information

Comanche Peak Steam Electric Station (CPSES) Units 1 and 2 are Westinghouse pressurized water reactors with steel-lined, reinforced concrete containments. The units are under construction approximately 40 miles southwest of Fort Worth, Texas.

An extensive corrective action effort to correct numerous design and quality of construction deficiencies has been underway at CPSES over the past several years. This program has resulted in a significant number of modifications to bring the plants into conformance with NRC requirements. For various reasons, in March 1988, the applicant temporarily suspended work on Unit 2 to concentrate resources on Unit 1 completion. The applicant currently plans to begin loading fuel in Unit 1 on October 2, 1989. Hot functional testing (HFT) on Unit 1 has recently been completed* and integrated leak rate testing is scheduled July 1, 1989.

The NRC has established a policy to provide for the timely, thorough, and systematic inspection of significant events at nuclear power plants. This includes the use of an Augmented Inspection Team (AIT) to determine the causes, conditions, and circumstances relevant to an event and to communicate its findings, safety concerns, and recommendations to NRC management. An AIT was formed on May 15, 1989, to review events which occurred during Unit 1 HFT on April 23 and May 5, 1989. Although AITs generally evaluate events which have occurred at operating nuclear power plants, NRC management determined that these events warranted a team inspection conducted in accordance with AIT procedures.

---

*Unit 1 previously underwent HFT in 1985.

1.1 Description of the Events

Toward the end of Unit 1 HFT on April 23, 1989, levels suddenly decreased in Steam Generators (SGs) 1, 2, and 4 while all four SGs were being fed by Motor-Driven Auxiliary Feedwater Pump (MDAFWP) 02. The Turbine-Driven Auxiliary Feedwater Pump (TDAFWP) supply lines to SGs 1, 2, and 4 overheated, as evidenced by paint blistering and cracking on the pipes. The event was caused, in part, by concurrent opening of the TDAFWP test line isolation valve (1AF-042) and TDAFWP discharge valve (1AF-041). When both of these valves were opened simultaneously, a flow path to the Condensate Storage Tank (CST) was created from the SGs via TDAFWP piping (See Figure 1).

On May 5, 1989, a similar event resulted in the blowdown of steam generators Nos. 1 and 3 to the CST. On this occasion the MDAFWP test line isolation valve (1AF-055) and the MDAFWP discharge valve (1AF-054) were operated concurrently, creating a flow path through MDAFW and TDAFW piping to the CST. The second event was compounded after an attempt to close valve 1AF-055 resulted in this valve being left one-quarter turn open, which resulted in an additional blowdown from steam generators Nos. 1 and 3 to the CST through MDAFW piping. A diagram showing the feedwater system interface with the auxiliary feedwater system, and the backflow path is provided in Figures 2 and 3. The primary concerns with this event were (1) the equipment failures which could render the auxiliary feedwater system inoperable and (2) the temperature effects of the backflow on the auxiliary feedwater piping.

On May 15, 1989, the NRC Director Comanche Peak Project Division issued a Confirmation of Action Letter (CAL) to Texas Utilities. The letter confirmed that specified actions were to be taken by the applicant regarding the event of backleakage through the Borg-Warner check valves in the Auxiliary Feedwater System. The specified actions were subsequently completed by the applicant and the CAL was fulfilled as was noted in the AIT exit on June 16, 1989.

On May 19, 1989, TU Electric notified the NRC of a potential 50.55(e) construction deficiency relative to the AFW check valve backleakage events of April 23, 1989, and again on May 5, 1989. Additionally, the applicant informed Borg-Warner by letter TSC-89159 on June 1, 1989, that a defect, as defined in 10 CFR, Part 21, may exist within certain check valves supplied by them.

1.2 Augmented Inspection Team (AIT) Tasks

The AIT investigating the events was composed of a team leader from the NRC site inspection staff, three NRC resident inspectors assigned to Comanche Peak, the Comanche

Peak Project Manager from the Office of Nuclear Reactor Regulation (NRR), two technical specialists from NRR, and two NRC consultants assigned to the NRC Comanche Peak site inspection staff. AIT tasks were specified in a May 12, 1989, memorandum from the NRR Associate Director for Special Projects to the team leader. These tasks included:

a. Develop and validate a detailed sequence of events associated with the hot water intrusion into the Auxiliary Feedwater (AFW) System at Comanche Peak on April 23, 1989.

b. Evaluate the significance of the equipment failures with regard to safety system performance, safety significance, and plant proximity to safety limits as defined in the Technical Specifications.

c. Evaluate the accuracy, timeliness, and effectiveness with which the information on this event was reported to the NRC.

d. For each equipment malfunction, to the extent practical, determine:

   (1) Root cause.

   (2) If the equipment was known to be deficient prior to the event.

   (3) If equipment history would indicate that the equipment had either been historically unreliable or if maintenance or modifications had been recently performed.

   (4) Any equipment vendor involvement prior to or after the event.

   (5) Pre-event status of surveillance, testing, and/or preventive maintenance.

   (6) The extent to which the equipment was covered by existing corrective action programs and the implication of the failures with respect to program effectiveness.

e. Evaluate applicant's program for maintaining equipment operable after installation and initial testing/ inspection as it relates to this event. This should include surveillance testing and maintenance activities.

f. Evaluate the applicant's response to related experience and information, including NRC bulletins and notices and industry guidance provided in the INPO SOER on check valves and EPRI Application Guidelines.

g. Evaluate the applicant's thermal stress analysis of the piping affected by the hot water intrusion.

h. Evaluate the implications of the identified equipment failures during this event on other equipment in other safety systems at Comanche Peak.

i. Identify any human factors/procedural deficiencies related to the event.

j. Through operator and technician interviews, determine if any of the following played a significant role in each failure; plant material condition; the quality of maintenance; or the responsiveness of engineering to identified problems.

k. Evaluate operator action during the event.

l. Evaluate management involvement during the Unit 1 hot functional tests and the subsequent recovery from the event.

m. Evaluate the effectiveness of applicant's program for investigating events as it relates to the April 23, 1989 AFW intrusion event.

n. Evaluate the coordination of applicant's operations, engineering, maintenance, and other organizations in identifying and resolving the issues raised as a result of this event."

The primary focus of the AIT was on fact finding; any potential enforcement matters will be the subject of subsequent correspondence.

2.0 AIT Inspection

During the approximate six week period utilized by the applicant's AFW Task Team to address the resolution of this issue, the AIT team closely monitored the applicant's activities. This process typically involved the witnessing of valve disassembly, review of work controls and procedures, interviews with members of the applicant's staff, and attending selected meetings.

Efforts to reconstruct the precise timing of events during the incidents of April 23 and May 5, 1989, were difficult because the sequence-of-events computer was not in operation. The applicant was in the process of realigning the sequence-of-events computer to the emergency response system computer. The applicant utilized operator logs, strip chart recorders, and operator interviews to reconstruct the chronology of the individual events.

2.1 April 23, 1989, Event Description (PIR-89-110)

2.1.1 Conditions Preceding Event

On April 23, 1989, the applicant was nearing completion of an extensive hot functional testing program. The plant was in operational Mode 3 (hot standby) with the reactor coolant system at normal operating temperature and pressure (557°F and 2235 psig). The No. 2 motor-driven auxiliary feedwater (MDAFW) pump was running and feeding all four steam generators. Steam generator levels ranged from 56% to 59% with a feed rate of approximately 30 gpm per steam generator. The total steam generator blowdown rate was 45 gpm. The main feedwater isolation and main feedwater isolation bypass valves were closed and the preheater bypass isolation valves were open in each loop. A blackout start test of the turbine-driven auxiliary feedwater (TDAFW) pump had been completed at 0537 hours. The TDAFW pump was to be realigned to the condensate storage tank and run for three hours in preparation for a hot alignment check.

2.1.2 Event Chronology

At approximately 0610 hours, realignment of the TDAFW pump for recirculation flow to the condensate storage tank commenced. Standard Operating Procedure SOP-304A, Section 5.5.3, specifies closing TDAFW discharge valve 1AF-041 and then opening TDAFW test isolation valve 1AF-042 to perform this alignment. Contrary to this procedure, the two valves were operated concurrently. The auxiliary operator first cracked open 1AF-042 and then started to close 1AF-041. Three additional auxiliary operators were dispatched to provide assistance. Since valve 1AF-042 takes considerably less effort and time to open than is required to close 1AF-041, valve 1AF-042 was fully open before 1AF-041 was closed.

At approximately 0620 hours, the Reactor Operator noticed that levels in steam generators Nos. 1, 2, and 4 were decreasing rapidly. Temperature indicators 1-TI-2471 and 1-TI-2474 on AFW loops 1 and 4 were high off-scale (greater than 200°F) and 1-TI-2177B and 1-TI-2180B on feedwater (FW) loops 1 and 4 indicated approximately 500°F. The corresponding temperature indications on loops 2 and 3 remained unchanged at 105°F to 130°F. In an attempt to recover steam generator levels, the No. 2 MDAFW pump discharge flow was increased to 400 gpm. However, flow to steam generators 1, 2, and 4 indicated 0 gpm and steam generator levels continued to drop rapidly, approaching a level of 45%. Some flow was noted to steam generator No. 3 which indicated a slowly increasing level. The applicant stated that steam generator blowdown was secured on all steam generators at approximately 0625 hours. The AIT could not confirm this assertion as there is no indication of reduced outflow from the steam generators on the strip chart level recorders or any mention of this event in the operator's logs.

At approximately 0630 hours, the TDAFW pump room became steamy with a noticeable smell of paint fumes. The paint on some pipes in this room was observed to be "bubbling and peeling." Upon hearing this report, the control room ordered the auxiliary operator to shut valve 1AF-042. At 0635 hours, 1AF-042 was shut, and levels in steam generators Nos. 1, 2, and 4 began to recover from a low level in each of approximately 44%. The flow rate was increased to 50 gpm to each steam generator. Approximately two minutes later, loops 1 and 4 AFW temperature indications returned on scale.

A review of the event indicated that approximately 6000 gallons had drained from steam generators Nos. 1, 2, and 4 to the condensate storage tank (CST) through the TDAFW piping. Some increase in CST level was noted following the event. The applicant conjectured that an inadvertently closed motor-operated valve (1-HV-2493B) prevented blowdown of steam generator No. 3.

The backleakage of water from the steam generators to the CST through the TDAFW piping should have been prevented by the TDAFW supply line check valves. Based on the event scenario and subsequent testing, it is evident that these check valves were stuck open during the event. The other portions of the backflow path, from the steam generators to the TDAFW piping, could have taken one of four paths, as follows:

a. Through the two preheater bypass line check valves in the backflow direction.

b. Through the closed split-flow bypass valve and the outboard preheater bypass line check valve.

c. Through the closed feedwater isolation valve into the preheater bypass line.

d. Through the closed feedwater isolation bypass valve into the preheater bypass line.

Subsequent testing as described later in this report confirmed that steam generator water leaked back through the closed feedwater isolation bypass valve (d above, See Figure 1).

2.2 May 5, 1989, Event Description (PIR-89-129)

2.2.1 Conditions Preceding Event

On May 5, 1989, the applicant was performing the final portions of hot functional testing and was conducting a series of tests to determine which valves were responsible for the AFW backleakage event of April 23, 1989, (PIR-89-110). The plant was in operational Mode 3 (hot standby) with the reactor coolant system at normal operating temperature and pressure (557°F and 2235 psig). All AFW pumps were secured and the MDAFW cross-connect valves 1AF-090 and 1AF-091 were open. All AFW test discharge valves were closed. The main feedwater isolation and main feedwater isolation bypass valves were closed and the preheater bypass isolation valves were open in each loop.

2.2.2 Event Chronology

At 0055, preparations were initiated to perform a routine operational surveillance test, OPT-206A, "Auxiliary Feedwater System Operability Test." The purpose of performing this test was to provide precriticality training for operations personnel and to operationally check the surveillance procedure. The test was scheduled during HFT to take advantage of the then existing (hot) plant conditions. The No. 2 MDAFW pump discharge valve 1AF-054 was in the process of being closed at the same time that No. 2 MDAFW pump test valve 1AF-055 was being opened. This is contrary to procedure OPT-206A and SOP-304A (SOP-304A is referenced by OPT-206A) in that these procedures require 1AF-054 to be closed prior to opening 1AF-055. This mispositioning of valves was essentially identical to the April 23 event (paragraph 2.1.2). During the time both valves were open, a backleakage path similar to the April 23 event had been established from the steam generators through the leaking feedwater isolation bypass valves, through the preheater bypass line to the AFW inlet, into the AFW piping (See Figure 2). An analysis of steam generator level strip chart recorders revealed that backleakage occurred only from steam generators Nos. 1 and 3. Because steam generator No. 3 is located on the opposite end of containment from the feedwater penetration area, apparently no water from this steam generator entered the penetration area during this event. The flowpath from steam generator No. 1 was determined to be through TDAFW supply line check valve 1AF-078, into the TDAFW supply header, through TDAFW supply line check valve 1AF-106 (in the normal forward flow direction), through MDAFW supply line check valve 1AF-101, and through 1AF-054 and 1AF-055 to the CST (See Figure 3). The backleakage was stopped when valve 1AF-054 was fully closed, after which cross-connect valves 1AF-090 and 1AF-091 were closed.

At 0132, No. 2 MDAFW pump was started. After some data had been collected, this pump was secured at 0145. Steam generator levels had dropped due to steam-off and backleakage and the operator decided to realign the system to increase levels. At 0208, cross-connect valves 1AF-090 and 1AF-091 were opened, valve 1AF-055 was closed (but inadvertently left one-quarter turn open), and valve 1AF-054 was opened. This configuration reinitiated the backleakage predominantly through MDAFW supply line check valve 1AF-075, MDAFW cross-connect valves 1AF-090 and 1AF-091, and valves 1AF-054 and 1AF-055 to the CST. At 0230, No. 2 MDAFW pump was started, momentarily stopping backleakage from the steam generators. Although pump total flow indicated 300 gpm, the total flow to the steam generators was 80 gpm, indicating that 220 gpm from the No. 2 MDAFW pump was being diverted to the CST via valves 1AF-054 and 1AF-055. The operators did not know where the missing 220 gpm was going. They secured the No. 2 MDAFW pump at 0249. With all pumps secured, backleakage from the steam generator was again hydraulically permitted until, at 0251, No. 1 MDAFW pump was started. The same abnormal flow indications occurred, indicating that not all pump flow was reaching the steam generators. The No. 1 MDAFW pump was secured at 0305. This reinitiated the backleakage; however, within the next several minutes cross-connect valves 1AF-090 and 1AF-091 were closed, restricting the backleakage to steam generator No. 3. At 0323, No. 1 MDAFW pump was started in order to feed steam generators Nos. 1 and 2 and at 0326, No. 2 MDAFW pump was started in order to feed No. 3 and No. 4 steam generators. Normal flow conditions existed for No. 1 and No. 2 steam generators. However, a large flow mismatch was observed between No. 2 MDAFW pump flow and the flow to steam generators Nos. 3 and 4. Based on these indications, the operators at this time suspected that valve 1AF-055 was not fully closed. At 0340, valve 1AF-055 was found one-quarter turn open and when fully closed, ended the event.

During the approximately two hours of backflow, an estimated 3000 gallons blew down from steam generator No. 1 and a like amount from steam generator No. 3. Steam generators Nos. 2 and 4 were isolated. Based on the volume of piping from steam generator No. 3 to the feedwater penetration room, no water from steam generator No. 3 reached the AFW lines. The AIT notes that steam generator No. 1 is located in containment near the feedwater penetration room. Steam generator No. 3, on the other hand, is located on the opposite end of containment. Given the main feedwater piping volumne of Loop 3 (3261 gallons), there was insufficient backleakage from steam generator No. 3 to reach the main feedwater penetration.

2.3 Precursor Events

2.3.1 Historical Failure of Valves 1MS-142 and 1MS-143

A precursor to the April 23 and May 5, 1989, incidents occurred in 1983 when the auxiliary feedwater turbine driven pump steam supply line check valves (1MS-142 and 1MS-143) failed inspection following the first HFT. Test Deficiency Report (TDR) 1743, initiated in July 1983, described the disks to be eroded, bent, and unable to perform the designed function. The valves (along with similar Unit 2 valves 2MS-142 and 2MS-143) were returned to Borg-Warner where, on each valve, the stud was shortened and a stop extending below the bonnet was added. In addition, the face of the stop which contacts the stud was machined to a 20° angle to be perpendicular to the stud axis. This modification was performed per Design Change Authorization (DCA) 18917, and was apparently necessary due to the sudden high pressure differential applied to the valves when steam is released into the line.

The Unit 1 valves were again inspected on January 17, 1985, after five cold starts of the turbine driven auxiliary feedwater pump (TDAFWP). Valve 1MS-142 was found to have a damaged seat, cracked disk, and a cracked disk stud bushing. Problem Report (PR) 85-132 stated that the valve had apparently been assembled with the disk not properly aligned with the seat and contacting the bottom of the valve body.

Failure Analysis Report (FA) 85-001 was generated by maintenance engineering to address damaged valve 1MS-142. Revision 0 of FA 85-001 describes the cause of the failure: "The bonnet and retainer were incorrectly placed too low in the body, thus, preventing the disk from hitting the seat squarely. Construction procedures were followed. However, construction and operations procedures and the manufacturer's technical manual omit steps on setting the depth of the bonnet during reassembly."

The action to prevent recurrence stated in FA 85-001, Revision 0, was:

"All valves of the same type will be disassembled, inspected for damage, and properly reassembled. The procedures will be revised to include the correct method for reassembly."

FA 85-001, Revision 1, was later issued to revise the cause of the failure of valve 1MS-142 and the required action to prevent recurrence. The revised root cause of the failure was harsh flow conditions during the cold starts of the TDAFWP. The valve disk and stud were replaced and the valve seat was reconditioned. The revised actions to prevent recurrence were: (a) to replace or modify the valve or (b) to modify the system to prevent harsh flow conditions. Maintenance Engineering contacted Borg-Warner after issuing FA 85-001, Revision 0, and changed the cause of the failure after Borg-Warner confirmed that the failure was not due to incorrect installation and that the earlier modifications (DCA 18917) were apparently unsuccessful.

The two revisions of FA 85-001 were addressed in the engineering review section of PR 85-132. PR 85-132 states that test engineers involved in the cold starts of the TDAFWP did not observe any indications of water hammer and noted that valve 1MS-142 had indentations which indicated that the disk did not line up with the seat. PR 85-132 concluded that:

"Since the disk is not available for re-evaluation, the possibility that the failure resulted from incorrect installation cannot be totally dismissed. Nevertheless, since one or both of the valves have failed after each heatup, a design review of the valves and the system operating conditions is needed."

Investigation by the AIT revealed that the design review had been requested in TU Electric office memorandum TCF-85227 dated May 20. 1985.

The AIT has requested has additional information from the applicant regarding documentation of the 1985 discussions with Borg-Warner which led to the decision that the valves were correctly reinstalled. At the conlcusion of this inspection, no documentation had been provided.

The AIT also asked the applicant for information regarding the design review requested by memorandum TCF-85227. Design modification DM-85-273, "Turbine Driven Auxiliary Feedwater Steam Supply Line Modifications," dated January 29, 1986, describes hardware modifications and operational changes to the TDAFW steam supply lines to minimize the effects of water hammer. Apparently, no design review of the adequacy of the check valves was performed even though the design review was specifically requested by memorandum TCF-85227.

After review of the documentation provided to date and discussions with the applicant, the AIT concluded that: (1) incorrect valve reassembly was initially identified as the cause of check valve failure in 1985, (2) discussions with Borg-Warner convinced the applicant that the valve failure was due to other factors, and (3) no design review of the adequacy of the check valves was performed. Thus, in 1985, the applicant had identified the root cause of the check valve problem and had formulated corrective action plans which would have fully corrected the problem. The applicant apparently permitted the vendor to dissuade them from the correct course of action.

2.3.2 Check Valve Failures of April 5, 1989

A second precursor event occurred prior to heat up for Hot Functional Testing (HFT) activities on or about April 5, 1989, 18 days before the first AFW backleakage event. This second precursor event identified that three TDAFW supply line check valves were failing to seat properly. The discovery of this condition occurred during the process of draining and filling steam generators to resolve secondary chemistry problems. During a filling operation, water was observed flowing into the TDAFW pump. In addition, water was discovered on the floor in the TDAFW pump room. The source of the water was determined to be backleakage through check valve 1AF-106. Procedure ODA-408, log No. 1-89-035 was written primarily to forward flush the TDAFW supply lines to the steam generators with reactor makeup water. Additional steps were added to this procedure to determine if the check valves in the remaining three TDAFW supply lines were leaking. This leak test revealed that two other TDAFW supply line check valves, 1AF-078 and 1AF-086, were not seating properly. Work requests were written to repair the valves and were assigned a normal priority. The work requests, however, did not quantify the amount of valve leakage. Work orders were initiated with a due date of May 26, 1989, after completion of the HFT.

The AIT interviewed the operations manager concerning the decision made to continue the HFT with three failed AFW check valves. The operations manager stated that he reviewed in detail only the original of procedure ODA-408 log No. 1-89-035 and missed the fact that the issued procedure included check valve leak testing. The three work requests did not specify the quantity of water leakage, which was substantial, and were not thoroughly reviewed by the operations manager, the systems engineer, or the shift operators for AFW operability. The operations manager also stated that the main thrust of the HFT at this time was to chemically clean the system and that in hindsight, a Plant Identification Report (PIR) should have been issued to give immediate attention to the leaking check valves.

Clearly, poor communication among operations personnel and a lack of operability awareness was evident. Because the check valve failures were not documented on a higher-profile document, such as a PIR or NCR, and inasmuch as operations supervision failed to follow-up on the fact that the check valves were not seating properly, management-level attention was not focused on this multiple failure of check valves. This event provided the applicant an opportunity to discover the full extent of the problem and to avoid the backflow events of April 23 and May 5, 1989. The applicant did not discuss the failed check valves discovered on April 5, 1989, with the AIT until the week of June 1, 1989.

The applicant stated that this event will be used as a learning experience to effect a change in the mindset of plant personnel from a construction to an operations perspective. The operators in this case considered the check valve failures to be strictly a hardware issue and did not consider the effect of these failures on the operability of the auxiliary feedwater system.

2.3.3 Failure of Valve 1AF-069

A third precursor event occurred on April 19, 1989, when in the course of AFW pump testing and hot functional testing, the suction relief lifted on the "A" MDAFW pump. Subsequent investigation revealed that the miniflow check valve, 1AF-069, was experiencing gross backleakage. The valve was disassembled and inspected. The valve disk was found to have rubbed the inside of the valve body on both sides in the open position. A small flaw was found on the swing arm in the area of the pivot pin (1/8" wide, 1/8" deep). The damage appeared to be caused by excessive jarring occurring when the valve disk slammed against the stop upon opening and by turbulent flow conditions resulting from the upstream breakdown flow orifice. NCRs 89-4484 and 89-4632 were issued and the valve was reworked under Work Order C890005265. The indicated flaw was dispositioned "use-as-is," whereas the rubbing of the disk was dispositioned "repair."

Additional weld material was added to the end of the valve stop to prevent the valve disk from coming into contact with the back of the valve body (and possibly becoming lodged in the open position). The gap between the swing arm and the disk was reduced to limit the amount of axial play in the disk as an added measure to ensure the disk would not contact the valve body.

It is believed that valve 1AF-069, prior to being reworked, exhibited a stuck-open configuration (later found in the 4-inch AFW valves) with the top of disk under the lip of the seat. Subsequent backflow tests revealed that the rework effort was effective in stopping the backleakage. The reduction in the axial play of the disk raised the top of the disk enough to allow the disk to seat properly.

At the time of valve rework, the applicant believed the problem to be isolated to one valve which had excessive axial play. An investigation into root cause and generic implications may have presented the opportunity to discover the full extent of the check valve problems.

The proximity of the 3-inch miniflow check valves to the upstream orifice may have contributed to the failure of valve 1AF-069 by causing an increase in the axial play of the disk. In addition, the increased flow turbulence and valve tapping damage resulting from this configuration would greatly reduce the life span of this valve. The AIT recommends an design change, as soon as possible, to separate the 3-inch miniflow check valves from their associated orifices.

2.4 Equipment Performance and Analysis

2.4.1 Check Valves

2.4.1.1 Component Description

The following component descriptions are applicable to the events of April 23, 1989, and May 5, 1989, which involved multiple failures of check valves in the AFW system. All of the valves that failed were Borg-Warner 900 lb., pressure seal swing check valves. There are a total of 28 of these valves in each unit. The failed valves included, for Unit 1, two of the three 3-inch check valves, located on the AFW miniflow recirculation line, which were determined to be partially stuck open and all eight of the 4-inch check valves, located in the AFW discharge lines to the steam generators, which were also identified as being partially stuck open (i.e., the valve disk lodged under the seat ring). See Figures 4 and 5 for valve details.

In addition to the pressure seal check valves, the applicant utilizes 103 Borg-Warner bolted bonnet swing check valves in selected low pressure applications (i.e. 150 and 300 lb. systems). The bolted bonnet valves have, by design, a fixed vertical relationship between the bonnet/disc assembly and the seat ring such that subsequent to assembly at the manufacturer's facility the bonnet and disk assembly cannot normally be adjusted. Therefore, the bolted bonnet valves are not considered to be susceptible to the same failure mechanism experienced in the pressure seal valves.

Excessive axial play could, however, potentially result in degraded or inoperable check valves.

A design feature which is common to both the pressure seal valve and the bolted bonnet valve is the tolerance stack up in the disk arm bushing assembly referred to as the "axial tolerance or axial play." The axial tolerance was not historically regarded as a critical parameter by Borg-Warner. However, in order to assure that axial play would not affect the operability of the valve, Borg-Warner has committed to establish a maximum/minimum axial play acceptance criteria.

As part of the assessment of the AFW check valve inoperability issue, the following synopsis of check valve applications was provided by the applicant. A total of 160 Borg-Warner check valves were installed in Unit 1, Unit 2, and areas common to both units. Out of this total, 114 check valves are located in safety-related systems, including 16 4-inch AFW supply line check valves (8 in each unit and all 8 in Unit 1 were determined to leak), 6 3-inch AFW pump miniflow recirculation check valves (3 for each unit, 2 of 3 in Unit 1 were determined to leak), 2 8-inch TDAFWP discharge check valves (1 per unit, tested satisfactory in Unit 1), 4 6-inch MDAFWP discharge check valves (2 for each unit, both tested satisfactory in Unit 1), 2 8-inch TDAFWP suction check valves (1 per unit), 2 6-inch MDAFWP suction check valves (Unit 1 only), and 24 6-inch check valves located in the preheater bypass line to the upper feedwater penetration (12 per unit). Thus, out of the 114 Borg-Warner check valves located in safety-related systems, 56 are located in the area of interest defined by the backleakage event.

2.4.1.2 <u>Equipment History</u>

In order to evaluate the applicant's program for maintaining and ensuring the Borg-Warner check valves operable following installation and initial testing, the AIT reviewed the maintenance records for the pressure seal check valves. This review included the examination of construction operation travelers, nonconformance reports, startup work authorization forms, maintenance action requests, work orders, and NIS-2 forms.

This review revealed that the AFW check valves had been installed in the 1979-1980 time frame and that all of the check valves were disassembled and inspected in 1983 for the presence of full fillet welds on the disk to the disk stud and on the disk stud to the stud retaining nut. A change from the original specification of tack welds to full fillet welds was recommended by the vendor as a result of a valve failure.

In January 1983, while disassembling the containment spray heat exchanger, the disengaged parts of an upstream Borg-Warner check valve were discovered. Valve failure was determined to be due to a broken tack weld which had previously secured the disk to the stud. Tack welds were also used to secure the stud to the disk nut. Other defective tack welds were found in similar valves. Consultations with the vendor revealed that the problematic tack welds had been replaced with fillet welds as the standard valve design. The applicant decided to disassemble and inspect all Borg-Warner check valves, even those which had been procured after the vendor's design change. Any tack welds found were replaced with full fillet welds by site welders. Approximately 50 percent of the check valves required the installation of full fillet welds. A vendor representative was present during this modification process and extensive QA and QC oversight was provided. However, no post-modification retests of the check valves were conducted. Since all the valves were disassembled and reassembled, the final status was left uncertain in light of the inadequate installation instructions provided in the vendor's O&M Manual. The vendors O&M manual was inadequate in that it did not provide any instructions for backing off the retainer ring for valve flapper and seat alignment. For some pressure seal bonnet check valves, this resulted in the full insertion of the retainer ring which had previously been backed off to adjust bonnet elevation. This rendered the valve inoperable because the disk was positioned too low with respect to the seat ring.

The AIT investigation also revealed that the Comanche Peak Review Team (CPRT) in Issue-Specific Action Plan (ISAP) VII.b.2 identified the population of all valves that had been disassembled and reassembled under the construction QA program. Included in the population were Borg-Warner supplied check valves that were disassembled in 1983. Borg-Warner valves (1AF-0075, 1AF-0098, and 1FW-0202) associated with the Unit 1 Auxiliary Feedwater System were included in the CPRT sample.

CPRT compiled an inspection package for each sampled valve. Each package was reviewed for any indications of incorrect valve reassembly including variances in internal component serial numbers. No such cases were found. Each accessible valve was then physically inspected to verify that the correct body and bonnet were installed. No deviations were identified by CPRT for any Borg-Warner valves selected in the sample. No Borg-Warner valves were disassembled by CPRT.

In 1985, the system underwent initial hot functional and preoperational testing. These programs did not detect any abnormal check valve backleakage or operational deficiency relative to the valve disk hanging up under the seat ring. In arriving at this conclusion, it is recognized that the procedures used for preoperational testing did not test these valves in the backflow direction.

It was determined by the AIT (based on interviews with operations personnel) that a thorough flushing of sections of the AFW system could not be accomplished utilizing the existing system drain valves. Therefore, over the years, the applicant often removed selected check valve internals to allow for increased flushing flow rates. The AIT requested clarification on this policy from members of the applicant's AFW Check Valve Task Team. This practice of removing check valve internals was also used numerous times historically as a means of draining the system in order to effect welding repairs.

The applicant informed the AIT that it was a routine policy at the site to remove check valve internals to enhance system flushing or draining. The AIT's concern is that the numerous failures of the AFW system's check valves to seat properly may be related to the applicant's "routine" practice of removing check valve internals for the purpose of flushing and draining. The valves were not designed for routine disassembly. The lack of sufficient documentation following the completion of their maintenance activities appears to be historical.

The AIT also determined based on reviews of maintenance histories and discussions with both startup and system engineering personnel that no provisions were made for surveillance testing or maintenance preservation during the period from completion of preoperational testing in 1985 until the recently completed hot functional tests.

2.4.1.3 <u>Check Valve Investigative Action</u>

<u>AFW Check Valve Testing Subsequent to April 23, 1989</u>

The AIT witnessed the implementation of backleakage tests conducted on the AFW check valves subsequent to the April 23, 1989 event. The purpose of these tests was to determine if the check valves allow backflow past the seats. The valves tested included: (1) the eight AFW supply line check valves, (2) the three AFW pump discharge check valves, and (3) the two motor driven AFW pump miniflow check valves. The turbine driven AFW pump miniflow check valve could not be isolated and tested due to the design of the system.

The tests required unique valve alignments for each check valve The alignments isolated each valve and provided backflow pressures ranging from approximately 22 psig to 95 psig depending on the test procedure. A drain valve was opened to insure that the presence of flow could be detected should a check valve leak. Minimum hold times, generally 15 minutes, were specified.

An initial test of the AFW supply line check valves (8) was performed on May 2 and May 4, 1989, using steam generator pressure to create a backpressure of approximately 1150 psig. Additional tests were performed after HFT to provide assurance that similar tests, conducted after the valves were repaired and reassembled, would provide adequate assurance that the check valves were functioning properly.

All of the AFW supply line check valves, and all of the motor driven AFW pump miniflow check valves failed the tests and showed leakage. The three AFW pump discharge check valves did not leak.

As a result of these check valve failures, a total of 23check valves were radiographed (RT'd). The results of these RTs indicated that ten check valves were partially stuck open. Of these ten valves, eight were 4-inch valves and two were 3-inch valves. Additionally, the RTs for valves 1MS-142 and 1MS-143 indicated that the valve discs were contacting the seat ring at the top but that they were laying slightly off the seat ring at the bottom of the valve.

Following the identification of the inoperable check valves in the AFW system, the AIT inspectors witnessed the disassembly and inspection of selected Borg-Warner pressure seal swing check valves. During this process 14 check valves were disassembled. Valve disassemblies were conducted initially using Mechanical Maintenance Manual MMI-801, Revision 0, titled "Borg-Warner Check Valve Inspection." This procedure was later superseded by Maintenance Section - Mechanical Manual MSM-CO-8801, Revision 0, titled "Borg-Warner Check Valve Maintenance." These procedures appeared technically adequate for valve disassembly and the observed work activities were well controlled. During the disassembly process, various methods were utilized to capture information including the use of video recording equipment as well as boroscopic and radiographic processes.

Physical disassembly of the check valves was typically conducted in a well controlled and disciplined manner by the mechanical maintenance personnel. The AIT also determined that QC involvement appeared to be adequate and that QC hold points were correctly accomplished. The following is a synopsis of general observations by the witnessing AIT inspectors relative to the 3-inch and 4-inch pressure-seal swing-check valves manufactured by Borg-Warner.

- Some of the 4-inch check valve bonnets did not appear to be installed with the disk assembly parallel to the seat ring.

- The bonnet spacers on several of the check valves were deformed inward indicating overtorquing of the bonnet stud fasteners.

- Correspondingly, for the 4-inch valves that exhibited concave bonnet spacers, the studs were also deformed (bent) inward which also indicates overtorquing of the fasteners.

- Upon disassembly very little internal wear was observed on the disk seating surface and the seat ring was generally in a serviceable condition.

- For the 4-inch check valves identified as being stuck open there was some minor indication of disk contact on the seat ring in approximately the 12 o'clock position.

- For the one 6-inch check valve which was disassembled (1FW-198) the retainer ring was determined to be backed off approximately 0.150 inches.

- The bonnet assemblies were typically installed with an approximate .015 to .030 inch dimensional differential between the top of the bonnet retainer to the top of the bonnet (indicating that the bonnet fasteners were not tightened uniformly and sequentially).

- A variety of valve seat angles were encountered ranging from approximately 3° to 12° from the vertical.

- Axial play, although not dimensioned on the assembly drawing, was determined to range from 0.124 to 0.315 inches.

- Approximately half of the discs exhibited weld bead overlay remnants on the O.D. of the valve disk from the hardfacing process.

- Generally the hinge pins showed only minimum play.

- The disk stud on the 3-inch check valves associated with the miniflow lines indicated signs of deformation where it impacted the bonnet stop.

- On some of the disk assemblies the disk washer was loose.

Subsequent to the disassembly of the 14 AFW system check valves, the applicant performed detailed dimensional measurements of the valve bonnets and bodies to ensure their conformance to the manufacturers drawings. This review concluded that there were no dimensions outside of the manufacturing drawing tolerances with the exception of the wide variance of axial play dimensions. Axial play is not a specified dimension on the Borg-Warner assembly drawing.

On May 30, 1989, the applicant sent the internals from 13 check valves (consisting of 3 each 3-inch and 10 each 4-inch valve bonnet/disc assemblies) to Borg-Warner's Nuclear Valve Division in Los Angeles, California. The same valve internals were returned on June 14, 1989, after the vendor had performed dimensional checks and computer aided drawing (CAD) modeled verification of the as-built configuration. It is noted that while the subject valve internals were at the manufacturer's facility, no disassembly or destructive examination was performed. The assemblies were returned essentially in the as found condition.

The AIT determined that the programmatic controls and administrative procedures utilized for the identification, storage, packaging, and shipping of the subject valve internals to Borg-Warner for analysis were adequate.

See Figure 6 for summary of valve findings.

2.4.1.4 <u>Root Cause</u>

In order to assess the root cause determination, the AIT reviewed the BW/IP letter to TU Electric dated June 7, 1989, concerning Borg-Warner high pressure, swing check valves. Specifically, this BW/IP letter identified the cause of the identified failure of the 3 and 4-inch check valves to be inconsistencies between the supplier's valve assembly technique and the procedural guidance contained in Borg-Warner supplied Operation and Maintenance Manual. The AIT reviewed the applicant's maintenance procedures applicable to the 3 and 4-inch check valves, MMI-801, Revision 0. The prescribed reassembly technique was to install and bottom out the retainer which ultimately located the disk assembly low enough in the body to allow the disk to catch under the seat ring as shown in Figure 5. Other factors which were identified as contributors included axial play in the valve disc-arm assembly and the residual fillet weld at the juncture of the disk to disk stud.

Axial Play

Axial play is the total amount of movement within the disk arm socket in the axial direction. Physically it is a measurement of the distance between the inside of the disk stud washer to the back side of the disk minus the disk arm thickness at the stud bore axis. The axial play component was a consideration in the applicant's evaluation of the inoperable AFW check valves in that it contributes to the allowed dynamic interaction of the disk to the seat ring for both the pressure seal and bolted bonnet type Borg-Warner check valves. The relative significance of the axial play component was addressed in Borg-Warner's letter to TU Electric dated June 7, 1989, concerning Borg-Warner high pressure swing check valves. In part, this letter stated that historically the axial play was not considered to be a critical component. However, in order to assure that the axial play would not adversely affect the operability of the subject valves, Borg-Warner will establish a maximum/minimum dimensional acceptance criteria for this feature. This dimensional acceptance criteria had not been provided at the conclusion of the AIT inspection and will be evaluated later.

Bolted Bonnet Check Valve Issues

Concurrent with the AIT inspection efforts associated with Borg-Warner pressure seal check valve failures in the AFW system, two other similar but apparently unrelated incidents occurred involving Borg-Warner bolted bonnet check valves. The first event occurred on May 31, 1989, and involved a 4-inch 150 lb. check valve installed in the Service Water System (1SW-048). The valve exhibited excessive backleakage and was determined to have the disk separated from the swing arm at a point roughly parallel to the ball disk assembly. The failure mechanism and root cause for this valve, along with an investigation of known deficiencies associated with the corresponding valve on Unit 2, are currently being conducted by the applicant.

A second suspected check valve failure was reported on PIR 89-168, dated June 9, 1989, and involved the potential leakage of one or both of the 300 lb. check valves located in the discharge piping immediately downstream of the containment spray pump CP-1-01. The AIT witnessed a special test to determine the nature of the reported check valve deficiency. This test was conducted on June 15, 1989, under the auspices of nonstandard alignments and evaluations procedure 1-89-0072. This test essentially recreated the operational conditions of the containment spray system when the original pressure pulsation (check valve leakage) was identified. Test observations and procedure review conducted by the AIT indicated that the reported condition was apparently not the result of leaking check valves because the system operated correctly.

2.4.1.5 Corrective Action

2.4.1.5.1 Review of Retainer Ring Calculations

To assist in determining the cause of the backleakage, the applicant, based on information obtained by radiographs of several Borg-Warner valves, preliminarily concluded that the cause of the problem appeared to be that the valve disk was stuck in the open position due to interference with the internal valve seat. To confirm if this was indeed the cause of the backleakage, the applicant developed Computer Aided Design (CAD) models based on dimensions taken from the "as-installed" valves. This process was performed on several sizes of Borg-Warner check valves and these models confirmed the suspected cause of the problem (i.e. that the top of the valve disk was binding on the bottom of the upper portion of the valve seat). This condition was caused by the bonnet being set too low into the valve body. A secondary, minor contributor to this condition identified by the vendor representative, was the amount of axial play in the valve disk stud. This additional axial play could cause the top edge of the valve disk to sit even lower in the valve body thereby increasing the possibility of interference with the seat.

The applicant intends to restore check valve function principally by backing out the retainer ring attaching the bonnet to the valve body. This procedure will increase the shear stress acting on the individual threads of the retainer ring due to a reduction in the total shear area available. The AIT reviewed calculations prepared by the vendor (Borg-Warner Job No. 891-H-2984) concerning the minimum thread engagement required to ensure that the retainer ring can resist the shear stresses anticipated at the design pressure of the AFW system. From these results, a maximum retainer ring backout for each size valve was calculated, ranging from 0.25 inches for 4-inch valves to 0.678 inches for 8-inch valves. The applicant intends to set an administrative limit for retainer ring backout based on the calculated results. The applicant reviewed and concurred with the vendor calculations. Likewise, the AIT concluded that the calculations were acceptable and that they were based on conservative design input assumptions.

2.4.1.5.2 Corrective Action Plan

To resolve the backleakage concerns for the Borg-Warner check valves associated with the AFW system that were determined to be inoperable, the applicant issued nonconformance report (NCR)-89-6637. This document defines the measurements that are needed, the methodology to be followed to calculate the required "retainer backout," the additional rework required, and the actual retainer backout for the thirteen AFW check valves known to have been leaking. The AIT reviewed the methodology for determining the required retainer backout and concluded that the analytical technique was adequate. This NCR also includes written concurrence from Borg-Warner.

For the remaining Borg-Warner check valves in Unit 1, the applicant issued NCR-89-7476. This is an explanatory NCR which defines the dimensional data to be obtained in order to calculate the amount of "retainer backout" required to ensure proper function of the remaining Borg-Warner pressure seal check valves. This NCR also provides the direction necessary to determine if the axial play (amount of free movement of the swing arm relative to the bushing) in the Borg-Warner bolted-bonnet check valves is within allowable limits to ensure proper operation. Borg-Warner is to provide the applicable minimum and maximum value of axial play that will not affect proper operation of these valves.

These two NCRs will ensure that all Borg-Warner check valves in Unit 1 will be inspected prior to fuel load. The need for rework due to the exploratory NCR will be determined by engineering with all work committed to be complete as soon as practical prior to fuel load.

Rework for Unit 2 has not been scheduled to date.

2.4.1.5.3 <u>Post Modification Testing</u>

After the pressure seal check valves have been disassembled, measured, and reassembled with the proper amount of retainer backout as calculated by the method outlined in NCR-89-6637, the applicant intends to perform post-work testing. This testing consists of subjecting the valves to a fluid flow in the reverse direction and measuring the relative drop in downstream system pressure after opening the upstream drain valve to confirm that the corrective action was effective.

Testing for the bolted bonnet valves will be performed to a generic post-work test procedure and will test all valves that can be tested based on current plant conditions (i.e. existence of drain connections, etc.).

The applicant is in the process of developing a generic in-service test procedure.

2.4.2 <u>Feedwater Isolation Bypass Valves</u>

2.4.2.1 Valve Description and Design Function

The feedwater isolation bypass valves and the feedwater preheater bypass valves are 3-inch globe valves designed for feedwater system isolation and provide a portion of the pressure boundary of the steam generators. The valves use air to open and spring pressure to shut. This design allows for tight shutoff against the maximum postulated inlet pressure. The valves are used during startup and shutdown of the plant and are closed during plant operations. The valves receive automatic signals to close within five seconds to isolate feedwater from the steam generators. They are designed with the capability to isolate against the containment design maximum pressure of 50 psig with minimal leakage. Backpressure greater than 50 psig opens the valve against spring pressure.

2.4.2.2 Plant Backleakage Simulation and Valve Leak Tests

The AIT witnessed a test entitled "AFW Backleakage Event Simulation Under Controlled Conditions" (ODA-408A, 1-89-049, Section 5.7) conducted May 7-8, 1989. This test simulated plant conditions existing at the time of the AFW backleakage event of April 23, 1989, and was designed to determine the leak flow path and leak rate associated with that event. One motor driven AFW pump was lined up to supply 50 gpm to each steam generator and valve 1AF-042 (turbine driven AFW pump recirculation to CST) was opened. Then, separately for each loop, one valve in the preheater bypass line and two valves in the feedwater isolation bypass line were opened to simulate the plant line-up existing during the event (e.g., for loop 4, valves 1FW-0203, 1FW-0207, and 1FW-208 were opened). Backleakage was detected by monitoring temperatures of the upper and lower feedwater penetrations (MI-8 and MV-17 for Loop 4) and by measuring flow rate with a strap-on ultrasonic unit. The results were nearly identical for each loop, indicating that approximately 120 gpm leaked back through the spring-operated feedwater isolation bypass valve (valve 1-HV 2188 for Loop 4). Apparently no leakage occurred through the feedwater isolation valve in any of the loops because after the preheater bypass line valve (1FW-0203 for loop 4) was opened (with the feedwater isolation bypass line still isolated), no signs of any leakage were noted. Only after the feedwater isolation bypass line was unisolated (1FW-0207 and 1FW-0208 opened for loop 4) was leakage evident. This test, therefore, demonstrated that the backleakage experienced during the April 23 and May 5 events flowed through the feedwater isolation bypass line and the feedwater preheater bypass line to the AFW system. The differential pressure across the feedwater isolation bypass valve apparently overcame spring pressure, unseating this valve in each loop.

This valve is designed for 50 psi backpressure for containment isolation purposes. During the event, approximately 1000 psi backpressure lifted the seat against spring pressure, allowing a backleakage flow of approximately 120 gpm. A flow path, therefore, was created from the steam generators through the leaking turbine-driven AFW pump supply line check valves to the CST.

The four main feedwater isolation bypass valves were calibrated by the Instrumentation and Control group (I&C) on May 9, 1989. The valve set points were checked to verify that the valves were actually fully open or fully closed as indicated. All four valves were found to be satisfactory.

The AIT witnessed the implementation of a backleakage test of the eight main feedwater preheater bypass line check valves on May 7, 1989. The line configurations currently do not allow for the individual isolation of the two check valves in each of the four main feedwater lines. The two check valves were tested in series and only one nonleaking check valve was needed for satisfactory test results. The test was conducted in a manner similar to the tests of the AFW check valves described in Section 2.4.1.3 of this report. The test results were satisfactory leading to the conclusion that at least four of the eight valves held. All eight main feedwater preheater bypass line check valves have been or are scheduled for disassembly, repair, reassembly, and leak testing.

2.4.2.3 Applicant Intent and Corrective Action

The applicant informed the AIT of their intent to administratively isolate the feedwater isolation bypass valves during startup and shutdown conditions except when the valves are actually needed. This would be done by closing the manual block valves in the feedwater isolation bypass line. The applicant is also considering eliminating the currently installed interlock between the feedwater isolation bypass valves and the feedwater preheater bypass valves. This interlock forces one of these two valves to be open and the other closed at all times other than during a feedwater isolation signal (when both close).

2.4.3 Analysis of Auxiliary Feedwater Piping, Hangers, and Penetrations

2.4.3.1 Evaluation of Event Effect on Piping

Following the April 23 and May 5, 1989 events, significant discolorization of the protective coatings of the AFW supply lines for steam generators Nos. 1 and 4 was identified. This discolorization was most pronounced on the piping for

Loop 1. A significant amount of blistering and flaking of the paint occurred as a result of the higher than anticipated temperatures.

The piping is designed to ASME Section III Code Classes 2 and 3. The Class 2 pipe from the steam generator back to the first motor operated valves in the Safeguards Building on each loop was analyzed to a design temperature of 500°F and pressure of 1185 psi. The Class 3 portion from the pump discharge to the Class 2 portion, which also saw higher temperatures was designed for 150°F. Temperatures during the event could have been as high as SG temperature (557°F); therefore, the thermal portion of the piping analysis required, at a minimum, a review of stress levels to ensure that there were no excessive stresses induced by the significantly higher temperature. The design pressures for the feedwater and auxiliary feedwater are essentially the same; therefore, stress levels due to the 1185 psi water pressure are not a concern. Due to the higher temperatures, the pipe supports will need to be reviewed by the applicant for the effects of higher than anticipated thermal forces experienced during the backflow events.

The analysis of the piping associated with the reverse flow event in the auxiliary feedwater (AFW) system, is being addressed in two parts. The first is associated with the April 23 event. During this event, the fluid from the steam generators Nos. 1 and 4 (SGs) flowed toward the condensate storage tank (CST) via the discharge lines of the turbine driven auxiliary feedwater pumps (TDAFWP). The temperature assumption for this portion of the analysis was that the piping experienced SG temperature (557°F) from its connection to the feedwater system to the junction of turbine driven and motor driven AFW lines. From this point back to the header piping the temperature was assumed to be 325°F. The reduction in temperature at this point in the piping is based on the fact that the motor driven AFW line was running and circulating water toward the SG at a temperature of 100°F (approximately). From the header back to the CST the temperature assumption used is 200°F.

The second backleakage event resulted in a more severe condition from a thermal stress standpoint. In this event, SGs 2 and 4 were isolated from the AFW system and based on volume change in the SGs and capacity of the piping, the flow path for the reverse flow occurred in loop 1. The backleakage into the AFW piping was calculated to be approximately 3000 gallons which was sufficient to fill the affected pipes. Due to intermittent operation of the pumps, the amount of mixing of lower temperature fluid is indeterminate. Also, the amount of severely discolored pipe suggests higher temperatures. Accordingly, the temperatures used for the thermal analysis extend the higher temperature farther into the system. Specifically, SG temperature (557°F) past the junction of the turbine and motor driven pump discharge lines is considered to have travelled approximately 200 linear feet farther upstream. The temperature is then reduced to 400°F from this point back to the header piping. The temperature is then reduced incrementally to 200°F back to the CST.

For each of the two scenarios, there are portions of the pipe which are overstressed, and stresses were most severe for the second event. After the second event, it was noticed that support AF-1-096-023-S33R had failed. This support is located in the tunnel at the 810'-0" elevation just before the piping turns south toward the pump rooms. This support has been replaced in accordance with the disposition of NCR 89-6332, Revision 0. Also, the piping analysis shows that the location of the maximum thermal stress for both events is adjacent to this failed support.

As mentioned above, there are several areas in the piping which experienced thermal stresses higher than code allowables. These areas were identified by analyzing the piping using the higher temperatures outlined above. In determining the effect of these overstressed conditions, SWEC is performing the following steps. First, the allowable stress level was increased to agree with the one time allowable provided by the code; further, in determining this allowable stress, actual physical properties from the applicable certified material test reports (CMTRs) were used. Based on these values, only two areas of concern remain: (1) the elbow adjacent to the failed support and (2) some instrument connections. There exist additional conservatisms for the instrument connections which should eliminate these connections as areas of concern: first, the use of high stress intensification factors (SIFs) for the connections and second, ignoring the existence of gaps which will reduce the actual rigidity in the structural frames restraining the instrument lines. Evaluating these connections by more precisely modeled field conditions will result in greatly reduced stress values. For the elbow, even if the assumption is made that the failed support does not exist, a relatively high stress still would have existed. However, when the worst case analysis is considered in light of actual material behavior, a small amount of yielding would have occurred and then the stresses would be redistributed in the system with a minimal impact on the elbow itself. To ensure against any potentially adverse conditions, RT and UT were performed on the elbow to determine if any cracks exist. The results of this nondestructive examination did not disclose any cracks.

Therefore, it was concluded that replacement of the elbow was unnecessary. The AIT concurs with this assessment.

2.4.3.2 Evaluation of Event Effect on Pipe Supports/Restraints

On the AFW piping system, there are 563 supports, restraints, and anchors. To date there are load increases on 59 cases where the deadweight and thermal load due to this backflow event exceeded the design load used in the original calculations for the particular support. Additionally, there are several levels of review which will be followed to completely evaluate the need for rework. For example, if the transient load due to the backflow events is higher than the original design load, a review of remaining design margin will be conducted. At this point, the design margin relates to code allowables based on minimum expected material properties. The supports that exceed code allowable will be reviewed against a one-time allowable value. If necessary, the final determination of acceptability will be dependent on a full consideration of actual physical conditions (i.e. gaps to accommodate thermal expression, actual stiffness, actual material properties etc.). The AIT has reviewed the proposed method for resolution of the actual load increases and concurs with the approach presented.

2.4.3.3 Evaluation of the AFW Event Effect on Penetrations

The applicant evaluated the structural integrity of auxiliary feedwater cold penetrations (MV-17 to MV-20) subsequent to the April 23 and May 5, 1989, auxiliary feedwater backflow events. A preliminary analysis was performed conservatively assuming the penetrations experienced 550°F as no definitive indication of the temperature at the penetrations during either event was available. The actual maximum temperature experienced by these penetrations is thought by the applicant to be much lower than 550°F. The check valves inside the containment in the feedwater preheater piping did not leak and these penetrations apparently were not part of the backflow path.

The preliminary analysis was reviewed by the AIT and found to be very conservative in nature and to adequately address expected failure modes. The analysis included an evaluation of concrete bearing from the shear lugs, moment applied to the welding on the lugs, punching shear in the concrete, radial loads in the concrete, and pipe wall stresses. The analysis concluded that thermal expansion at 550°F of the pipe penetration should have caused spalling and/or crushing of the concrete.

Visual walkdowns of the penetrations by the applicant following the AFW events showed no evidence of any concrete distress or pipe movement. Hairline cracks typical of cracks observed following the structural acceptance test of the Unit 1 containment building were observed radiating outward from some of the penetrations. The applicant concluded that the penetrations did not experience temperatures as high as 550°F and that the penetrations were not adversely affected by the AFW events.

The AIT discussed the occurrence of the hairline cracks with the applicant and inspected the penetrations. The AIT concluded that the penetrations have not been damaged. It should also be noted that ASME B&PV Code, Section III, Division 2, Subsection CC, specifically CC-3430, stipulates that local areas of concrete (containments) are allowed to reach a temperature of 650°F for a short term period, where short term is defined as 24 hours or less (based on a code interpretation).

2.5 Personnel Actions/Human Factors

2.5.1 Operator Actions

The AFW events of April 23, 1989, and May 5, 1989, resulted from combinations of operator errors and equipment failures. In the first event, the auxiliary operator (AO) operated two valves out of sequence; i.e., he opened valve 1AF-042 prior to fully closing valve 1AF-041. This operational error, coupled with multiple check valve failures, resulted in an open flowpath backward from the steam generators (SGs) to the condensate storage tank (CST). A nearly identical out-of-sequence valve operation occurred during the second event (May 5). This time the operators opened valve 1AF-055 prior to fully closing valve 1AF-054. In both of these events, operator actions played a significant role. In Part 2 of the second back-flow event, however, operator actions figured in less significantly. Here, the inability of the operators to detect a less-than-fully-shut valve (due to extremely long, articulated valve/handwheel linkage) resulted in a similar backward flowpath being established from the SGs to the CST.

The first event was initiated while the operators were aligning the turbine-driven auxiliary feedwater (TDAFW) pump to recirculate to the CST. The applicable procedure, SOP-304A, "Auxiliary Feedwater System," Revision 5, clearly specifies that the TDAFW pump discharge isolation valve, 1AF-041, be closed prior to opening the TDAFW pump test isolation valve, 1AF-042. The reactor operator (RO) reviewed the procedure with the AO, ordered the AO to shut valve 1AF-041 and open 1AF-042, and dispatched the AO to accomplish the task. When the AO arrived in the TDAFW pump room, he noticed that the OPEN/CLOSE direction tag for valve 1AF-041 was missing; therefore, he was confused as to which direction to turn the handwheel to close the valve. In order to quickly determine the CLOSE direction for valve 1AF-041, the AO went to valve 1AF-042, similar in design to 1AF-041, and spun its handwheel in the OPEN direction while observing its gearbox. Doing so served two purposes: (1) by observing the gear motion on 1AF-042 while turning it in a known direction, the AO could determine which direction to turn 1AF-041 handwheel in order to close it and (2) 1AF-042 needed to be opened anyway; with multiple check valve protection, opening it slightly ahead of time should logically not cause any problem. After cracking valve 1AF-042 off its seat, the AO contacted the control room for assistance; he knew that closing 1AF-041 alone would require one half hour. Then, he began to close 1AF-041. When the extra AOs arrived, he directed them to open 1AF-042 and to relieve him in the task of closing 1AF-041. Just prior to his exiting the TDAFW pump room, the control room contacted him via radio and told him that steam generator levels were dropping rapidly. At this point, he went to the two motor-driven AFW pump rooms and verified that the test isolation valves for the two motor-driven AFW pumps were shut. After verifying the valves were shut, he reentered the TDAFW pump room and noticed what appeared to be steam. He then noticed that the paint on the AFW lines was blistering. At this point, the AO contacted the control room, apprised the control room personnel of the situation, was ordered to shut 1AF-042, and ensure that the TDAFW pump room was evacuated. In the meantime, the RO stopped SG blowdown. With 1AF-042 shut and blowdown secured, the event was over. Within 15 minutes of securing SG blowdown, SG levels had recovered to their normal levels.

In evaluating the operator actions for this first event, it is apparent that the AO violated procedure by not operating the valves in the correct sequence. Furthermore, it is clear that he did not appreciate the potential for check valve backleakage and its consequences. Finally, it is clear that the AO was under considerable pressure to complete the valve alignment by the end of the shift.

The AO is not the only operator who performed poorly. The RO, unit supervisor, and the shift supervisor share some of the responsibility for the poor performance. It should have been apparent to them that to send one AO to the TDAFW pump room to manipulate the two valves (1AF-041 and 1AF-042) at the end-of-shift was unreasonable. Control room personnel should have dispatched more than one AO or left the manipulation for the next shift.

The second event on May 5, 1989, occurred during the performance of the AFW system operability test. This test was being performed in accordance with Procedure OPR-206A, "Auxiliary Feedwater System Operability Test," Revision 2, as part of the surveillance test program. Basically, the test began with placing AFW pump 1-02 in recirculation to the CST per Procedure SOP-304A, "Auxiliary Feedwater System." Data was taken and the pump was secured. As in the first event, improper sequence of valve manipulation resulted in backflow from the steam generators to the CST. Because SG levels had decreased during the pump run, valves 1AF-090 and 1AF-091 (cross connect valves) were opened, valve 1AF-055 (test line isolation valve) was closed (but inadvertently left partly open), valve 1AF-054 (motor-driven AFW pump 1-02 discharge isolation valve) was opened, and AFW pump 1-02 was started. After starting the pump, the operators noticed that total pump flow was 300 gpm (abnormally high), but flow to the steam generators totaled only 80 gpm. Therefore, AFW pump 1-02 was secured. The operators then started AFW pump 1-01 and again observed that total pump flow was 300 gpm, again, abnormally high. Because SG levels were very low, the pump was allowed to run for ten minutes. After securing it, valves 1AF-090 and 1AF-091 were closed and AFW pump 1-01 was started to feed SGs 1 and 2. Several minutes later AFW pump 1-02 was started to feed SGs 3 and 4. The operators noticed that AFW pump 1-02 total flow was, again, 300 gpm. Therefore, suspecting backleakage on valve 1AF-055, an AO and the unit supervisor checked it to verify that it was shut. They discovered that they could turn the valve shut another one-quarter turn. Upon doing so, total flow for AFW pump 1-02 dropped to 80 gpm. Soon afterwards, an AO informed the control room that the AFW lines in the TDAFW pump room were hot and reported this information to the control room.

In evaluating Part 2 of the May 5, 1989 event of AFW backleakage, it is apparent that operator actions (errors) did not play as direct or significant a role as in the April 23, 1989 event or Part 1 of the second event. Instead, management and supervisory factors figure heavily into this event. Essentially, management was clearly aware of the events surrounding the first event. The question comes to mind: Why perform an auxiliary feedwater system operability test (OPT-206A) knowing that multiple check valve failures would not permit the system to operate as designed? Performing this test under such circumstances virtually ensured that more piping would be overheated - and it was. Furthermore, by the time of the second event, it was clear that the TDAFW pump discharge and test line isolation valves (1AF-041 and 1AF-042) were not "operator friendly." Should not the corresponding valves (1AF-054 and 1AF-055) be suspected of having the same, or similar characteristics?

2.5.2 Management Involvement/Oversight

Within hours of the April 23, 1989, event (which occurred towards the end of the graveyard shift), a Plant Incident Report (PIR-110) was initiated by the shift supervisor; personnel statements were obtained from him and the auxiliary operators involved in the event by the next day. The Operations manager met with the operating crew on April 24, 1989. A management meeting, attended by an NRC resident inspector, was also conducted on April 24, 1989. The latter resulted in the development of test plans to leak test check valves between the AFWPs and steam generators with the intent of identifying possible backflow leakage paths.

It was not until May 1, 1989, that the applicant established a task team with responsibility for investigation of all aspects of the AFW system check valve problem. The team was directed by the Operations manager and comprised of engineering representatives from Unit 1 Projects, Scheduling, Technical Support (Results), Performance and Test, Licensing, and Consolidated Engineering and Construction Organization (CECO). Within two weeks of the April 23, 1989, event, the Task Team had established an action plan for investigation of the problem, assessment of input on the affected system and equipment, identification of corrective actions, and determination of generic implications for other plant systems/equipment. The Task Team met daily for team members to report on the status and results of various action plan activities and to identify necessary additional actions. AIT members attended these meetings. The AIT found that the Task Team approach, provided a means for cordinating the various organizations in identifying and resolving the issues raised as a result of the events. Borg-Warner representatives were on site intermittently during activities concerned with assessing the check valve failure mechanism. Additionally, the applicant used an onsite engineering consultant, Kalsi Engineering, Inc., to provide advice and recommendations during the course of the investigation.

The AIT found the applicant's action plan to be comprehensive. The AIT observed that the applicant's preparations for execution of valve testing and disassembly were coordinated and conducted in an orderly fashion with an appropriate level of management oversight. However, it should be noted that more than six weeks passed after the initial event before the applicant arrived at a conclusion on the root cause of the valve failure and determined corrective action to be taken on the AFW system check valves. The results of the thermal stress analysis on the affected piping and the generic implication of the check valve failure for other plant systems and equipment have not yet been completed. While the slow pace at which action plan activities progressed contributed to the comprehensiveness of the evaluations completed to date, it was indicative of the lesser importance placed by applicant management on the events investigation as compared to other plant activities necessary for Unit 1 licensing. The AIT considered the events to have significant safety importance and expected that the applicant management would have caused the investigation to proceed in a more expeditious fashion.

In the course of its inspection, the AIT learned that at least some first-level supervision was aware that during a prior flushing operation (April 5, 1989), a number of check valves were found to be leaking. Despite this information, which appears was not adequately communicated to higher levels of management for evaluation, the applicant proceeded with HFT and subsequently experienced the subject events of concern.

The AIT also observed that the applicant's record retrieval capability was slow and this somewhat hampered the progress of several action plan activities.

2.5.3 Procedural/Human Factors Deficiencies

Personnel directly involved in both the April 23, 1989, and May 5, 1989, events were interviewed. The purpose of the interviews was to determine the extent to which any of the following played a significant role in each failure: plant material condition; the quality of maintenance; or the responsiveness of engineering to identified problems.

During the April 23 event, the first shift crew was preparing to perform a full-flow hot alignment test on the turbine driven AFW pump (TDAFWP). This was the last of several hot functional tests conducted during that shift. Based on comments obtained during the AIT's interviews, the crew had been very productive during the shift and there was an apparent press to complete preparation for this last test prior to shift turnover.

The balance of the plant reactor operators (TU) was directed by the unit supervisor to prepare the TDAFWP for full flow recirculation to the condensate storage tank (CST). The RO subsequently reviewed Procedure SOP-304A with the AO, which describes the steps necessary to operate a motor driven auxiliary feedwater pump or the TDAFWP for recirculation flow to the CST. The procedure directs the equipment operator to locally close the discharge valve (1AF-041) on the TDAFWP and to open the recirculation test line isolation valve (1AF-042). The order in which the valves are to be manipulated is explicit in the procedure. Responses to questions posed during AIT interviews indicate that the proper order of valve manipulation was not specifically emphasized during the review of the test procedure.

The AO entered the TDAFWP room where he "cracked 1AF-042 off the seat, approximately 1/4 turn . . . then proceeded to start closing 1AF-041." He requested that the RO provide him with assistance in performing his task. The RO dispatched three other personnel to the TDAFWP room to help in manipulating the valves. Two of the AOs responded immediately while the third was delayed for approximately 5 minutes. Although the three personnel indicated a general familiarity with the alignment being attempted, none had actually reviewed the procedure prior to entering the pump room. When the two operators entered the pump room, one was directed by the original AO to open valve 1AF-042 and the other was requested to manipulate the 1AF-041 valve. As a result of these actions, both valves were being manipulated concurrently rather than consecutively. This resulted in a flow path from the steam generators through the system's faulty check valves to the TDAFWP recirculation test line into the condensate storage tank (CST).

The first AO's rationale for conducting this procedure out of sequence derived from his familiarity with the system, his anxiousness to complete his task, and some minor frustration associated with the valve operators. The first AO's familiarity with the system led him to believe that he could rely on the system's check valves to prevent any back leakage which would possibly result from his operating the valves out of sequence. He also was aware that his shift was nearly over and he felt a need to expedite the completion of the alignment.

The frustration to which the AO referred was precipitated by the system's valve design, the valve's physical orientation, and the applicant's practices with respect to maintenance of valve packing. The 1AF-041 valve requires about 1000 turns of the handwheel to fully stroke while the 1AF-042 valve requires only about 60 turns. (The discharge valves on the motor driven pumps require about 460 turns to stroke.) A large electrical junction box is located in proximity to the 1AF-041 valve handwheel. This box presents an obstacle to AOs when they attempt to manipulate the valve handwheel often resulting in bumped and bruised knuckles. Also, the plant policy with respect to the installation and maintenance of valve packing is to tighten the packing to the point where "the valve stem squeaks." This results in difficulty in manipulating valves "particularly when the valve is backseated." The remote mechanical linkage by which the TDAFWP valves are manipulated necessitates rotating the hand wheel in the direction opposite that which would normally be expected if the valve were operated directly. Although there is normally indication affixed to the hand wheel to indicate proper direction of rotation, this indication was not present on the 1AF-042 valve. This required unnecessary distraction on the part of the AO in determining the proper direction of hand wheel rotation. This problem alone is not particularly significant; however, in concert with other existing conditions, this problem helped to exacerbate the actions of the AO.

On May 5, 1989, the second hot water intrusion event occurred at the plant. This event took place on the second shift. Part 1 of this event was essentially identical to the first event (April 23). Part 2 portion was initiated by equipment operability problems. A full flow test of the No. 2 motor driven AFW pump (MDAFWP) had been conducted in accordance with AFW operability test Procedure OPT-206A. This test entailed closing the No. 2 MDAFWP isolation valve 1AF-055 and running the pump to obtain various operability data. The test was performed satisfactorily. Because steam generator levels decreased during the test, the cross connect valves between the discharge headers of the two motor driven pumps were opened to allow the No. 2 pump to supply feed flow to all steam generators. Valve 1AF-054 was opened. An unsuccessful attempt was made to close valve 1AF-055; however, the fact that the valve remained partially open was not determined until the event was well underway.

The reason the valve was not fully closed is tied to the type of operator used to manipulate the valve. This valve has been characterized as "difficult to operate." The remote operator consists of a 15 foot reach rod connected to a 10 foot reach rod through a ninety degree universal joint. During the event, the valve handwheel had been fully rotated in the closed direction; however, the valve remained 1/4 turn open. Because of the physical configuration of the reach rod operator, either binding occurred in the universal joint connecting the rod sections or an excessive amount of the force applied by the AO in turning the hand wheel was expended in establishing torsional (twisting) forces in the rods. One or both of these conditions gave the AO the "feel" that the valve was seated. Because of the location of the valve relative to the handwheel, the AO was unable to visually determine the degree of closure of the valve.

With the opening of the No. 2 pump discharge valve, a low pressure flow path was established from the steam generators, through the faulty check valves, and through the open recirculation valve to the condensate storage tank.

This scenario was similar to the April 23 event and again resulted in the backflow of hot water into the AFW system. After a period of about two hours, the open recirculation valve was discovered. The unit supervisor and the AO, together, were able to turn the remote operator an additional 1/4 turn, fully seating the valve and terminating the event.

The AIT asked personnel involved in both the April 23 and May 5 events their impression of the general material condition of the plant and whether it played a role in either event. The consensus of opinion was that, not withstanding the severity of the check valve failures and the concerns regarding valve mechanical operators, the plant's response during the conduct of this hot functional test was better than anticipated, given the length of time the plant has been under construction. However, several of those interviewed indicated adverse personal experiences with remote valve operators.

There is a perception among those interviewed that the use of remote valve operators at the plant is abundant. (One person interviewed told that there was an "overuse" of these operators.) The design and placement of some of these operators appears to have been executed without proper regard to human factors issues. For example, the recirculation test line isolation valve on one motor driven AFW pump has a chain operator, while the equivalent valve on the other pump is manipulated with reach rods.

When asked if there had been any attempt on the part of any of the personnel interviewed to make their concerns known to appropriate management regarding perceived design or operational deficiencies, responses varied. Although there are formal procedures in place at the plant for requesting changes to system or equipment design, there was some uncertainty with respect to the appropriate vehicle to be used for a given change request. This may be attributed to the fact that unit construction is continuing and a concerted effort toward apprising plant personnel of the availability and proper use of formal plant procedures for requesting changes or reviews is yet to be instituted.

There appeared to be a consensus among those interviewed that management is currently more responsive than in prior years to personnel requests and suggestions. This altered management attitude is attributed to the nearly complete change in senior management which has occurred at the plant. The availability of hydraulically operated "man lifts," the construction of "catwalks," and the use of portable air operated wrenches for some "long winded" valves are changes which have resulted from increased management attention to the needs of operating personnel.

AIT personnel found that the necessary presence of construction equipment and materials; i.e., scaffolding, test instruments, etc., could present significant obstacles to personnel in their attempts to manipulate some equipment. However, this was not found to be a factor in the operating events currently under review.

All personnel interviewed were asked their opinions regarding whether engineering at the site was sufficiently responsive when design problems were identified. Individual views were mixed, but, generally, all felt that the change in management at the plant has had a beneficial impact on the quality and responsiveness of engineering personnel at the plant. However, because of the current phase of plant construction, it is often difficult to obtain adequate response to identified problems.

Each of the personnel interviewed was requested to provide an opinion regarding the quality of maintenance at the plant and his perception of its impact on the events under review. Again, most responded that the quality of maintenance has improved as a result of the management change that has occurred in recent years. However, there were two comments which were somewhat critical. The first comment questioned the policy at the plant of removing check valve internals in the AFW system to facilitate flushing. The second related to the scarcity of documentation associated with a completed maintenance procedure.

As was previously discussed in paragraph 2.4.1.2 with respect to the policy of removing check valve internals, it was stated to the AIT that a thorough flushing of sections of the AFW system could not be achieved with the existing system drain valves. Therefore, the applicant removed the appropriate check valve internals to allow for increased flushing flow rates. This was perceived as a possible system design flaw on the part of the person reporting. The AIT requested clarification on this policy from members of the applicant's AFW Check Valve Task Team. The applicant's team informed the AIT that it was a "routine" policy at the site to remove check valve internals to enhance system flushing. (The task team did not state that this policy existed to allow for back-flushing of the system.) The AIT's concern is that the numerous failures of AFW check valves to seat properly may be related to the applicant's routine practice of removing check valve internals for the purpose of flushing and draining. The valves were not designed to be routinely disassembled.

The concern regarding the lack of sufficient documentation following the completion of maintenance activities appears to be historical in nature. Apparently the maintenance policies in place prior to the major management change discussed above were derived from the practices at fossil plants. These practices tended not to be as sensitive to quality assurance requirements as one would expect from a nuclear based system. However, with the change in management, the person interviewed believes a change in the attitude regarding the importance of proper and complete documentation in support of maintenance activities is forthcoming.

2.6 Quality Assurance Considerations

The events of April 23 and May 5, 1989, represent in part the failure of the applicant's quality assurance program to detect the latent problems underlying these events and to provide corrective measures to prevent them from occurring. Quality assurance is most effective when events are prevented beforehand rather than as a reaction afterward in an effort to prevent recurrence.

However, the individual elements which combined to create these incidents, for the most part, transcended what is normally construed to be the responsibility of site quality assurance. The error in the vendor's technical manual regarding check valve installation was clearly the primary root cause for the backflow events. Only a highly detailed and somewhat fortunate vendor audit could have detected this problem. The secondary root cause was the failure of post-maintenance and post-modification testing to perform backleakage tests of the check valves. Although these tests would have been prudent and indicative of good engineering judgement, they were not procedurally required, due in part to the fact that the various applicable codes and standards emphasized only the forward-flow capabilities of check valves. In this perspective, again, the culpability of site quality assurance is minimal. Therefore, it can be stated that quality assurance failures played only a minor role in the two principal root causes for the events under investigation.

The report addresses several precursor events (see paragraph 2.3) which considered collectively should have led a reviewer to suspect that a generic check valve problem existed. It is here where quality assurance may have failed to notice the adverse trend. But the timing of these events is critical to the severity of this judgement. The failure of main steam valves 1MS-142 and 1MS-143 occurred in 1983 and 1985, but the failures of miniflow check valve 1AF-069 and the three check valves in the AFW system (1AF-106, 078,

086) all occurred within 2 1/2 weeks of the first backflow event. Very little time existed for a quality assurance trend evaluation.

Another quality-related issue that was instrumental in this event was the training of plant operators. Somewhere in this training, the essentials of in-sequence valve operation were not sufficiently emphasized. The applicant has committed to conduct additional training in this area. Another training-related issue was the failure of plant operators to document the discovery of three failed AFW check valves (discussed above) on an NCR or PIR and to recognize the resultant impact on the operability of the AFW system. The applicant recognized this failure, pointing out that the mindset of plant operators is still ingrained in construction. The applicant has committed to raising the awareness of plant operators to operational issues.

Another area where quality assurance may have gained insight into the check valve problem was the steam binding issue raised by I&E Bulletin 85-01. This bulletin suggested the possibility of AFW check valves allowing leakage by their seats in sufficient amounts to thermally bind a pump. The corrective action which resulted from the bulletin was to utilize AFW temperature sensors and to feel the pump discharge piping every shift to detect the presence of leakage. The NRC considered this commitment to sufficiently address the issues of the bulletin. Little can be said negatively of the applicant's actions on this issue except to suggest the possibility of the more proactive approach of physically testing a few valves to determine whether the problem currently existed.

In summary, the AIT team has concluded that the AFW check valve events do not suggest a major problem in the site quality assurance organization. These events do, however, point out weaknesses where programmatic enhancements would be prudent.

2.7 Applicant Evaluation

2.7.1 Evaluation of Applicant's Timeliness and Accuracy in Reporting the AFW Incidents to the NRC

In the first event, the NRC Senior Resident Inspector was notified promptly. While all the details were not yet apparent at the time of notification, it appears that the applicant reported this first AFW event in a timely and accurate manner. In contrast, the applicant was not nearly as timely in reporting the second event. Basically, various NRC inspectors learned at different times that "more pipe had overheated" during the performance of an operability surveillance test on the AFW system. In fact, the second event received so little attention that the Shift Test engineer for the AFW system was not aware of the newly overheated piping until several days after the event. Furthermore, a plant event report was not written until more than a week after the event and then only at the insistence of the AIT.

In summary, the applicant displayed an insensitivity to the seriousness of the second event. Apparently, the Operations department felt that running an AFW system operability surveillance test was a routine operational procedure even on a system in which multiple mechanical failures were evident.

2.7.2 Evaluation of the Implications on Other Equipment in Other Safety Systems at Comanche Peak

In light of the observed failures of eight 4-inch and three 3-inch Borg-Warner check valves in the Unit 1 Auxiliary Feedwater System, the question exists whether other Borg-Warner check valves located in other safety-related systems may have similar failures and thereby degrade the safety and reliability of the plant.

A total of 58 Borg-Warner check valves are located in safety-related systems other than auxiliary and main feedwater and are distributed as follows:

Component Cooling Water System

4 3-inch check valves (2 per unit), 150#
10 4-inch check valves (5 per Unit), 150#
2 8-inch check valves (1 per Unit), 150#
2 10-inch check valves (1 per Unit), 150#

Main Steam System (supply to TDAFWP)

4 4-inch check valves (2 per Unit), 900#

Containment Spray System

4 4-inch check valves (Unit 1 only), 300#
8 10-inch check valves (4 per Unit), 300#
12 16-inch check valves (6 per unit), 150-300#

Service Water System

4 4-inch check valves (2 per Unit), 150#
8 10-inch check valves (4 per Unit), 150#

The applicant has committed to physically examine, make necessary adjustments, and test the internals of each Borg-Warner check valve in Unit 1 and common prior to fuel load (Unit 2 is to be addressed at some later time). This effort should restore confidence that these check valves will perform as designed.

2.7.3 Applicant Action on EPRI Guidelines and INPO Significant Operating Experience Report SOER 86-03

As a result of several events involving check valve malfunctions, the NRC contacted the four NSSS Owners Groups in February of 1986, urging them to take a leadership role in addressing the design, testing, and maintenance of safety-related check valves. The Institute of Nuclear Power Operations (INPO) issued a Significant Operating Experience Report SOER 86-03, "Check Valve Failure or Degradation" dated October 1986, on this subject. In preparing the SOER, check valve failure data on 15,400 check valves included in Nuclear Plant Reliability Data System (NPRDS), Licensee Event Reports (LERs), and previous INPO publications were analyzed. In addition, check valve manufacturers and architect-engineers were contacted to identify the causes of check valve failures. Some broad recommendations to prevent check valve failures or degradation were provided in an EPRI report titled, "Evaluation of NUREG-1190 findings on the Adequacy of Check Valve Applications and Maintenance/Surveillance Practices." This report was developed to provide guidance to utilities in responding to SOER 86-03. Kalsi Engineering, Inc. is assisting TU Electric in developing and implementing a program based on SOER-86-03 recommendations.

It was initially decided that Kalsi would proceed with its evaluation on a system-by-system basis beginning with the Chemical Volume and Control System. After the check valve event, Kalsi was asked to shift their effort from the CVCS to the AFW system. Kalsi has completed their evaluation of the AFW check valves. Evaluation of the other systems (Main Steam, Service Water, Diesel Generator and Auxiliaries, Chemical and Volume Control, Safety Injection, Residual Heat Removal and Feedwater) is expected to be completed by June 30, 1989. The main objective of this program is to develop a preventative maintenance schedule and inspection procedure for each check valve located in the above-mentioned systems. Program priorities are based on many considerations, such as the consequence of valve failure, the location and orientation of the valve, the expected operational environment, and its maintenance history.

Review of the AFW check valves by Kalsi, Inc. is complete and a summary of their recommendations is as follows. All the check valves in the auxiliary feedwater system were reviewed. These are 3, 4, 6, and 8-inch Borg-Warner swing check valves. The 4-inch valves in the turbine and motor driven supply lines are located anywhere from 18 to 36 inches from 1-inch diameter flow limiting orifices which are treated as high turbulence sources.

Based on the design conditions specified in the FSAR, the flow through the 4-inch valves will be 286 gpm and typical usage would be less than 50 hours per year. Under these flow conditions, the disk is predicted to be oscillating at high levels. Because of the low usage, the calculated wear and fatigue indices are both very low and are considered acceptable. However, during the plant pre-start-up period the use of these valves is likely to last considerably longer and may be at significantly higher flow rates. Analyses performed at flow rates of 500 and 570 gpm for operation during Hot Functional Tests predict tapping and oscillating of the disk. The calculated Wear Index is acceptable because valve usage is very low during a 12-month plant cycle. Fatigue Index is, however, unacceptable due to high stresses developed when the disk is tapping. Kalsi therefore recommended that these higher velocities should be avoided. Kalsi also recommended during inspection of the 4-inch valves, the hinge pin, bushings, disk stud/hinge connection, disk and seat should be checked for wear and damage.

The 3-inch Borg-Warner swing check valves in the AFW pump miniflow lines are even more susceptible to the high flow turbulence. Based on analysis and review of recent backleakage problems with similar valves and inspection of 1AF-0069, inspection of each valve prior to plant start up is recommended by Kalsi in order to rectify the disk and seat alignment problems. During this inspection, the following areas should be checked for wear and damage:

a. Hinge pin and bushings.
b. Disc stud and stud-hinge connection.
c. Disc and seat.

Inspection of check valve 1AF-0069 revealed signs of considerable damage due to tapping contact with the disk stop such as a bent and peened disk stud and impact depressions on the disk stop. Kalsi has recommended design revision for the three 3-inch check valves located in the AFW pump miniflow lines. Kalsi states that if the situation is not corrected, these valves will suffer from exceptionally short life due to the high stresses developed during tapping. In the absence of quantitative assessments on how long these valves could operate without failure, it is recommended by Kalsi that corrective action be initiated immediately. In addition, higher flow rates (>500gpm) should be avoided due to fatigue related problems in the AFW motor driven pumps 1 and 2 and turbine driven pump supply lines.

2.7.4 Applicant Action on Other Site Failures and Generic Communications

The applicant performed a search through INPO and retrieved failures of Borg-Warner check valves at other sites listed in the Nuclear Plants Reliability Data System (NPRDS). A total of 38 failures of Borg-Warner check valves were retrieved. Of the total failures, 23 were identified as disk seating failures. Of these 23 failures, approximately 75 percent were reportedly caused by either foreign material caught between the disk and seat, disk distortion, improper installation of disc-stud-hinge arm assembly, or erosion/corrosion of valve internals. The remaining seating failures were attributed to normal wear or indeterminate causes.

Individual contacts were made with four plants identified through NPRDS to discuss their specific problems. No other plant experienced the exact disk binding found at CPSES although all expressed concern with the general quality of their Borg-Warner valves. One plant, St. Lucie, had to have the clevis of their 12-inch Borg-Warner check valves machined prior to shipment. They were told by Borg-Warner that this was a "one of a kind" fix and that future maintenance of these valves would have to consider the shorter clevis. It is unclear at this time whether or not this is significant to the CPSES incidents. Investigation into this item is continuing.

The McGuire Units 1 and 2 experienced full backflow through Borg-Warner pressure bonnet swing check valves under circumstances very similar to CPSES backflow events. The valves were replaced before a definite cause was determined. It is suspected; however, that the vertical positioning of the disk assembly caused the failure. Based on information obtained by the applicant from several plants (St. Lucie and Diablo Canyon) in regards to seal ring and pressure sealed valve applications, it appears that these plants have experienced other problems with Borg-Warner check valves such as bonnet leakage.

The applicant's actions in response to a number of IE Bulletins and Notices on related check valve failures were reviewed by the AIT. Some of those considered significant are summarized below:

- IEB 85-01, "Steam Binding of Auxiliary Feedwater Pumps." This IEB was issued because of reported events where hot water leaked into AFW systems and flashed to steam, disabling the AFW pumps. TU Electric letter TXX-4937 dated August 1, 1986, stated that work instructions for keeping a log for monitoring conditions leading to steam binding had been developed and implemented. Specifically, the procedure addressed equipment inspections, procedures for handling steam binding, and continued use of these methods until Generic Issue 93 was resolved. Licensee actions in response to this IEB were reviewed by the NRC and the IEB was closed by NRC Inspection Report 50-445/87-36; 50-446/87-27 dated February 10, 1988. The licensee had developed and implemented operating procedures and log keeping instructions to address the subject steam binding.

- IEB 83-03, "Check Valve Failures in Raw Water Cooling Systems of Diesel Generators." This IEB was issued after numerous licensee event reports (LERs) documented check valve failures. This IEB required operating licensees to review their plant pump and valve in-service test program per Section XI of ASME Subsection 1WV-3520 and modify, if necessary, to include check valves in cooling water flowpaths. It also required licensees to develop test procedures and conduct tests to verify valve integrity. The applicant's action in response to the IEB was reviewed and closed in NRC Inspection Report 50-445/88-12; 50-446/88-10 dated March 17, 1988. The file included the IEB and several other documents. The contact/inquiry record forms in the file documented evaluations of this issue. TU Electric concluded that Crane valve bodies were made from cast iron; however, stainless steel bodies were used at Comanche Peak. In addition, some valves in the cooling water flow path were identified which were to be tested quarterly per Procedure OPT-207A, Revision 1. (Service Water System Operability test). This procedure was developed to ensure compliance with technical specification requirements relative to valve position verification, valve exercising requirements of ASME, Section XI, Subsection 1WV-3522, and flow, pressure, and vibration measurement during pump start-up.

- IEN 80-16, "Shaft Seal Packing Causes Binding in Main Steam Swing Disc Check and Isolation Valves." (Closed 10/07/88).

  During disassembly of the main steam isolation valves at Indian Point 2, it was observed that all four reverse flow check valves were stuck at or near fully open.

During testing in the hot standby mode at the Trojan Nuclear Plant three of the four main steam isolation valves failed to close when manually actuated.

The cause of these events was excessively tight shaft packing. Although CPSES uses globe valves powered by compressed nitrogen accumulators, the concern of overtightening the shaft packing still affects the main steam isolation valve. This concern is addressed by MMI-818, Revision 0, "Rockwell MSIV Valve Repair," which has a caution to not exceed 75 foot-lbs. of torque for any reason when tightening the packing gland fasteners.

- IEN 80-41, "Failure of Swing Check Valve in the Decay Heat Removal System at Davis-Besse Unit No. 1," (closed 9/30/87).

During leak rate testing, an RHR pressure isolation check valve had excessive leakage. On disassembly, the valve disk and arm were found lodged under the valve cover plate. The valve is a swing check valve manufactured by Velan Valve Corporation.

- IEN 81-30, "Velan Swing Check Valves," (Closed 1/2/87).

Upon disassembly of a Velan 6-inch swing check valve at Salem 2, it was found that the valve disk stud had broken and the valve disk was on the bottom of the valve body. Cracks in the disk and bushings were found, along with a warped hinge pin and elongated hinge pin holes. Similar check valves at Point Beach 1 were found stuck open due to interference.

CPSES has Velan swing check valves, but not the specific models which failed at the plants described in this report. Although the specific failure mode is not applicable to CPSES, the general concerns of this report are addressed by SOER 86-03, "Check Valve Failures or Degradation." The SOER recommended that a check valve maintenance and inspection program be established as discussed in Section 2.7.3.

- IEN 81-35, "Check Valve Failures," (Closed 12/31/86).

Corrosion of the seat holddown devices caused loose internals in 3-inch 1500# Crane tilting disk check valves at Three Mile Island Unit 1. Failure of the hinge lugs on a 3-inch Series 900 Mission check valve at Fort Calhoun Unit 1 allowed the valve disk to migrate to the steam generator. Broken disk pins were found on 4-inch anchor darling swing checks at Arkansas Nuclear Unit 1.

Anchor Darling check valves are not used at CPSES. All Crane check valves at CPSES are rated at 900# or lower and are of the "swing" disk type, not the 1500# "tilting" disk type discussed in the report. Although CPSES uses missing duo-check valves, none are the 3-inch size discussed in the report. CPSES has two Velan swing check valves in the Boron Thermal Regeneration System. While the specific valve failures are not applicable to CPSES, the general concerns are addressed by SOER 86-03.

The applicant should insure that the log keeping instructions and operating procedures developed in response to IEB 86-01, "Steam Binding of Auxiliary Feedwater Pumps" as well as other procedures developed as a result of related IE Bulletins discussed above are included in the check valve maintenance and inspection program being established at CPSES.

2.8 Safety Significance of the Identified Check Valve Failures

A review of the FSAR, TS, design basis documents, and other pertinent material was conducted to determine the safety significance of the identified equipment failures and anomalies, listed as follows:

- All 4 4-inch MDAFW supply line check valves fail to prevent backleakage.
- All 4 4-inch TDAFW supply line check valves fail to prevent backleakage.
- Two or three of the three 3-inch AFW pump miniflow check valves fail to prevent backleakage.
- The feedwater isolation bypass valve allows backflow at a differential pressure of greater than 50 psid (this is in accordance with valve specifications, but may not be adequate for this application).

The result of this review, which included consideration of feedwater line breaks, steam generator tube ruptures, AFW piping ruptures, and main steam system breaks, concluded preliminarily that one credible accident occurring in conjunction with the as-found equipment failures could result in the plant exceeding its design basis. The postulated accident is a rupture of the MDAFW piping upstream of the supply line check valves resulting from an earthquake which also causes a loss of offsite power. Both main feed pumps are lost as a result of the loss of offsite power and the AFW system is automatically started to ensure that the steam generators can reliably remove decay heat from the reactor coolant system (RCS). A single active failure, the loss of one emergency diesel generator, is also postulated. The MDAFW pump associated with the failed diesel generator is lost and the other MDAFW pump is assumed to discharge its entire flow to the line break. This leaves only the TDAFW pump to supply the FSAR required flow of at least 215 gpm to at least two steam generators. The TDAFW pump is rated at 860 gpm and is ordinarily sufficient by itself to provide adequate flow to the steam generators. However in this case, the MDAFW supply line check valves would fail to isolate the upstream pipe rupture, allowing the TDAFW pump to feed the break. It is doubtful, given the as-found condition of the MDAFW supply line check valves, that a significant amount of flow from the TDAFW pump would reach the steam generators, and it is very likely that the design basis flow rate would not be achieved. A line break between the MDAFW supply line check valve and the upstream orifice would exacerbate the accident, since the orifice would not be available to limit the directed flow. If the design basis steam generator flow rate was not achieved, the decay heat entering the RCS would not be adequately removed in the steam generators. The RCS could overheat and overpressurize, causing the power-operated relief valve (PORV) and/or safety valves to open and release radioactive steam to the atmosphere. It is possible that this release of airborne radiation could exceed the limits of 10 CFR Part 20.

The multiple failure of check valves could have gone undetected as the plant entered the operations phase. Had this occurred, the plant would have been in a degraded condition and could have exceeded its design basis AFW flow as described above. It is unlikely, however, that either of the two Reactor Safety Limits would be challenged by this hypothetical accident. In the event of a loss of all AFW, with the steam generators boiling dry, the reactor coolant system could still be cooled by a procedure known as "feed and bleed." The power operated relief valve (PORV) is opened to the atmosphere (or vents are opened to containment) and the blowdown is compensated by normal RCS charging. This procedure should keep the average temperature (T-ave) below the approximate 660°F limit of the Reactor Core Safety Limit. The pressure-relieving capacity of the PORV should keep RCS pressure below the 2735 psig limit of the RCS Pressure Safety Limit.

Another potential failure mode of the AFW system is steam binding of the AFW pumps caused by backleakage through the inoperable check valves. Severe steam binding of AFW pumps could result in insufficient flow to the steam generators during emergency conditions. Prior to the AFW backleakage events, the applicant had committed in response to I&E Bulletin 85-01 to monitor AFW piping for backleakage every eight-hour shift. An operator will touch the discharge piping of the AFW pumps to detect any increase in temperature and the AFW temperature indicators will be monitored for any abnormal reading. Considering the amount of backflow necessary to cause significant steam binding, the applicant's method of detection appears adequate. It is noted, however, that this process of checking the temperature of the AFW discharge piping would not have detected the existence of numerous inoperable AFW system check valves in that with the absence of system flow to a low pressure point there would have been no thermodynamic migration.

2.9 Potential for Re-occurence

Discussions with the applicant indicate that in the future if engineering determines that a check valve has a safety-related function, there will be in-service and post-work functional testing which will include backleakage checks. This procedure will include requirements for QA surveillance. The periodic and post-work testing as described, should preclude the recurrence of similar incidents during plant operations. The formal procedure has not been issued.

2.10 Radiological Consequences

During the time frame of the event, there were no radiological consequences. The plant was at normal operating temperature and normal operating pressure, but no fuel was installed in the Reactor Vessel. Fuel load is currently scheduled by TU Electric for October 2, 1989.

3.0 Findings of Fact

Historical Observations

- A similar Borg-Warner check valve failure was identified in 1985 by Failure Analysis Report 85-001.

- Three Borg-Warner check valves in the TDAFW supply lines to the steam generators were found to be leaking on April 5, 1989.

- Proper evaluation and resolution of the April 5, 1989, events may have prevented the April 23 and May 5, 1989, events.

- Borg-Warner MDAFW pump miniflow check valve 1AF-069 leakage was identified April 19, 1989.

- Industry experience with faulty Borg-Waner check valves was well documented.

April 23, 1989 Event

- Misalignment of valves caused backflow of high temperature water through the AFW piping.
- Duration of event was approximately 20 minutes.
- The event caused the paint on AFW piping to discolor, blister and flake due to excessive heat.
- No visible damage to piping during this event.
- Temperature indicators off scale during this event.
- Backleakage flow path was through the feedwater isolation bypass valves. These valves are designed to resist 50 psi backpressure and, when tested, met this design criteria.

May 5, 1989 Event

- Backflow of high temperature water through the AFW piping due to improper valve alignment.
- Duration was approximately two hours.
- Intermittent pump operation during this event allowed higher piping temperatures to extend further upstream in the AFW system than during the April 23, 1989 event.
- One support visibly damaged by thermal expansion.
- Formal documentation of second event was not timely.

Root Causes and Effects of the AFW Events

- Leak Testing performed subsequent to the April 23, 1989 event identified several stuck open Borg-Warner check valves which allowed reverse flow.
- The cause of the stuck open AFW check valves was determined to be improper adjustments (vertical elevation) of the bonnet-disc assembly combined with possible excessive axial play in the disc-arm assembly.
- The improper vertical adjustment of the valve bonnet resulted from inadequate installation instructions in the vendor's O&M manual.

- A contributing cause of the 3-inch miniflow check valve inoperability may have been close proximity to an upstream breakdown orifice.

- Applicant's evaluation of piping indicates that several areas were stressed beyond ASME code allowable.

- Inspection of penetrations revealed no concrete distress.

4.0 Conclusions and Recommendations

4.1 Conclusions

4.1.1 The identification of three inoperable check valves in the TDAFW supply lines on April 5 should have been aggressively pursued. Instead, it was assigned a normal work request priority. This event reflects a lack of understanding of the system operability implications of failed components and a lack of aggressiveness of Operations management to follow-up on the results of the system flush they had specifically scheduled to determine the scope of the original identified check valve problem. This event was clearly a missed opportunity to discover the full extent of the check valve problem in time to prevent the April 23 and May 5 events from occurring.

4.1.2 The overall response by control room personnel to both events (falling steam generator levels) was weak (See paragraph 2.1.2).

4.1.3 Continuing to test the AFW system after the April 23, 1989 event with known multiple failures of check valves without taking appropriate precautions shows a potential lack of respect for degraded plant conditions. It also shows lack of communications between shifts.

4.1.4 It took an inordinately long period of time for Operations to adequately identify the second May 5 event and to report it as such, escpecially considering that it had a greater magnitude of severity than the April 23 event. The applicant's originally stated intent of including this event within the first PIR (110) appeared to be slow. In fact, PIR-89-129 was only written at the NRC's AIT insistence.

4.1.5 The out-of-sequence operation of valves in the May 5 event, occurring 12 days after a fundamentally identical out-of-sequence valve operation in the April 23 event, reflects a significant weakness in the applicant's ability to prevent an operational error from recurring.

4.1.6 Sending only one auxiliary operator near the end of shift to operate valves 1AF-041 and 1AF-042 reflects a lack of understanding in the control room regarding task manpower requirements.

4.1.7 The AIT considers the difficulty of operation of valves 1AF-041 and 1AF-054 to be a contributing cause to the April 23 and May 5 events, but of minor safety significance. The AIT supports the applicant's intent to make these valves easier to operate.

4.1.8 The evaluative process, which ultimately determined the root cause for the check valve failures appeared to be unnecessarily protracted in that it required almost six weeks from the inception of the AFW Task Team until the development of a definitive root cause and corrective action program. This protracted process, although not directly related to any regulatory requirement, is an example of the applicant's lack of management aggressiveness in the resolution of a safety-significant issue. This issue involved the multiple failures of passive components in a system intended to mitigate the consequences of an accident. For an NTOL plant, the applicant's response did not reflect the style of proactive Operations management philosophy normally associated with safe reactor plant operation. The AIT notes that when the applicant's Project Management took charge of the Task Team on May 26, 1989, efforts were significantly more timely and reflected a stronger commitment to corrective action. The applicant's Task Team went to the vendor Borg-Warner and made things happen. This aggressive attitude by management brought to light the root cause and brought about a corrective action plan in a timely manner.

4.2 Recommendations

4.2.1 Create a minimum equipment list that would aid Operations personnel to make judgements regarding the effect of failed components on system operability.

4.2.2 Assign system engineers the in-line task of reviewing all work requests related to a given system. The engineer would evaluate the impact of all component failures in regard to system operability.

4.2.3 Provide training to control room personnel and supervisors regarding manpower requirements for certain types of plant evolutions.

4.2.4 Provide continued emphasis on training plant personnel to comply with procedures. Steps are to be performed in sequence unless otherwise specifically approved.

4.2.5 Provide better communications between operations staff, especially during shift changes.

4.2.6 Provide a large and conspicuous plant status board in the control room, sufficient to provide significant "night order" information and to facilitate the transfer of information between shifts.

4.2.7 Initiate an immediate design revision to separate the 3-inch miniflow check valves from their associated orifices. The present configuration, if not corrected, lends itself to an exceptionally short lifespan for the check valves due to flow turbulence and valve tapping damage (see paragraph 2.3.3).

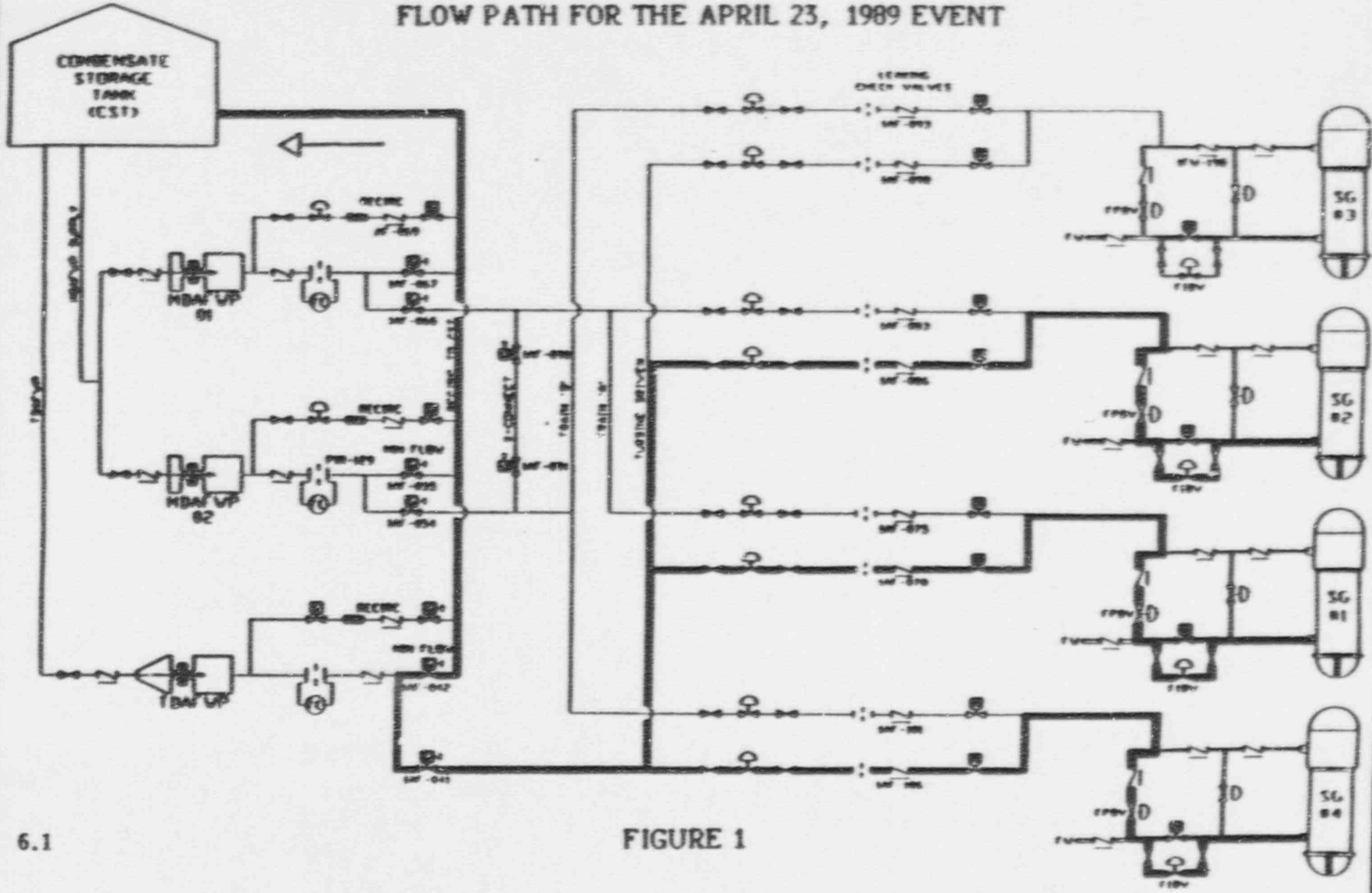
4.2.8 The AIT recommends that an Information Notice (IN) be issued in order that all licensees will be aware of necessary corrective action. The AIT has drafted an IN and submitted same to the NRC Generic Communications Branch on June 16, 1989.

5.0 Persons Contacted

K. Backus, Engineering Operations, TU Electric
M. Bagale, Assistant Project Completion Manager, TU Electric
R. Barr, Operations, QA Surveillance, TU Electric
C. Bishop, Reg. Adm., TU Electric
M. Blevins, NUC Operations Support, TU Electric
H. Bruner, Senior Vice President, TU Electric
W. Cahill, Executive Vice President, NEO, TU Electric
J. Donahue, Operations Manager, TU Electric
S. Ellis, Performance and Test Manager, TU Electric
B. Garde, CASE
W. Guldemond, Licensing, TU Electric
B. Hardison, NSSS System Completion Manager, TU Electric
T. Heatherly, Licensing Engineer, TU Electric
J. Hicks, Chief Engineer, TU Electric
C. Hogg, Chief Engineer, TU Electric
T. Hope, Licensing, TU Electric
J. Kelly, Manager of Plant Operations, TU Electric
D. McAfee, QA, TU Electric
C. Montgomery, Feedwater System Engineer, TU Electric
J. Muffett, Manager of Engineering, CECO
E. Ottney, CASE
S. Palmer, NEA, TU Electric
P. Pellette, Operations Technical Support, TU Electric
C. Rau, Projects Completion Manager, TU Electric
M. Samuel, Technical Interface
A. Scott, Vice President, Nuclear Operations, TU Electric
S. Shuman, Engineering Manager, CECO
J. Smith, TU Electric
R. Smith, Engineering Management, CECO

R. Smith, Operations, TU Electric
M. Street, Projects Scheduling, TU Electric
C. Terry, Projects, TU Electric
M. Thero, Citizens for Sound Energy (CASE)
O. Thero, CASE
G. Trieste, Projects Manager, TU Electric
J. Woods, Projects, TU Electric

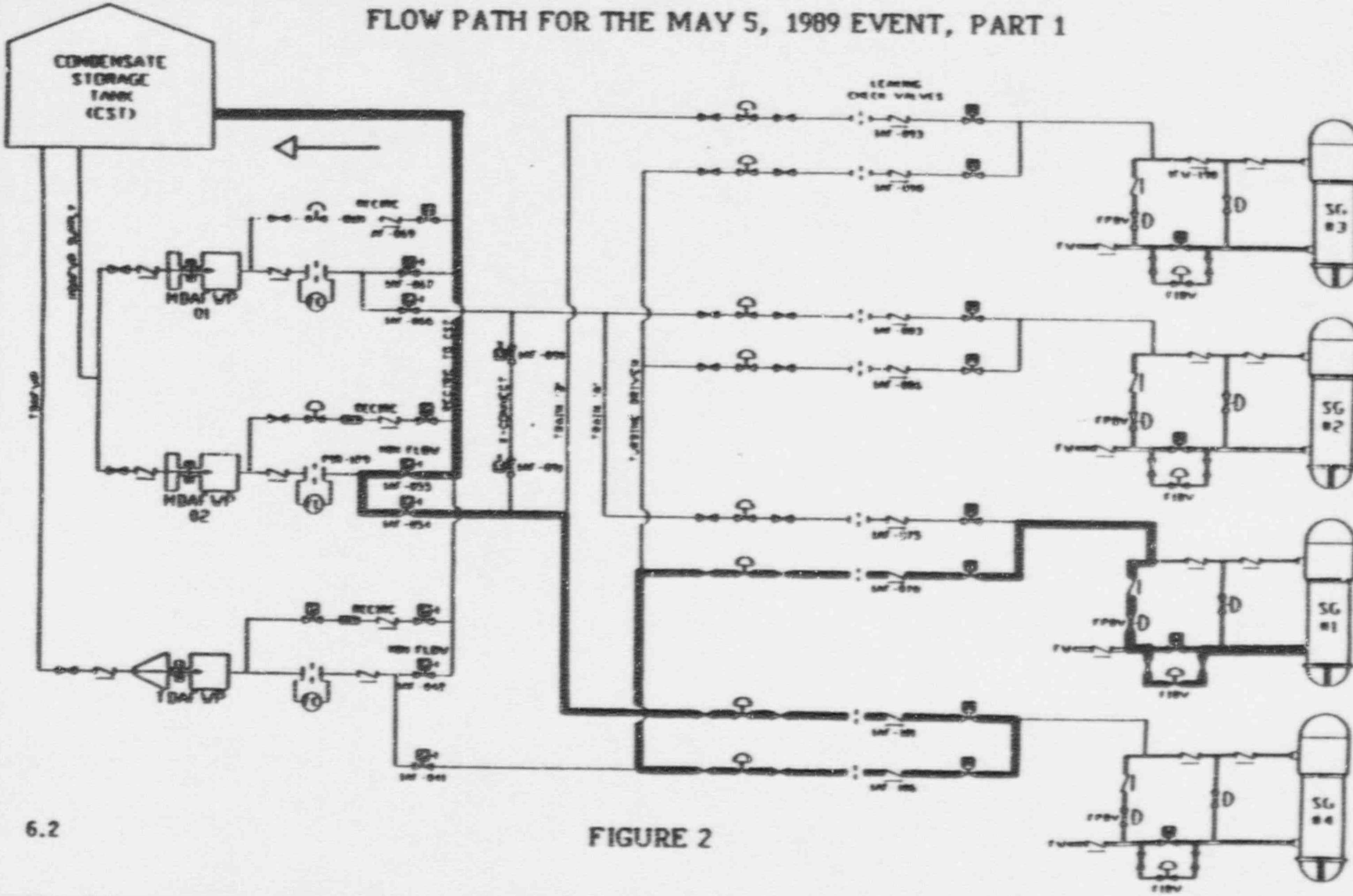
FLOW PATH FOR THE APRIL 23, 1989 EVENT



6.1

FIGURE 1

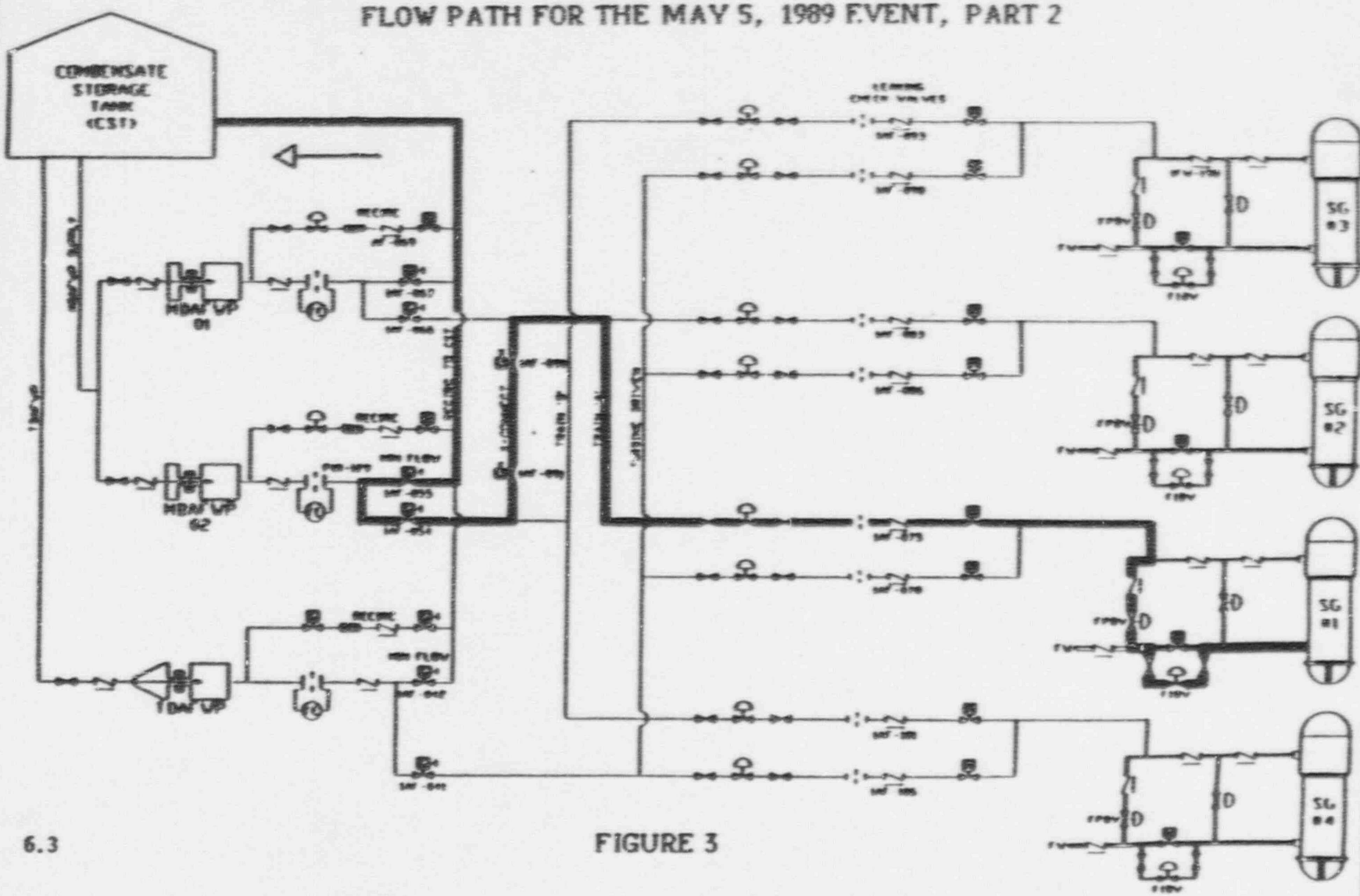
FLOW PATH FOR THE MAY 5, 1989 EVENT, PART 1



6.2

FIGURE 2

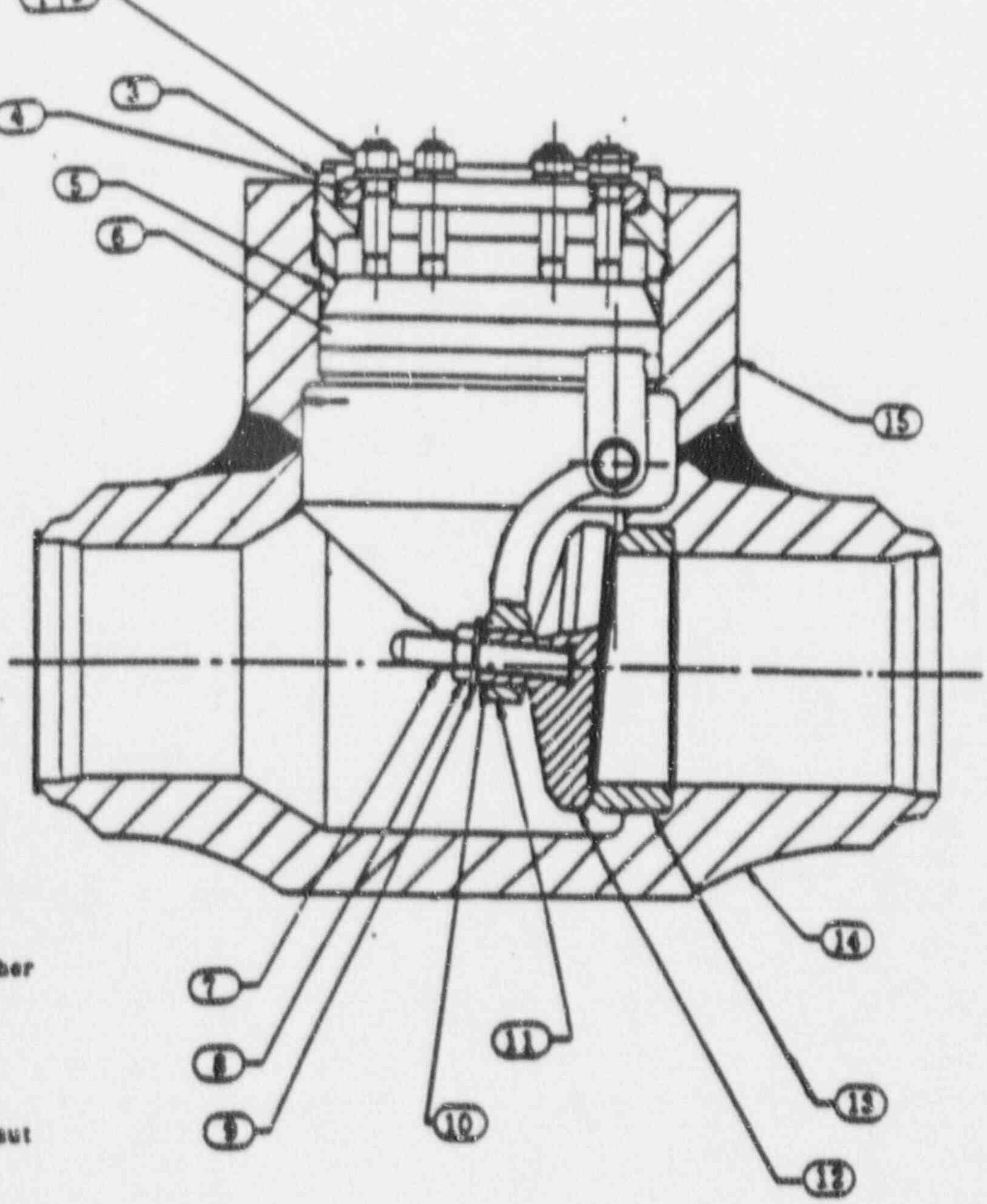
FLOW PATH FOR THE MAY 5, 1989 EVENT, PART 2



6.3

FIGURE 3

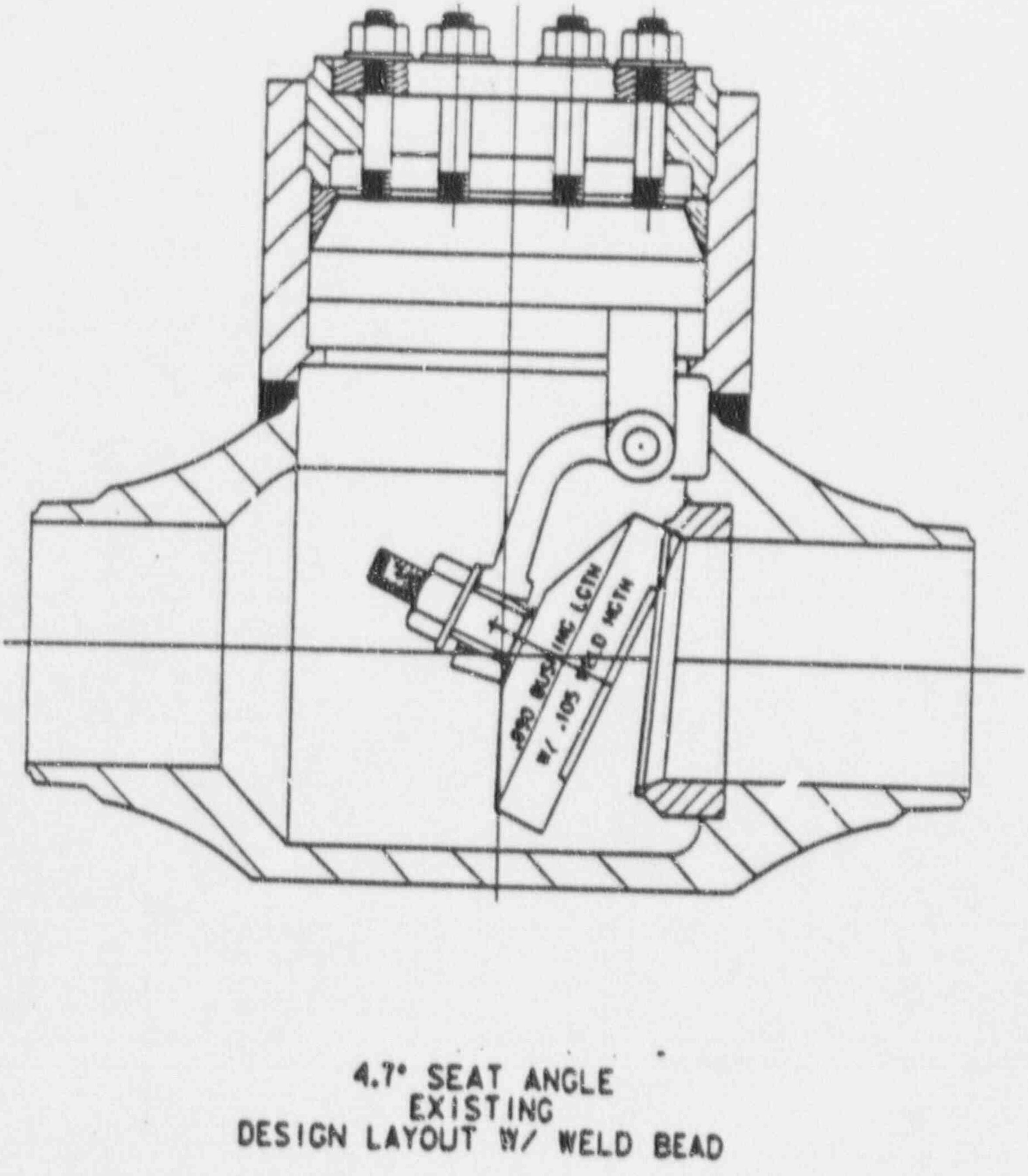
Typical Borg-Warner Check Valve Assembly



1 Hex nut
2 Lock washer
3 Retainer
4 Spacer
5 Seal
6 Bonnet
7 Stud
8 Hex jam nut
9 Washer
10 Bolt
11 Arm
12 Disk
13 Seat
14 Body
15 Neck

FIGURE 4

CAD Model of Valve 1AF-106 (As Found Condition)



4.7° SEAT ANGLE
EXISTING
DESIGN LAYOUT W/ WELD BEAD

FIGURE 5

6.6

FIGURE 6

Matrix of Unit 1 Borg-Warner Check Valves
(As Found Conditions)

	SIZE	RATING	RT COMPLETE	RT RESULTS	PROPER ALIGNMENT	RETAINER POSITION	SEAT ANGLE	DISC EDGE MACHINED	FILLET WELD MACHINED	AXIAL PLAY		
1AF-009	3	150										
1AF-014	6	150										
1AF-024	6	150										
1AF-032	8	150										
1AF-038	8	900										
1AF-045	3	900	X	OPN	N	0	7.7	N	N	.223		
1AF-051	6	900										
1AF-057	3	900	X	OPN	Y	0	5	Y	N	.142		
1AF-065	6	900										
1AF-069	3	900	X	CLSD	Y	0	8.3		N	.078		
1AF-075	4	900	Y	OPN		0	12	N	N	.165		
1AF-078	4	900	X	OPN	N	0	5	Y	N	.180		
1AF-083	4	900	X	OPN	Y	0	5	N	N	.206		
1AF-086	4	900	X	OPN	Y	0	5	Y	N	.19[illegible]		
1AF-093	4	900	X	OPN	Y	0	12	N	N			
1AF-098	4	900	X	OPN		0	5	Y	Y	.147		
1AF-101	4	900	X	OPN	Y	0	5	N	N	.210		
1AF-106	4	900	X	OPN	N	0	5	N	N	.179		
1AF-167	8	150										
1FW-191	6	600										
1FW-192	6	600										
1FW-193	6	600										
1FW-194	6	600										
1FW-195	6	600	X	CLSD								
1FW-196	6	600	X	CLSD								
1FW-197	6	600	X	CLSD								
1FW-198	6	600	X	CLSD	Y	.150	5	Y		.315		
1FW-199	6	600	X	CLSD								
1FW-200	6	600	X	CLSD								
1FW-[illegible]	6	600	X	CLSD								
1FW-[illegible]	6	600	X	CLSD								
1MS-[illegible]	4	900	X	OPN	Y	0	3.2	Y	Y	.194		
1MS-[illegible]	4	900	X	OPN	Y	0	3.8	Y	N	.124		
1CT-[illegible]			X									
1CT-[illegible]			X									

7.0 TABLE OF ACRONYMS

AFW	Auxiliary Feedwater
AFWP	Auxiliary Feedwater Pump
AIT	Augmented Inspection Team
AO	Auxiliary Operator
ASME	American Society of Mechanical Engineering
CAD	Computer Aided Design
CECO	Consolidated Engineering and Construction Organization
CMTR	Certified Material Test Report
CPRT	Comanche Peak Response Team
CPSES	Comanche Peak Steam Electric Station
CST	Condensate Storage Tank
CVCS	Chemical Volume and Control System
DCA	Design Change Authorization
DM	Design Modification
EPRI	Electric Power Research Institute
FA	Failure Analysis Report
FSAR	Final Safety Analysis Report
FW	Feedwater
HFT	Hot Functional Test
I&C	Instrumentation and Control
IEB	Information and Enforcement Bulletins
I&E	Inspection and Enforcement
INPO	Institute of Nuclear Power Operations
ISAP	Issue-Specific Action Plan
LER	Licensee Event Reports
MDAFW	Motor Driven Auxiliary Feedwater
MDAFWP	Motor Driven Auxiliary Feedwater Pump
MMI	Mechanical Maintenance Manual
MSM	Maintenance Section-Mechanical Manual
NCR	Nonconformance Report
NPRDS	Nuclear Plan Reliability Data System
NRR	Nuclear Reactor Regulation
NSSS	Nuclear Steam Supply System
OD	Outside Diameter
O&M	Operation and Maintenance Manual
PIR	Plant Identification Report
PORV	Power Operated Relief Valve
PR	Problem Report
QA	Quality Assurance
QC	Quality Control
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RO	Reactor Operator
RT	Radiograph Testing
SG	Steam Generator
SIF	Stress Intensification Factor

SOER	Significant Operating Experience Report
SWEC	Stone and Webster Engineering Corporation
TDAFW	Turbine Driven Auxiliary Feedwater
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TDR	Test Deficiency Report
TS	Technical Specifications
TU	Texas Utilities Electric Company (Formerly TUGCO)
UT	Ultrasonic Testing

Attachment D

CFUR 6

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

OCT 27 1989

In Reply Refer To:
Dockets: 50-445/89-30
50-446/89-30

Mr. W. J. Cahill, Jr.
Executive Vice President
TU Electric
500 North Olive Street, Lock Box 81
Dallas, Texas 75201

Dear Mr. Cahill:

This refers to the inspection conducted by Mr. H. H. Livermore and other members of the Augmented Inspection Team (AIT) during the period May 15 through June 16, 1989, concerning the check valve failures which allowed back-flow through the auxiliary feedwater system during hot functional testing of Unit 1 at the Comanche Peak Steam Electric Station. The team's findings were documented in Inspection Report 50-445/89-30; 50-446/89-30 and were discussed with you and members of your staff on June 16, 1989.

Our report requested you to submit a response summarizing lessons learned and planned corrective actions. You were also asked to address the weaknesses and recommendations identified by the AIT and the time frame for corrective actions. Your response to our July 10, 1989, letter was submitted to the NRC on August 18, 1989, by letter TXX-89596. A NRC request for clarification and additional information was transmitted to you by our letter dated September 14, 1989. Your response by letter TXX-89744 was dated October 16, 1989.

The collective significance of the potential violations identified in the enclosure to this letter suggest that, at least for the circumstances associated with this inspection, your evaluations of equipment and personnel failures lack thoroughness and depth, and your corrective actions were ineffective and untimely. Consequently, we believe that it would be useful to meet with you to discuss these findings.

You should be prepared to discuss the findings and conclusions of the AIT inspection at a noticed meeting within two weeks in Glen Rose, Texas.

Immediately following the meeting, we plan to conduct a brief enforcement conference with your management to discuss these and

8911020275

other regulatory matters identified in Enclosure 1 to this letter. At that conference please be prepared to present your assessment of safety significance, root cause(s), and your corrective actions.

You will be informed in writing of the NRC decision on enforcement action when that decision is reached after the conference. In accordance with 10 CFR 2, Appendix C, the enforcement conference will not be open to the public.

Your coop ration on this matter will be appreciated.

Dennis M. Crutchfield

D. M. Crutchfield, Associate Director
for Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Enforcement conference issues and
related regulatory requirements.

cc:
(See attached)

cc w/enclosure:

Roger D. Walker
Manager, Nuclear Licensing
TU Electric
Skyway Tower
400 North Olive Street, L.B. 81
Dallas, TX 75201

Juanita Ellis
President - CASE
1426 South Polk Street
Dallas, TX 75224

Texas Radiation Control
Program Director
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756

GDS Associates, Inc.
1850 Parkway Place, Suite 720
Marietta, GA 30067-8237

Honorable George Crump
County Judge
Glen Rose, Texas 76043

Ms. Billie Pirner Garde, Esq.
Robinson, Robinson, et al.
103 East College Avenue
Appleton, WI 54911

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

William A. Burchette, Esq.
Counsel for Tex-La Electric
Cooperative of Texas
Heron, Burchette, Ruckert & Rothwell
1025 Thomas Jefferson St., NW
Washington, DC 20007

TU Electric
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, Maryland 20814

E. F. Ottney
P. O. Box 1777
Glen Rose, Texas 76043

Jack R. Newman
Newman & Holtzinger
1615 L Street, NW
Suite 1000
Washington, DC 20036

George R. Bynog
Program Mgr./Chief Inspector
Texas Dept. of Labor & Standards
Boiler Division
P.O. Box 12157, Capitol Station
Austin, Texas 78711

Enclosure 1

Enforcement Conference Issues and Related Regulatory Requirements

1. The following activity appears to be contrary to:

Criterion V of Appendix B to 10 CFR Part 50 as implemented by Section 5.0, Revision 1, of the TU Electric Quality Assurance Manual states, in part, that activities affecting quality shall be prescribed by and accomplished in accordance with procedures.

Criterion XVI of Appendix B to 10 CFR Part 50 as implemented by Section 16.0, Revision 1, of the TU Electric Quality Assurance Manual which states, in part, that measures shall assure that significant conditions adverse to quality or plant safety are promptly identified and corrected to preclude repetition.

CPSES Operations Department Administration (ODA) Manual Procedure ODA-407, Revision 1, Section 6.1, which requires that plant systems and subsystems be operated in accordance with written approved procedures during normal, abnormal, and emergency conditions. Standard Operating Procedure SOP-304A, "Auxiliary Feedwater System," specifies steps necessary to perform various operations and alignments of the auxiliary feedwater system (AFW). The procedure specifically states that valve 1AF054 be closed prior to opening valve 1AF055.

On May 5, 1989, while performing steps in Procedure SOP-304A for system realignment, valves 1AF054 and 1AF055 were opened concurrently. This improper sequence allowed a reverse fluid flow path from the steam generators to the condensate storage tank via the AFW piping. This event occurred in a manner nearly identical to that of the April 23, 1989, event (see Violation 445/8924-V-01). Corrective actions for the April 23, 1989, event were inadequate to prevent recurrence on May 5, 1989.

2. The following activities appear to be contrary to:

Criterion XVI of Appendix B to 10 CFR Part 50 as implemented by Section 16.0, Revision 1, of the TU Electric Quality Assurance Manual which states, in part, that measures shall assure that significant conditions adverse to quality or plant safety are promptly identified and corrected to preclude repetition.

a. In 1985, Problem Report (PR) 85-132 and Failure Analysis Report (FA) 85-001, Revision 0, stated that the bonnet and retainer for check valve 1MS142 were incorrectly installed and placed too low in the body preventing proper closure of the check valve. The action to prevent recurrence stated in FA 85-001, Revision 0, included revising the assembly procedure and correctly reassembling the check valve. Thus, in 1985 the applicant had identified the root cause of the check valve back-leakage problem and had formulated corrective action which should have corrected the problem. The applicant failed to take this appropriate corrective action in a timely manner. Rather, the cause was changed and the failure was attributed to harsh flow conditions. The valve disc and stud were replaced and the valve seat was reconditioned. A recommended design review was not performed.

b. During Hot Functional testing (HFT) on April 5, 1989, the applicant identified significant back-leakage from the steam generators through three of the AFW supply line check valves. A Problem Report was not written and management was not informed. Work requests were written to repair the failed valves but were not given proper priority attention. The applicant failed to properly evaluate the back-leakage and failed to provide adequate and timely corrective action to prevent recurrence. As a result, significant backleakage occurred on April 23 and May 5, 1989.

c. On April 19, 1989, AFW pump testing revealed that miniflow check valve 1AF069 was experiencing significant back-leakage. The individual valve was reworked. At the time of valve rework, the applicant believed the problem to be isolated to valve 1AF069 which had excessive axial play. Generic corrective action was not addressed and the applicant failed to identify the root cause and to take adequate corrective action to prevent recurrence.

d. The AIT notes that it took an inordinately long period of time for the applicant to adequately identify the May 5 event and to report it as such, especially considering that it had a greater magnitude of severity than the April 23 event. The AIT team and the applicant's task team were not notified of the second event until May 15, 1989. The event was identified by PIR 89-129 only because the AIT persisted to question the event.

e. During physical disassembly of the system check valves. the AIT observed the following:

   (1) Some of the 4-inch check valve bonnets did not appear to be installed with the disk assembly parallel to the set ring.

   (2) The bonnet spacers on several of the check valves were deformed inward indicating over torquing of the bonnet stud fasteners.

   (3) Correspondingly, for the 4-inch valves that exhibited deformed bonnet spacers, the studs were also deformed (bent) inward which also indicates overtorquing of the fasteners.

   These potential deficiencies were not recorded by nonconformance reports (NCRs) or any other means that would ensure identification, disposition, and root cause determination.

3. The following activities appear to be contrary to:

   Criterion XI of Appendix B to 10 CFR Part 50 as implemented by Section 11.0, Revision 1, of the TU Electric Quality Assurance Manual which states, in part, that testing shall demonstrate that systems and components will perform satisfactorily in service. Contrary to the above, the following examples were identified:

   a. The applicant failed to perform post-modification and/or maintenance tests of Borg-Warner check valve internals that were removed and reworked in 1983, 1985, and on April 5, 1989.

   b. Under the applicant's preoperational test program, no testing was performed or planned, prior to plant operation, to ensure the AFW check valves were operable and capable of performing their intended function of preventing back-flow.

The NRC staff believes that the collective significance of the foregoing potential violations indicate that, at least for the circumstances associated with this inspection, your evaluations of equipment and personnel failures were inadequate and, similarly, the resulting corrective actions were ineffective. While actions are usually taken to correct known deficiencies, the actions are occasionally superficial or constrained to the immediate problem. Further, it appears that the large workload and schedule pressures continue to be at least a contributing causal factor. We also believe that these findings suggest that your quality assurance program is not sufficiently aggressive or inquisitive so as to anticipate and correct problems like these, before they occur.

Attachment E

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

January 25, 1990

Docket No. 50-445
EA-89-219

Mr. W. J. Cahill, Jr.
Executive Vice President
TU Electric
500 North Olive Street, Lock Box 81
Dallas, Texas 75201

Dear Mr. Cahill:

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTIES - $30,000 (NRC INSPECTION REPORT NO. 50-445/89-30; 50-446/89-30)

This refers to the inspection conducted by Mr. H. H. Livermore and other members of the Augmented Inspection Team (AIT) during the period May 15 through June 16, 1989, concerning the check valve failures which allowed backflow through the auxiliary feedwater system during hot functional testing of Unit 1 at the Comanche Peak Steam Electric Station. The team's findings were documented in Inspection Report 50-445/89-30; 50-446/89-30 and were discussed with you and members of your staff at the plant site on June 16 and again at NRC Headquarters in Rockville, Maryland, on July 17, 1989.

Our report of July 10, 1989, requested you to submit a response summarizing lessons learned and planned corrective actions. You were also asked to address the weaknesses and recommendations identified by the AIT and the time frame for corrective actions. Your response to our July 10, 1989, letter was submitted to the NRC on August 18, 1989, by letter TXX-89596. An NRC request for clarification and additional information was transmitted to you by our letter dated September 14, 1989. Your response by letter TXX-89744 was dated October 14, 1989. A public meeting and an enforcement conference were held in Arlington, Texas, on November 17, 1989. During those meetings, you presented a summary of the events and corrective actions to prevent recurrence.

On April 23, 1989, backflow occurred in the auxiliary feedwater system because (1) the plant operators failed to follow system alignment procedures and (2) check valves in the system were inoperable (stuck-open) because the disks had been misaligned as a result of incorrect valve assembly procedures. The first error occurred primarily because the operators did not have the proper sensitivity to the importance of system operability. Although corrective actions were taken following that event, a similar backflow event occurred during subsequent testing on May 5, 1989. The attitudes and practices demonstrated by workers and management during these events, if carried over to future power operations, would have constituted a significant operational safety problem.

~~9002010338~~*

Had these incidents occurred during plant operation, they would likely have warranted a Severity Level III categorization. However, because substantial construction activities were still underway during the conduct of the hot functional testing, we have concluded that TU Electric's actions during these events should be judged against the examples in Supplement II of Appendix C to 10 CFR Part 2. The three violations cited in the proposed enforcement action do not appear, even in the aggregate, to fit the examples for a Severity Level III issued under Supplement II, but they clearly have more than minor safety significance.

In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1989), Supplement II, the violations described in the enclosed Notice of Violation and Proposed Imposition of Civil Penalties (Notice) have been classified as Severity Level IV violations.

The corrective actions taken in response to the April 23, 1989 event should have prevented recurrence of the event and in view of the prior history of procedural violations and weaknesses in your corrective actions for equipment failures, the staff has concluded that a civil penalty for Violations A and B in the Notice is warranted. I have been authorized, after consultation with the Director, Office of Enforcement, and the Deputy Executive Director for Nuclear Materials Safety, Safeguards and Operations Support, to issue the enclosed Notice in the amount of $30,000. The base civil penalty for each of the two Severity Level IV violations is $15,000.

This civil penalty is being proposed to emphasize the need for management to ensure that the plant workers understand that quality is everyone's responsibility. During these events, the operations personnel failed to effectively recognize and act on conditions adverse to quality. Employees have to take proper precautions to prevent problems and the recurrence of problems. Managers should instill this attitude in subordinates and demonstrate it by example in their daily actions. In view of the completion schedule at that time, the plant staff should have been in an operational frame of mind. The adjustment factors have been considered in the decision to propose the civil penalty for this case. Therefore, the factors were not further considered in assessing these civil penalties. No civil penalty was proposed for Violation C because of the generic aspects related to inadequate backflow testing requirements for check valves.

We will evaluate the effectiveness of your corrective actions before authorizing the issuance of an operating license for Unit 1.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. After reviewing your response to this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Associate Director
for Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Notice of Violation and Proposed
Imposition of Civil Penalties

cc w/enclosure:
See next page

cc w/enclosure:
Mr. Robert F. Warnick
Assistant Director
for Inspection Programs
Comanche Peak Project Division
U. S. Nuclear Regulatory Commission
P. O. Box 1029
Granbury, Texas 76048

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Ms. Billie Pirner Garde, Esq.
Robinson, Robinson, et al.
103 East College Avenue
Appleton, Wisconsin 54911

Mrs. Juanita Ellis, President
Citizens Association for Sound Energy
1426 South Polk
Dallas, Texas 75224

E. F. Ottney
P. O. Box 1777
Glen Rose, Texas 76043

Mr. Roger D. Walker
Manager, Nuclear Licensing
Texas Utilities Electric Company
400 North Olive Street, L. B. 81
Dallas, Texas 75201

Texas Utilities Electric Company
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, Maryland 20814

William A. Burchette, Esq.
Counsel for Tex-La Electric
Cooperative of Texas
Heron, Burchette, Ruckert & Rothwell
1025 Thomas Jefferson Street, NW
Washington, D.C. 20007

GDS ASSOCIATES, INC.
Suite 720
1850 Parkway Place
Marietta, Georgia 30067-8237

Jack R. Newman, Esq.
Newman & Holtzinger
1615 L Street, NW
Suite 1000
Washington, D.C. 20036

Chief, Texas Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756

Honorable George Crump
County Judge
Glen Rose, Texas 76043

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTIES

TU Electric
500 North Olive Street, Lock Box 81
Dallas, Texas 75201

Docket No. 50-445

EA-89-219

During an NRC inspection conducted on May 15 through June 16, 1989, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1989), the Nuclear Regulatory Commission proposes to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2282, and 10 CFR Part 2.205. The particular violations and associated civil penalties are as follows:

A. Criterion V of Appendix B to 10 CFR Part 50 as implemented by Section 5.0, Revision 1, of the TU Electric Quality Assurance Manual requires that activities affecting quality be prescribed by and accomplished in accordance with documented procedures.

CPSES Operations Department Administration (ODA) Manual Procedure ODA-407, Revision 1, Section 6.1, requires that plant systems and subsystems be operated in accordance with written approved procedures during normal, abnormal, and emergency conditions. Standard Operating Procedure SOP-304A, "Auxiliary Feedwater System," specifies steps necessary to perform various operations and alignments of the auxiliary feedwater system (AFW). The procedure specifically states that valve 1AF054 be closed prior to opening valve 1AF055.

Contrary to the above, on May 5, 1989, while performing steps in Procedure SOP-304A for system realignment, valves 1AF054 and 1AF055 were opened concurrently. This improper sequence allowed a reverse fluid flow path from the steam generators to the condensate storage tank via the AFW piping. This failure to follow procedure and the resulting reverse fluid flow were nearly identical to the event on April 23, 1989 (see Violation 445/8924-V-01).

This is a Severity Level IV violation (Supplement II) (445/8930-V-01).
Civil Penalty - $15,000.

B. Criterion XVI of Appendix B to 10 CFR Part 50 as implemented by Section 16.0, Revision 1, of the TU Electric Quality Assurance Manual requires significant conditions adverse to quality or plant safety be promptly identified and corrected to preclude repetition. The identification of the significant condition adverse to quality shall be documented and reported to the appropriate levels of management.

9002010340

Contrary to the above:

B.1. In 1985, Problem Report (PR) 85-132 and Failure Analysis Report (FA) 85-001, Revision 0, identified a significant condition adverse to quality. The applicant failed to take adequate measures to assure that the cause of the failure was determined and corrective action taken to prevent recurrence. In the evaluation of a failure of check valve 1MS142, those reports concluded that the bonnet and retainer of the valve were installed too low in the valve body which prevented proper closure of the valve. The action to prevent recurrence stated in FA 85-001, Revision 0, included revising the assembly procedure and correctly reassembling the check valve. In addition, PR 85-132 recommended a design review. Upon further review, the applicant erroneously attributed the valve failure to harsh flow conditions, replaced the valve disk and stud, and reconditioned the valve seat, but did not perform the recommended design review. As a result of not following up on the initially identified cause of this precursor event, the applicant failed to take adequate corrective action and similar valve failures due to improper bonnet retainer installation occurred in 1989.

B.2. During Hot Functional testing (HFT) on April 5, 1989, the applicant identified a significant condition adverse to quality regarding back-leakage from the steam generators through three of the AFW supply lines. The applicant failed to take adequate measures to assure that the cause of the event was determined and corrective action taken to preclude recurrence. Work requests were written to repair the failed valves but did not adequately describe the backleakage. Consequently, the work requests were not given proper priority attention by management and a plant incident report was not written to require a prompt evaluation. As a result of not adequately identifying, evaluating, and correcting the cause of this precursor event, similar valve failures occurred on April 23 and again on May 5.

B.3. On April 23, 1989, the applicant identified a significant condition adverse to quality regarding backleakage from the steam generators through the AFW supply line check valves wherein operators failed to adhere to Standard Operating Procedure (SOP) 304-A. The applicant failed to take measures to assure that the cause of the event was adequately determined and corrective action taken to preclude recurrence. Consequently, a second failure to adhere to SOP 304-A resulted in a similar backleakage event on May 5, 1989.

B.4. On May 5, 1989, the applicant identified a significant condition adverse to quality regarding backleakage from the steam generators through the AFW supply line check valves. The applicant failed to promptly document this significant condition adverse to quality and to report it to appropriate levels of management. Specifically, the task team that was assigned with the lead responsibility for investigating check valve failures was not promptly informed of the event. Even after being notified, the task team did not actively investigate or document the May 5 event on a plant incident report, as required by Procedure STA-503, until May 12, 1989, after the NRC's Augmented Inspection Team insisted that these actions take place.

This is a Severity Level IV violation (Supplement II) (445/8930-V-02). Civil Penalty - $15,000.

C. Criterion XI of Appendix B to 10 CFR Part 50 as implemented by Section 11.0, Revision 1, of the TU Electric Quality Assurance Manual requires testing to demonstrate that systems and components will perform satisfactorily in service, including requirements and acceptance limits in applicable design documents.

Contrary to the above:

C.1. The applicant failed to provide post-modification and/or post-maintenance testing requirements for Borg-Warner check valves and did not perform testing of check valves whose internals were removed and reworked in 1983 and in 1985. As a result, the applicant failed to adequately demonstrate that these components would perform satisfactorily in service, in accordance with their applicable design requirements (see for example, the current Design Basis Documents (DBD)-ME-203 and DBD-ME-206).

C.2. Under the applicant's preoperational test program, no testing was performed or planned, prior to plant operation, to ensure the AFW check valves were operable and capable of performing their intended function of preventing backflow. The in-service test program in effect during the conduct of hot functional testing in 1989 did not require reverse flow testing of check valves. The applicable post-work test Procedure STA-623, Revision 3, only provided reference retest guidelines for the reverse flow testing of check valves subsequent to disassembly/repair/rework. No procedures for reverse flow testing existed at the time of the April 23 and May 5, 1989 events for check valves other than those specified as reactor coolant system boundary valves and those required for containment integrity.

This is a Severity Level IV violation (Supplement II) (445/8930-V-03).

Pursuant to the provision of 10 CFR 2.201, TU Electric is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. In accordance with

Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required under 10 CFR 2.201, the licensee may pay the civil penalty by letter addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, with a check, draft, or money order payable to the Treasurer of the United States in the cumulative amount of the civil penalties proposed above or may protest imposition of the civil penalties in whole or in part by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the licensee fail to answer within the time specified, an order imposing the civil penalty will be issued. Should the licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation listed in this Notice in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section V.B of 10 CFR Part 2, Appendix C, should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282c.

The responses to the Director, Office of Enforcement, noted above (Reply to a Notice of Violation, letter with payment of civil penalty and answer to a Notice of Violation) should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Associate Director for Special Projects, and a copy to the NRC Resident Inspection staff of the Comanche Peak Project Division.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Associate Director
for Special Projects
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
the 25th day of January 1990.

Attachment F

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

January 12, 1990

Docket No.: 99900030/89-01

RECEIVED
OCT 16 1990

Mr. P. C. Valli, Chief Executive Officer
BW/IP International, Incorporated
200 Oceangate Boulevard
Suite 900
Long Beach, California 90802

Dear Mr. Valli:

This refers to the inspection conducted by Mr. R. Pettis, Mr. M. Snodderly, Mr. C. Hammer, and Mr. S. Matthews of this office on September 11-14, 1989 of your facility at Vernon, California and the discussions of the findings with Mr. F. Burgers, Vice President of Operations, of your staff at the conclusion of the inspection.

The inspection was conducted as a result of TU Electric's 10 CFR 50.55(e) report to the NRC which identified several swing check valves, manufactured by BW/IP International, Incorporated (BW/IP), which failed to backseat during hot functional testing performed at the Comanche Peak Steam Electric Station (CPSES) in May 1989. Subsequent to this event, TU Electric informed the NRC of a broken cast swing arm, a critical component internal to the swing check valve, and several other swing arms which failed certain metallurgical tests. These valves were installed in several key safety-related systems at the CPSES and raise concerns over the improper use of commercial grade nonpressure boundary items in safety-related applications.

During this inspection it was found that the implementation of your quality assurance (QA) program failed to meet certain NRC requirements. The most significant problem was the failure of BW/IP to adequately qualify suppliers of internal parts, per BW/IP established procedures, which were subsequently installed in safety-related check valves and pumps furnished to the nuclear industry. In one example, BW/IP had no documentation to support the use and qualification, since 1985, of ACME Castings, Incorporated, as a supplier of cast valve internals, including swing arms, which have been installed in swing check valves used in nuclear safety-related applications. ACME's quality program had been found unacceptable in 1985 by BW/IP; however, they were retained and utilized as an approved vendor without a documented basis. BW/IP also relied on certificates of conformance from ACME without a valid basis for accepting such certifications. A recent order for replacement swing arms for the CPSES was supplied by ACME.

It was also identified during the inspection that contrary to BW/IP procedures, BW/IP failed to perform implementation audits for suppliers holding a current Certificate of Authorization issued by the American Society of Mechanical Engineers (ASME). It should be noted that licensees and their subcontractors

9001180174

are responsible for ensuring that the supplier is effectively implementing the approved QA program as discussed in NRC Information Notice 86-21, issued March 31, 1986.

The inspectors also identified that BW/IP performed an inadequate review for suitability of 8 commercial grade replacement swing arms for safety-related use at the CPSES. BW/IP's dedication was inadequate with respect to verifying the mechanical and chemical properties of the swing arm material. In addition, the results of BW/IP's visual and dimensional inspection were not documented. At present, the NRC is preparing an Information Notice to all licensees on this subject. A copy of such notice will be sent to BW/IP upon its issuance.

The specific findings and references to the pertinent requirements are identified in the enclosures to this letter. Areas examined during the inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector.

The enclosed Notice of Violation is sent to you pursuant to the provisions of Section 206 of the Energy Reorganization Act of 1974. You are required to submit to this office within 30 days from the date of this letter a written statement containing: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed. You are also requested to submit a similar written statement for each item which appears in the enclosed Notice of Nonconformance. We will consider extending the response time if you can show good cause for us to do so.

In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2 Appendix C (1989), the violation described in the enclosed Notice has been classified as a Severity Level III problem because a Part 21 report by BW/IP or notification of a significant deviation to NRC licensees would have been required if BW/IP had adequately performed the required evaluation. This violation is of significant regulatory concern. However, a civil penalty is not being proposed because pursuant to 10 CFR 21.61, the failure to perform the evaluation did not appear to be the result of a knowing and conscious failure to provide the required notice.

The responses requested by the accompanying notices are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 95-511. In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and its appendices will be placed in the NRC's Public Document Room. In addition, a copy of this report will be forwarded to TU Electric and ASME for their review and information.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Original signed by
Brian K. Grimes

Brian K. Grimes, Director
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A-Notice of Violation
2. Appendix B-Notice of Nonconformance
3. Appendix C-Inspection Report No. 99900030/89-01

cc: Mr. W. J. Cahill, Jr., Executive
Vice President
TU Electric
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Melvin R. Green, Executive Director
Codes and Standards
American Society of
Mechanical Engineers
345 East 47th Street
New York, New York 10017

DISTRIBUTION: VIB R/F, RLPettis, JLieberman
DRIS R/F, MSnodderly, JRichardson, BKGrimes
CGHammer, EWBrach, WShier, BNL, ETBaker
Docket Files/RIDS Code IE:09

*Previously concurred

Document Name: BORG WARNER LETTER

OFC	:VIB:DRIS:NRR*	:VIB:DRIS:NRR*	:VIB:DRIS:NRR*	:EMEB:DET*	:BNL*	:VIB:DRIS:NRR*
NAME	:RLPettis	:MSnodderly	:SMMatthews	:CGHammer	:WShier	:ETBaker
DATE	:11/22/89	:11/22/89	:11/21/89	:11/22/89	:11/21/89	:11/24/89

OFC	:OE	:C:VIB:DRIS*	:D:DRIS:NRR*	:DD:EDO	:	:
NAME	:JLieberman	:EWBrach	:BKGrimes	:HThompson	:	:
DATE	:01/9 /90	:12/21/89	:12/22/89	:01/11 /90	:	:

OFFICIAL RECORD COPY

BW/IP International, Incorporated
Long Beach, California

EA-89-244

APPENDIX A

NOTICE OF VIOLATION

During an inspection conducted at the Vernon, California facility on September 11-14, 1989, two violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1989) the violations are listed below:

I. Section 21.21 "Notification of failure to comply or existence of a defect," of 10 CFR Part 21 requires, in part, that each individual or other entity subject to the regulations provide for evaluating deviations or informing the licensee or purchaser of the deviation in order that the licensee or purchaser may cause the deviation to be evaluated.

a. Contrary to the above, BW/IP could not provide documentation to support their basis for informing TU Electric that a deviation reported to them by TU Electric on June 1, 1989, did not constitute a reportable condition pursuant to the provisions of 10 CFR Part 21. The deviation concerned improper adjustment height of the check valve swing arm which is considered by BW/IP as a nonpressure boundary item however critical to the overall operation of the check valve. Disassembly and reassembly of the swing check valves by Comanche Peak personnel, performed in accordance with Borg-Warner (presently BW/IP) Procedure No. OMM 1003, dated March 15, 1977, caused the valve disc to sit too low within the valve body which led to excessive backleakage through 13 safety-related swing check valves. On June 9, 1989, BW/IP provided an expanded assembly manual, BW/IP Operation and Maintenance Instruction OMM 2361, originally dated March 5, 1984, to TU Electric to enhance TU Electric's ability to use manufacturer's recommended reassembly techniques. However, no other customers had been made aware of this revision nor had the BW/IP Evaluation Board performed an evaluation of the deviation in accordance with BW/IP procedures to support their conclusion that the deviation was not reportable under 10 CFR Part 21.

b. Contrary to the above, at the time of the inspection, BW/IP had not initiated an evaluation of a broken cast swing arm or several other swing arms that were metallurgically tested and determined to have material flaws (hot cracks). These deviations were discovered after TU Electric performed hot functional testing at the CPSES in May 1989. BW/IP had actual knowledge of these deviations since July 1989 when a copy of a Stone and Webster Engineering Corporation (SWEC) technical report was made available to BW/IP during a SWEC inspection of the Vernon facility.

~~9001180182~~

In both cases above, BW/IP failed to notify all affected customers of the deviation which would have resulted in the filing of a 10 CFR Part 21 report if BW/IP had adequately evaluated the deviation (89-01-01).

These two examples have been classified as a Severity Level III Violation (Supplement VII).

For The Nuclear Regulatory Commission

Brian Grimes

Brian K. Grimes, Director
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Dated at Rockville, Maryland
This 12th day of January, 1990

BW/IP International, Incorporated
Long Beach, California

APPENDIX B

NOTICE OF NONCONFORMANCE

During an inspection conducted at the Vernon, California facility on September 11-14, 1989, the implementation of the BW/IP quality assurance (QA) program was reviewed. The results of the inspection revealed that certain activities were not conducted in accordance with NRC requirements. These items are set forth below and have been classified as a nonconformance to the requirements of 10 CFR Part 50, Appendix B, imposed on BW/IP by contract, and the BW/IP Nuclear Program Quality Manual (NPQM), Second Edition, dated June 1, 1988.

I. Criterion III, "Design Control," of 10 CFR 50, Appendix B, requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Contrary to the above, BW/IP failed to adequately demonstrate the suitability of 8 replacement check valve swing arms supplied to the Comanche Peak Steam Electric Station. BW/IP's dedication consisted primarily of a material identity test, a visual, and a dimensional verification. However, the NRC inspectors determined that BW/IP's dedication was inadequate since the swing arms' primary critical characteristics, mechanical and chemical properties, could not be verified using the test instrument employed. The device used was only capable of sorting between generic alloy groups such as austenitic and martensitic stainless steels, but could not distinguish between any one of the four typical martensitic specifications used by BW/IP. In addition, the results of the visual inspection performed on the arms was not documented.

The check valve swing arm, classified by BW/IP as a critical nonpressure boundary item, is essential to the operation of the swing check valve which is used in various safety-related applications at the Comanche Peak Steam Electric Station and other nuclear facilities (89-01-02).

II. Criterion VII, "Control of Purchased Material, Equipment and Services," of 10 CFR Part 50, Appendix B, requires, in part, that measures be established to assure that purchased material, equipment, and services conform to the procurement documents and include provisions for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery. In addition, the effectiveness of the control of quality by the contractor shall be assessed at intervals consistent with the importance, complexity, and quantity of the product or services.

A. Section 7-3, "Vendor Surveys and Audits," of the BW/IP NPQM which in part implements Criterion VII of 10 CFR Part 50, Appendix B, requires in Section 7-3.3.(6b) that suppliers of safety-related QL 1, 3, and 4 items shall be surveyed initially and audited triennially thereafter.

Contrary to the above:

1. BW/IP failed to perform implementation audits, to ensure that the supplier was effectively implementing its approved QA program. In addition, BW/IP failed to perform triennial audits for 17 suppliers of safety-related material due to their status as holders of Quality Systems Certificates issued by the American Society of Mechanical Engineers (ASME). Items furnished to BW/IP by these suppliers included, but were not limited to, fasteners, castings, valves and valve internals, piping, vessels, special testing services, and filler material (89-01-03).

2. BW/IP failed to adequately qualify ACME Castings, Incorporated as a supplier of safety-related QL 3 and 4 items. ACME's quality program, based on Military Specification MIL-I-45208A, "Inspection System Requirements," was disapproved by BW/IP on November 11, 1985. On June 8, 1987, ACME's vendor status was changed to that of a QL 3 and 4 supplier. This change was based solely on ACME's certification that they comply with the provisions of 10 CFR Part 21 (89-01-04).

3. Forty-three suppliers, currently on the BW/IP Approved Vendor List (AVL) as suppliers of safety-related QL 1, 3, and 4 items, were not surveyed initially and have not been audited triennially (89-01-05).

B. Section 7-2, "Evaluation and Selection of Suppliers," of the BW/IP NPQM which in part implements Criterion VII of 10 CFR Part 50, Appendix B, states in paragraph 7-2.1 that the Supervisor of Quality Audits is responsible for evaluation of the prospective supplier's quality assurance program and for conducting surveys when required. Section 7-3, "Vendor Surveys and Audits," states in paragraph 7-3.4(1) that the results of each audit shall be summarized by the lead auditor on audit reports per BW/IP Procedure 18-1.

Section 18-1, "Quality Assurance Program Audits," of the BW/IP NPQM which in part implements Criterion XVIII of 10 CFR Part 50, Appendix B, further states in paragraph 18-1.3(1), that elements selected for audit shall be evaluated against requirements and that objective evidence shall be examined as necessary to determine if elements are implemented effectively.

Contrary to the above, Quality Survey/Audit Reports and Quality Audit Checklists for vendors/suppliers evaluated by BW/IP and currently on the BW/IP AVL do not provide sufficient objective evidence to demonstrate that the supplier's quality program had been effectively implemented (89-01-06).

III. Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

Section 16-1, "Corrective Action," of the BW/IP NPQM which in part implements Criterion XVI of 10 CFR Part 50, Appendix B, states that Requests for Corrective Actions (RCAs) may be issued as a result of any condition which is considered to be detrimental to quality. RCAs shall be issued to the Department Manager for instances involving ASME code deficiencies and for violations involving by-passed hold tags.

Contrary to the above, BW/IP's corrective action program is considered inadequate in that RCAs are not issued for conditions considered detrimental to quality for nonpressure boundary, non-ASME Code safety-related items (89-01-07).

IV. Criterion XVII, "Quality Assurance Records," of 10 CFR Part 50, Appendix B, requires that sufficient records shall be maintained to furnish evidence of activities affecting quality and shall include at least the following: operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analysis. Records shall also be identifiable and retrievable.

Section 17-1.0, "Control and Maintenance of Quality Records," of the BW/IP NPQM which in part implements Criterion XVII of 10 CFR Part 50, Appendix B, requires in Section 17-1.4(3) that quality assurance records be retained as outlined in Section 17-1.4, Table 1. Such records include code data reports, engineering design calculations, and drawings.

Contrary to the above:

1. BW/IP did not provide a system for adequate quality record retention and retrieval. The engineering design calculations supporting the basis of two valve product lines, used in safety-related applications, could not be produced during the inspection. The two valves identified were a 3-inch, 150 lb., stainless steel, manual gate valve supplied to the CPSES, Units 1 and 2, and a 12-inch motor operated gate valve supplied to Bellefonte, Units 1 and 2 (89-01-08).

2. Engineering Change Notices and related calculations were not available to support the identification of cause and the specific corrective actions taken to prevent recurrence for two deficiencies related to bolt torquing specifications for valves in the BW/IP product line. These valves are identified on BW/IP drawings 79760 and 80590 and were used in nuclear safety-related applications (89-01-09).

ORGANIZATION: BW/IP INTERNATIONAL, INCORPORATED
VERNON, CALIFORNIA

REPORT NO.: 99900030/89-01	INSPECTION DATE: September 11-14, 1989	INSPECTION ON-SITE HOURS: 150

CORRESPONDENCE ADDRESS: BW/IP International, Incorporated
2300 East Vernon Avenue
Vernon, California 90058

ORGANIZATIONAL CONTACT: Mr. R. Donald Ham, Manager of Quality
TELEPHONE NUMBER: (213) 587-6171

NUCLEAR INDUSTRY ACTIVITY: Manufacturer of valves and pumps used in safety-related nuclear applications.

ASSIGNED INSPECTOR: Robert L. Pettis Jr. 12/20/89
R. L. Pettis, Jr., Reactive Inspection Section No. 1 (RIS-1), Vendor Inspection Branch (VIB) Date

OTHER INSPECTOR(S): S. Matthews, Quality Assurance Specialist, VIB
M. Snodderly, Reactor Engineer, VIB
C. Hammer, Mechanical Engineer, NRC/EMEB
W. Shier, Brookhaven National Laboratory

APPROVED BY: Greg Cwalina 12/21/89
Gregory C. Cwalina, Acting Chief, RIS-1, VIB Date

INSPECTION BASES AND SCOPE:

A. BASES: ASME Section III, Subsection NCA 4000; 10 CFR 50, Appendix B; 10 CFR Part 21; and the BW/IP International, Incorporated (BW/IP) Nuclear Program Quality Manual, Second Edition.

B. SCOPE: Verify implementation of BW/IP's quality assurance program as a result of check valve failures reported at the Comanche Peak Steam Electric Station in May 1989.

PLANT SITE APPLICABILITY: Multiple

9001180188

ORGANIZATION: BW/IP INTERNATIONAL, INCORPORATED
VERNON, CALIFORNIA

A. VIOLATIONS:

1. Contrary to Section 21.21, "Notification of failure to comply or existence of a defect," of 10 CFR Part 21, BW/IP International, Incorporated (BW/IP) could not provide documentation to support their basis for informing TU Electric in a letter dated June 22, 1989, that a previous deficiency related to the adjustment height of the swing arm did not constitute a reportable condition pursuant to the provisions of 10 CFR Part 21. This condition led to excessive backleakage through 13 safety-related check valves. In addition, BW/IP also failed to notify all of its nuclear customers of the deviation. A 10 CFR Part 21 report would have resulted if BW/IP had evaluated the deviation.

   In addition, at the time of the inspection BW/IP, had not initiated an evaluation of a deviation concerning a broken cast swing arm and several other swing arms that were metallurgically tested and determined to have material flaws (hot cracks) which were discovered after TU Electric performed hot functional testing in May 1989 at the Comanche Peak Steam Electric Station (CPSES). BW/IP had actual knowledge of these deviations from a July 7, 1989 Stone and Webster Engineering Corporation (SWEC) technical report furnished to BW/IP during a SWEC inspection of BW/IP in July 1989 (89-01-01).

B. NONCONFORMANCES:

1. Contrary to Criterion III, "Design Control," of 10 CFR 50, Appendix B, BW/IP failed to adequately review for suitability, eight replacement swing arms supplied to the Comanche Peak Steam Electric Station (CPSES). The swing arm, classified by BW/IP as a critical nonpressure boundary item, is essential to the operation of the swing check valve which is used in various nuclear safety-related applications at the CPSES and other nuclear facilities (89-01-02).

2. Contrary to Criterion VII "Control of Purchased Material, Equipment, and Services," of 10 CFR 50, Appendix B, and Section 7-3.3(6b), "Vendor Surveys and Audits," of the BW/IP Nuclear Program Quality Manual (NPQM), Second Edition, dated June 1, 1988:

   a. BW/IP failed to audit 17 suppliers of nuclear safety-related items due to their status as holders of an American Society of Mechanical Engineers (ASME) Quality System Certificate. Items furnished to BW/IP from these suppliers included, but were not limited to, fasteners, castings, valves and valve parts, piping, vessels, special testing services, filler material, and wrought products (89-01-03).

b. BW/IP failed to qualify ACME Castings, Incorporated as a supplier of safety-related QL-3 and 4 items. ACME's quality program, based on Military Specification MIL-I-45208A, "Inspection System Requirements," was disapproved by BW/IP on November 11, 1985. On June 8, 1987, ACME's vendor status was changed to that of a QL-3 and 4 supplier based solely on ACME's certification that they comply with the provisions of 10 CFR Part 21 (89-01-04).

c. BW/IP failed to survey initially and audit triennially 43 suppliers of safety-related QL-1, 3, and 4 items currently on the BW/IP Approved Vendors List (AVL) (89-01-05).

d. Quality Survey/Audit Report's and Quality Audit Checklists for vendor/suppliers evaluated by BW/IP are incomplete and/or inadequate to determine that the supplier's quality program had been effectively implemented (89-01-06).

3. Contrary to Criterion XVI of 10 CFR 50, Appendix B, and Section 16, "Corrective Action," of the BW/IP NPQM, Requests for Corrective Actions (RCAs) are not issued for conditions considered detrimental to quality for nonpressure boundary, non-ASME Code safety-related items (89-01-07).

4. Contrary to Criterion XVII, "Quality Assurance Records," of 10 CFR 50, Appendix B, and Section 17, "Control and Maintenance of Quality Records," of the BW/IP NPQM, an adequate system for quality record retention and retrieval did not exist. The engineering calculations to support the design basis of a 3-inch, 150 lb. stainless steel, manual gate valve supplied to the CPSES, and a 12-inch motor-operated gate valve supplied to Bellefonte and used in a safety-related application, could not be produced during the inspection (89-01-08).

   Contrary to the above, Engineering Change Notices and supporting engineering analyses were unavailable to support field changes of bolt torque specifications implemented as a result of two deficiency reports submitted by the Tennessee Valley Authority to the NRC for a 6-inch and a 12-inch motor operated gate valve installed in safety-related applications at the Bellefonte and Watts Bar nuclear power plants (89-01-09).

C. UNRESOLVED ITEMS:

1. Section 21.51, "Maintenance of Records," of 10 CFR Part 21 requires that records be maintained to assure compliance with the regulation. However, BW/IP was unable to produce records that documented evaluations for three occurrences that were reported to the NRC by licensees through 10 CFR 50.55(e). These licensee reports included:

   a. Overtorqued bolts on a flow control valve at Bellefonte Units 1 and 2, reported to the NRC by the Tennessee Valley Authority on November 20, 1981.

   b. Overtorqued studs on gate valve motor operators at Watts Bar and Bellefonte, reported to the NRC by the Tennessee Valley Authority on February 16, 1981.

   c. Oversized motor-operated valve stem keys that were supplied by BW/IP to the Perry Plant. This item was reported to the NRC by Cleveland Electric Illuminating Company on January 11, 1984.

   In each case, BW/IP was unable to produce documentation to support that an evaluation of these deviations was conducted as required by 10 CFR Part 21. BW/IP stated that these records may be in storage. This item will be reviewed during a future inspection (89-01-10).

2. During the inspection it was noted that BW/IP performs an Acceptance Test Procedure (ATP) on safety-related check valves prior to delivery. Based on Criterion XVII of 10 CFR 50, Appendix B, and Section 17 of the BW/IP NPQM, the results of these tests should be maintained as quality records. However, BW/IP was unable to produce the ATP results for the 3 and 4-inch check valves supplied to the CPSES, which subsequently failed during hot functional testing. BW/IP stated that these records may be in storage. This item will be reviewed during a future inspection (89-01-11).

3. Documentation was unavailable during the inspection to support the procurement, qualification of suppliers, and the overall nuclear quality assurance program in-place at the Borg-Warner Nuclear Valve Division, Van Nuys, California, prior to 1986 for

the swing check valve product line. BW/IP stated that these records may be in storage. This item will be reviewed during a future inspection (89-01-12).

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

This area was not reviewed during the inspection.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

Background:

The Borg-Warner Corporation was a large company with many branches. Of these branches, the industrial products branch consisted of three divisions including the Nuclear Valve Division located in Van Nuys, California. Each division had at its location, Quality Assurance, Engineering, and Procurement programs. The Nuclear Valve Division and the Byron Jackson Pump division had N-stamps and provided material to the nuclear industry. In late 1986, the nuclear valve product line was transferred from Van Nuys to the Byron Jackson Pump Division, located in Vernon, California. All activities are now controlled by the Byron Jackson Quality Assurance Program. The Nuclear Valve Division discontinued it's N-stamp at that time and became the Fluid Controls Division. In 1987, Borg-Warner Corporation sold the industrial products group to its existing management and it was renamed BW/IP International, Incorporated.

1. Root Cause Analysis and Evaluation of Failed Swing Arms at the Comanche Peak Steam Electric Station (CPSES).

In May 1989 while performing hot functional testing at the CPSES, several swing check valves failed which allowed backflow through the auxiliary feedwater system. As a result, an NRC augmented inspection was conducted on May 15-June 16, 1989. The results of this inspection are documented in NRC Report No. 50-445 and 446/89-30, dated July 7, 1989. The licensee, TU Electric, contracted with the Stone and Webster Engineering Corporation (SWEC) to perform a root cause analysis of three swing check valve swing arms. The results indicated that one swing arm was broken, leaving the disk completely detached from the valve body, while the other two swing arms were found to contain flaws, but were not broken. The swing arms were originally specified to be of alloy 17-4 PH martensitic stainless steel in accordance with Aerospace Material Specification (AMS) 5398 and heat treated to an H1100 condition per Military Specification MIL-H-6875, Class D. The

ORGANIZATION: BW/IP INTERNATIONAL, INCORPORATED
VERNON, CALIFORNIA

REPORT NO.: 99900030/89-01	INSPECTION RESULTS:	PAGE 6 of 17

SWEC report, "Evaluation of Swing Arm Failure/Casting Flaws," (Report No. 19245-ME(B)-1, dated July 7, 1989) was provided to BW/IP during a SWEC inspection of the Vernon, California facility in July 1989. The SWEC report concluded that the swing arms were improperly cast and heat treated. The major conclusions of the report are as follows:

a. The overall quality of the swing arm castings is generally poor and contained porosity, hot-cracks, and chemical segregation.

b. The failure of the swing arm initiated from surface defects formed during solidification or cooling during the casting process.

c. The swing arms did not receive adequate heat treatment to produce the H1100 condition and had been weld repaired with austenitic weld material.

d. Normal nondestructive inspection techniques may not reveal hot cracks similar to those identified in the failed swing arm.

e. Alternative materials should be considered for the swing arm part.

The NRC inspection team traced the origin of the swing arms, identified as part numbers 72225 and 73994, to the Industrial Pattern and Casting Company, with subsequent heat treatment performed by the Valley Heat Treating Company. The records reviewed indicated over 1000 swing check valves have been supplied to various customers for eventual use in nuclear applications (Attachment I).

The inspectors reviewed BW/IP's Nuclear Stress Report (NSR) 75500, dated October 26, 1976, concerning the broken and flawed CPSES check valve. The methodology incorporated in the report included the effects of dead weight, seismic, and other occasional loadings, but did not include the effects of large dynamic loads and transients that are possible during rapid valve closure caused by reverse fluid flow. Stress levels analyzed in the report for the swing arms were noted to be low. It is the NRC staff's opinion that large dynamic loads and transients may result in failure of a flawed, but not yet broken, swing arm.

2. 10 CFR Part 21

BW/IP's Procedure L-A-16, "Compliance with 10 CFR Part 21," dated December 9, 1987, establishes standard practices for identifying, documenting, evaluating, and reporting identified deviations pursuant to 10 CFR Part 21. Deviations identified are evaluated by the BW/IP Evaluation Board which consists of the Manager of Quality, the Director of Engineering, and the appropriate Project Manager. The evaluation board determines if the deviation is reportable or not and documents the justification.

On June 1, 1989, TU Electric made BW/IP aware of a possible deviation concerning their swing check valves. The deviation, which involved the valve disc sitting too low within the valve body, resulted from improper disassembly and reassembly of the valves, which were performed by licensee personnel in accordance with Borg-Warner Procedure No. OMM 1003, dated March 15, 1977. On June 9, 1989, BW/IP provided an expanded assembly manual, BW/IP Operation and Maintenance Instruction OMM 2361, originally dated March 5, 1984, to TU Electric to enhance TU Electric's ability to use manufacturers recommended reassembly techniques. However, no other customers had been made aware of this revision nor had the BW/IP Evaluation Board performed an evaluation of the deviation in accordance with BW/IP procedures to support their conclusion to TU Electric that the deviation was not reportable under 10 CFR Part 21.

As a result of inspections conducted after the CPSES backseat issue, TU Electric later informed the NRC of a broken cast swing arm identified during their review. SWEC was contracted to perform a metallurgical analysis of the failed swing arm, which was documented in a July 7, 1989, technical report furnished to BW/IP during a SWEC inspection of the Vernon, California, facility in July 1989. As of the completion of the NRC staff's inspection, BW/IP had not evaluated the deviation identified to them by the SWEC report. As a result, Violation 89-01-01 was identified during this part of the inspection.

3. BW/IP Design Review

This area of the inspection concentrated on a review of the BW/IP design procedure, supporting analyses, and the quality system used to accomplish these activities. Independent calculations to verify BW/IP analysis methods were not performed by the NRC inspectors during this part of the inspection.

a. BW/IP's NSR 70180, dated April 27, 1973, and revised April 8, 1989, describes the stress analysis for a Class 1, 8-inch, 1500 lb swing check valve. The analysis included a description of the applicable ASME Code, pressure and temperature design specifications, and 13 plant transients that represented the operating environment that occurred throughout the life of the valve. This plant transient specification was used in the fatigue analysis supporting the valve design. Calculations were performed using referenced formulas for the stress analysis and the results were reviewed and approved by an independent reviewer. It was also noted that considerable margin was available with respect to the allowable stress for each valve analyzed. However, dynamic loads generated during operation of the valve were not included in this analysis.

b. NSR 75520, dated October 26, 1976, described the stress analysis of 3 and 4-inch, 150 lb, stainless steel check valves that were supplied to the CPSES and were designed to ASME Section III, Class 2 requirements. Areas of the valve that were analyzed included the valve body and arm, clevis and bolt, pivot pin, disk, flange and bolt, and the bonnet. Thermal transients were specified at 100°F/hour. The seismic load factor was 3g in each of two orthogonal horizontal directions and 2g in the vertical direction. These seismic accelerations were assumed to act simultaneously and appear to be typical values used for seismic load factors. The analysis results indicated that the calculated stress in the valve body was limiting with respect to the allowable stress and that the available margin was greater than a factor of 2 times the calculated maximum stress.

c. BW/IP report number 401HDC1-005, Revision A, dated March 28, 1989, describes the stress analysis of a 3-inch, 150 lb, stainless steel, manual gate valve supplied to the CPSES. The valves were designed as ASME III, Class 3 components with a design pressure of 275 psig for application in the plant service water system. Nine different valve sections were considered in the analysis with the limiting calculated stress occurring in the valve gate. It was noted that stresses computed for the faulted load condition were conservative compared with stress limits for the normal mode; however, a very small margin existed with respect to the allowable stress.

The NRC inspectors requested the engineering calculations to support the basis for the valve design; however, these design documents could not be located in the BW/IP files. Similarly, the engineering calculations to support the design of a 12-inch motor operated gate valve supplied to

ORGANIZATION: BW/IP INTERNATIONAL, INCORPORATED
VERNON, CALIFORNIA

Bellefonte, Units 1 and 2, also could not be located. Design information for this 12-inch valve was requested since the same valve was the subject of a 50.55(e) deficiency report issued to the NRC by the Tennessee Valley Authority (TVA). As a result of BW/IP not being able to retrieve the information, Nonconformance 89-01-08 was identified during this part of the inspection.

d. The inspectors reviewed two 10 CFR 50.55(e) deficiency reports for BW/IP valves supplied to the Bellefonte and Watts Bar plants. The deficiencies involved overtorquing of bolts which produced elongation and subsequent failure of the bolts when torqued to values specified on the BW/IP drawings. The product lines involved were the 12-inch motor operated gate valve (BW/IP drawing 80590) previously discussed in Item 3(c) above, and a 6-inch motor operated gate valve used in the auxiliary feedwater system (BW/IP drawing 79760). The resolution of both deficiencies was that incorrect bolt torquing values were specified on the drawings. The Engineering Change Notices and the supporting calculations were requested, however, BW/IP was unable to produce such documentation during the inspection. As a result, Nonconformance 89-01-09 was identified during this part of the inspection.

e. The NRC inspectors reviewed NSR 75500, dated October 26, 1976, which was prepared for the 3 and 4-inch, 150 lb, carbon steel swing check valves which failed during hot functional testing at the CPSES. The stress analysis indicted that considerable margin (greater than an factor of 2) existed with respect to the allowable stress at the limiting location in the valve body. BW/IP correspondence also indicated that these valves were performance tested prior to delivery in 1975. However, the NRC inspectors were unable to review the documentation since it was in storage at an offsite location. As a result, Unresolved Item 89-01-11 was identified during this part of the inspection.

f. During a review of the operating history associated with the Borg-Warner valve product line, several deficiency reports were selected for review at BW/IP. BW/IP was requested to supply documentation associated with the corrective action for the following issues:

   1. Overtorqued bolts on a flow control valve at the Bellefonte Nuclear Plant as reported by TVA on November 20, 1981.

2. Overtorqued studs on gate valve operators at the Watts Bar and Bellefonte Nuclear Plants as reported by the TVA on February 16, 1981.

3. Oversized motor-operated valve stem keys supplied to the Perry Nuclear Plants as reported by Cleveland Electric Illuminating Company on January 11, 1984.

As a result of BW/IP's stating the documentation could not be reviewed due to storage at an offsite location, Unresolved Item 89-01-10 was identified.

4. Review of Swing Arms as Replacement Parts

Borg-Warner Incorporated, Van Nuys, California procured swing arms and other valve internals for various models and sizes of swing check valves. The records, available for review during the inspection, indicated that Industrial Pattern and Casting Company and Valley Heat Treating supplied the majority of the castings for arms used in swing check valves. Historical receiving inspection reports reviewed indicate that originally the swing arms were heat-treated prior to machining. Subsequent orders for cast swing arms were procured with heat treatment as a post-machining operation versus a pre-machining operation. Traceability to material test reports and certificates of conformance were available for some orders; however, traceability to each casting could not be established. Several purchase orders to Industrial Pattern and Casting Company for the same part number imposed the requirements of a quality program and many others did not. Traceability distinction between the different purchase orders was not maintained and the total inventory of any particular part number would represent commingled castings from various purchase orders.

In late 1986 the remaining inventory of swing arms was transferred from Van Nuys, California to the newly formed BW/IP in Vernon, California, and rendered "Commercial Grade," as defined in 10 CFR Part 21, due to the lack of documentation supporting the qualification of this material. The inspection identified the following examples of Van Nuys inventory which were inadequately reviewed by BW/IP for suitability for use in safety-related swing check valves furnished to TU Electric And Arizona Public Services. As a result, Nonconformance 89-01-02 was identified during this part of the inspection.

a. BW/IP Job Number 891H2977 for TU Electric required that eight machined swing arms, Part No. 72225, be drawn from inventory. The route sheet required a material identity check per alloy identity procedure GS-1563, Revision D, dated May 24, 1988. This material identity check uses a comparison type instrument based on thermal conductivity differences between metals. Paragraph 3.0 of the procedure states, "Metals of the same or similar chemistry will produce instrument readings repeatable over established ranges, thereby generically sorting the test pieces." The instrument is capable of sorting between generic alloy groups, such as austenetic and martensitic stainless steels. However, the instrument cannot distinguish between any one of the four typical martensitic stainless steel specifications used by BW/IP. There was no verification of the mechanical properties of the swing arms and the verification of chemical properties is considered inadequate. In addition, traceability to a material manufacturer's material test report or certificate of conformance could not be established. Also, the results of BW/IP's visual and dimensional inspection performed on the arms were not documented.

b. BW/IP Job Number 861L0201 for Arizona Public Services required one machined swing arm, Part No. 73748, to be drawn from inventory and used in a bonnet/arm disk assembly. The route sheet did not describe the steps necessary to determine compliance with the material specification, dimensional and configuration conformance, part identification, or nondestructive examination (NDE) requirements of the as-cast or machined surfaces. Therefore, the quality of this arm is indeterminate.

c. BW/IP Job Number 861L2488 for Arizona Public Services required 11 machined arms, Part No. 72194, to be drawn from inventory. The route sheet again did not describe the necessary steps to determine compliance with the material specification or NDE requirements as stated in Item 4(b) above. Therefore, the quality of these arms is also indeterminate.

5. Review of Corrective Actions

A review of corrective actions performed by BW/IP indicates that Requests for Corrective Actions (RCAs) for non-ASME Code, nonpressure boundary parts used in safety-related applications covered by 10 CFR 50, Appendix B were not initiated. Section 16-1.2 of the BW/IP NPQM requires that RCAs be issued only for deficiencies identified in ASME Code items and violations involving by-passed hold tags. However, RCAs are not required by the NPQM for deficiencies identified in non-ASME Code, nonpressure boundary, safety-related items. Additionally, follow-up to RCAs is only required by Section 16-1.5 of the BW/IP NPQM to be performed for ASME Code items. The BW/IP quality inspector stated that logs used for trending to preclude repetition of RCAs are not maintained and RCAs are not generally issued for deficiencies identified in safety-related items that are non-ASME Code, nonpressure boundary. The process files for items excluded under this practice were reviewed during the inspection and the inspector verified the practice of not applying RCAs to those items. The BW/IP NPQM does not adequately provide measures required by Criteria V and XVI of 10 CFR 50, Appendix B to assure that all conditions considered detrimental to quality for safety-related, non-ASME Code, nonpressure boundary items are addressed by a corrective action program. As a result, Nonconformance 89-01-07 was identified during this part of the inspection.

6. Review of BW/IP's Approved Vendor List

The inspectors reviewed the Approved Vendor List (AVL) for nuclear safety-related QL-1, 3 and 4 items and services dated July 12, 1989. During this review it was determined that 43 vendors, available to supply nuclear safety-related items and services, were not surveyed initially and have not been audited triennially as required by Section 7-3.3 of the BW/IP NPQM. As a result, Nonconformance 89-01-06 was identified during this part of the inspection.

The review also identified 17 vendors holding ASME Quality System Certificates (QSC) as Material Manufacturers (MM) and/or Material Suppliers (MS) who also have not been audited due to their status as QSC holders. Therefore, BW/IP has not adequately ensured that the vendors are effectively implementing their quality program as required by BW/IP procedure. This issue was previously discussed in NRC Information Notice No. 86-21: Recognition of American Society of Mechanical Engineers Accreditation Program for N-Stamp Holders, dated March 31, 1986.

In one example, the NRC inspectors identified purchase orders for cast swing arms placed by BW/IP with the Atlas Foundry & Machine Company, an ASME QSC holder. The swing arms ordered were replacements for the failed swing arms identified by TU Electric. Atlas is one of the 17 QSC holders not audited by BW/IP to ensure effective implementation of their quality program.
As a result, Nonconformance 89-01-04 was identified during this part of the inspection.

In another example, the basis for qualifying the ACME Casting Company as an approved nuclear supplier of QL-3 and 4 cast valve internal parts was reviewed. The basis for approval of ACME relied on Military Specification MIL-I-45208A, "Inspection System Requirements." ACME's QA program was approved by the Byron Jackson Pump Division of Borg-Warner in May 1980. In 1985, ACME's program was reaudited and determined to be "inadequate requiring extensive manual revisions." As a result, ACME's status was changed to that of an unapproved supplier. On June 5, 1987, ACME was reclassified as an approved supplier of QL-3 and 4 safety-related items based upon ACME's certification that they complied with the provisions of 10 CFR Part 21. A review of safety-related purchase orders placed with ACME since 1986 identified 10 orders for various cast valve internals including the swing arm, yoke, and clevis. The NRC inspectors were unable to determine from the documentation reviewed, the customer or the nuclear facility involved in each of the procurements. As a result, Nonconformance 89-01-04 was identified during this part of the inspection. Qualification of the remaining 125 vendors was not reviewed during the inspection.

7. Review of Vendor Surveys and Audits Performed by BW/IP

The NRC inspectors reviewed the Quality Survey/Audit Reports and the Quality Audit checklist for several suppliers that have been evaluated and approved by BW/IP and are currently on the AVL for furnishing nuclear safety-related QL-1, 3, and 4 items and services. The QL-1 category applies to pressure boundary items and component supports in accordance with ASME Section III, Division 1, and NQA-1. This also includes activities related to Material Manufacturers and Material Suppliers holding a QSC. The QL-3 category applies to items manufactured or procured which require the highest level of quality as determined by BW/IP design engineering and references the requirements of NQA-1, ANSI N45.2, and 10 CFR 50, Appendix B. The QL-4 category applies to items manufactured or procured which require no more documentation than material test reports or certificates of conformance and references NQA-1, ANSI N45.2, and 10 CFR 50 Appendix B.

The Quality Survey/Audit Report and Quality Audit checklist for the suppliers discussed below were identified by the NRC inspector to not provide sufficient objective evidence to demonstrate effective implementation of the supplier's quality program. As a result, Nonconformance 89-01-06 was identified during this part of the inspection.

a. Eagle Pattern & Manufacturing Company, Seattle, Washington is currently listed on the AVL (dated July 12, 1989) as a supplier of QL-3, and 4 castings. The vendor was last audited by BW/IP on November 1, 1983. Documentation of the audit consisted of a four page "Vendor Quality Evaluation Questionnaire." A review of the questionnaire identified no objective evidence to substantiate the ability of the vendor to implement a quality program consistent with the applicable portions of 10 CFR 50, Appendix B. The Quality Control Manual (QCM) is dated March 1, 1980 (Revision 0).

b. M&N Metals, Incorporated, Odessa, Texas is currently listed on the AVL (dated July 12, 1989) as a supplier of QL-3, and 4 ferrous/nonferrous castings & wrought products. Welding, NDE, and heat treating is not within the scope of M&N as established by BW/IP. M&N was last audited on August 21, 1987 by BW/IP. An excerpt from the Quality Survey/Audit Report states, "Survey shows compliance to applicable portions of MIL-I-45208A and also meets safety-related requirements of 10 CFR 21 and 10 CFR 50, Appendix B. No NDE, heat treating or welding is allowed. Rough machined items only." However, the quality audit checklist reviewed does not describe any objective evidence evaluated by the auditor to substantiate M&N's ability to implement a quality program. The checklist also indicates that work instructions for machining is "Not Applicable" although rough machining is currently in M&N's scope. No procedures exist for the selection and surveillance of subcontractors, although metallurgical laboratory needs are subcontracted. Procedures for the identification, control, and issuance of material were not audited.

The inspectors independently reviewed M&N's QCM, dated January 2, 1987, and identified that the QCM does not adequately address the applicable criteria of 10 CFR 50, Appendix B. It was also noted that the format and wording was identical to the QCM for Eagle Pattern & Manufacturing Company described in Item 6(a) above.

c. GMC Precision Tool Corporation, La Habra, California, is currently listed on the AVL (dated July 12, 1989) as a supplier of QL-3 and 4 machined parts; including material, tooling and fixtures, and special processes. Welding is not allowed to be performed by GMC. GMC was last audited on

October 17, 1988 by BW/IP. The inspector reviewed the audit checklist for GMC which did not describe any objective evidence evaluated by the auditor to substantiate GMC's ability to implement a quality program. The audit also indicated that material supply is in GMC's authorized scope, although the audit checklist did not address procurement control or control of subcontractors as would be necessary to effectively supply material to BW/IP.

d. Toolex, Houston, Texas is currently listed on the AVL (dated July 12, 1989) as a QL-1 supplier of machining services for BW/IP furnished ASME Code material. Welding, NDE, and heat treating is not allowed to be performed by Toolex. For ASME Code material, Toolex was last audited by BW/IP on May 13, 1988 for QL-1 items and January 16, 1987 for QL-3, and 4 items. The quality audit checklist did not describe any objective evidence evaluated by the auditor to substantiate Toolex's ability to implement a quality program commensurate with the level of services furnished. Although the audit checklist did not address procurement control or control of subcontractors, the file contained a copy of Toolex's QCM with annotations where the BW/IP auditor had extensive revisions and rewrites.

8. <u>Review of the Borg-Warner Corporation, Nuclear Products Approved Vendor List (NP/AVL)</u>

The inspectors attempted to review the quality program requirements in-place during the period (prior to the end of 1986 when BW/IP was formed) when safety-related swing check valves were manufactured by the Nuclear Valve Division of Borg-Warner, Van Nuys, California (BW/NVD). Since BW/IP could only provide the BW/NVD, Nuclear Products Quality Assurance Manual (NPQAM), ASME Code Section III, Division 1, Revision K, dated February 18, 1980, the evaluation of the program requirements and documentation to support the qualification of vendors during the time when the failed CPSES swing arms were produced could not be accomplished. However, the inspectors determined from the BW/NVD NPQAM, Section 7.0, "Control of Purchased Material, Items, and Services," Paragraph 7.2.1, that prospective vendors of pressure boundary material and critical, nonpressure boundary parts shall be surveyed prior to being added to the NP/AVL. Paragraph 7.2.12 requires that approved vendors be audited at intervals not to exceed 12 months.

ORGANIZATION: BW/IP INTERNATIONAL, INCORPORATED
VERNON, CALIFORNIA

REPORT NO.: 99900030/89-01	INSPECTION RESULTS:	

Due to storage at an offsite location, BW/IP was unable to produce records of vendor surveys/audits performed by BW/NVD during the inspection. However, BW/IP produced a copy of the Borg-Warner Fluid Controls, Approved Nuclear Vendor List, dated January 16, 1984, with continued revision pages, representing all deletions and additions since November 25, 1972. The accuracy and completeness of this documentation could not be determined during the inspection. A review of the documentation identified that two vendors used in the procurement and manufacture of the failed swing arm did not appear on the list, Valley Heat Treating, used between December 1975 to January 1977, and Peabody, used between November 1978 to December 1978. Another vendor, Pacific Steel Treating, was used in June 1979 but did not appear on the list until August 1980 and remained on the list through February 1986. Unresolved Item 89-01-12 was identified during this part of the inspection.

F. PERSONS CONTACTED:

*F. Burgers, Vice President Operations
G. Godwin, General Manager
*D. Gibson, Manufacturing Operations Manager
*W. Klenner, Nuclear Valve Product Manager
L. Boswell, Project Engineering Manager
*R. Ham, Manager of Quality
*J. Bartholomew, Senior Project Engineer
*J. Soet, Purchasing Manager

* Attended Exit Meeting

ATTACHMENT I

Potential Recipients of BW/IP Swing Check Valves

Palo Verde
Arkansas Nuclear One
Pilgrim Station Unit No. 2
San Onofre Unit No. 1
Perry Nuclear Power Plant
St. Lucie Unit No. 1
Catawba Nuclear Station
McGuire Nuclear Station
Oconee Nuclear Station
Diablo Canyon
Ginna Station
Sequoyah Nuclear Project
Watts Bar Nuclear Plant
Browns Ferry Nuclear Plant
Comanche Peak Steam Electric Station

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

January 30, 1990

Docket Nos. 50-445
and 50-446

Attachment
G

Mrs. Betty Brink, Board Member
Citizens for Fair Utility Regulation
7600 Anglin Drive
Fort Worth, Texas 76140

Dear Mrs. Brink:

SUBJECT: ALLEGATION OSP 89-A-0089

This is in response to the concerns raised by the Citizens for Fair Utility Regulation (CFUR) in the Request for Stay, dated October 16, 1989, your letter of November 8, 1989 and our meeting of December 7, 1989. Although the Commission' Order of October 19, 1989 only addresses the technical concerns and settlement agreement issues raised in the Request for Stay, the NRC staff has endeavored to evaluate all of CFUR's concerns. The purpose of this letter is to describe the basis for the NRC staff's resolution of those concerns.

The enclosure to this letter presents the NRC staff's conclusions regarding the fundamental technical issues. CFUR has not raised any issues not already considered by the staff. However, we recognize your desire for a further explanation of the resolution of those issues. In addition to the specific issues addressed in the enclosure, CFUR has also raised several philosophical issues which we should explain so as to provide a context for our conclusions regarding the more specific technical issues.

First, several CFUR representatives have suggested that we should consider your concerns with respect to the viability of light-water reactor technology. The NRC's responsibilities and authority are predicated on the Atomic Energy Act and the Energy Reorganization Act which, in conjunction with applicable case law, establish the fundamental premise that light-water reactor technology can be used as an energy source so long as an applicant for a license satisfies the applicable Federal regulations for that technology. The Final Environmental Statement for Comanche Peak (NUREG-0775, September 1981) addresses alternative energy-sources in accordance with the National Environmental Policy Act and concluded that the addition of the two units to TU Electric's system is expected to result in significant savings in system production costs, decreased dependence on fuel supplies of uncertain availability and increased system reliability. None of the issues raised by CFUR adversely affect the viability of light water reactors.

~~9002080102~~

Second, CFUR has suggested that all of the issues associated with Comanche Peak should be considered collectively as representing a trend or pattern of unacceptable behavior by TU Electric. As a result, you have concluded that the TU Electric organization is incapable of operating Comanche Peak safely. Similarly, you asked whether there is a threshold number of violations or errors which would cause the NRC to deny a license. The applicable Federal regulations, NRC enforcement policy and underlying quality assurance principles are intended to preclude mistakes, but all recognize that mistakes will be made, particularly for a venture as massive and complex as the construction of a nuclear power plant, and there are means to correct those mistakes. Further, even when mistakes are repetitive, the NRC's enforcement policy provides for civil penalties to emphasize the importance of effective corrective actions. Our enforcement policy also provides the means to suspend, modify, or revoke a license when we are concerned that repetitive mistakes might jeopardize public safety. NRC inspection and preoperational testing of plants are intended to identify construction related problems. Rarely are construction related problems so great that they cannot be corrected. Even programmatic breakdowns during construction have been corrected. Consequently, the NRC does not have a "threshold" of violations which would cause the denial of a license.

Nevertheless, we have attempted to evaluate the collective significance of CFUR's concerns and their relationship to past construction errors. In this evaluation, we have relied on the results of our review of the independent Comanche Peak Response Team (CPRT) findings, as is described in Supplement 20 to the Safety Evaluation Report for Comanche Peak (NUREG-0797) which was issued in November 1988. Such an evaluation of collective significance involves a long period of time, a large number of people, a wide variety of construction activities, and a judgment of the significance of the construction deficiencies that were identified by both the NRC and TU Electric. Based on (1) the relative significance of the enforcement history for Comanche Peak, (2) the wide variety in the construction deficiencies and TU Electric's efforts to correct these deficiencies, and (3) the nature and evolution of the accepted industry practices for the design and construction of nuclear power plants over the time that Comanche Peak has been under construction, we conclude that, while TU Electric could have done some things better as is reflected in the CPRT findings, Comanche Peak deficiencies have been corrected and there is now no discernable trend or pattern that would raise a serious safety concern or provide a basis for denial of an operating license.

Although the NRC has taken a number of enforcement actions and continues to identify violations related to TU Electric's activities, these actions are not unusual nor, in our view, are they so significant as to raise a concern about the ability of TU Electric's organization to safely operate the plant. Moreover, enforcement action may be necessary in the future to ensure TU Electric's continued vigilance so that weaknesses are corrected.

In a related matter, CFUR has also expressed concern about the significance of the Augmented Inspection Team (AIT) findings (50-445/446-89-30/30) following the check valve failures during hot functional testing. The staff's concerns regarding those findings are described in the subsequent enforcement action

(EA-89-219) which was issued on January 25, 1990. However, we consider these findings to be related to TU Electric's transition from construction activities to an operational environment. In that regard, we will rely on the staff's ongoing inspection program as well as the NRC's Operational Readiness Assessment Team to assess whether TU Electric's corrective actions, in response to the AIT findings, have been effective.

Third, CFUR has expressed a broad concern about TU Electric's management, primarily with respect to attitudes and implied policies. CFUR has characterized TU Electric's management as "arrogant" and alleged that they have misled the NRC and the public. The NRC staff has determined that TU Electric's management has appropriate commercial nuclear experience and written policies related to nuclear safety. Based on the NRC staff's dealings with TU Electric management and the results of several investigations, including an NRC panel review of intimidation and harassment issues in 1985, we conclude that TU Electric has not demonstrated a pervasive behavior that would be detrimental to safe operation of the plant. Moreover, while the NRC panel concluded in 1985 that a number of TU Electric's past management practices may have generated mistrust and suspicion so as to contribute to a lack of management credibility, more recent experience has demonstrated that TU Electric's performance has substantially improved in this regard, particularly as evidenced by the low number and significance of employee concerns over time.

Finally, CFUR has alleged that concerns expressed by a former NRC inspector at Comanche Peak and a group of "Anonymous NRC Inspectors" constitute an attempt by the NRC to "whitewash" Comanche Peak issues. On the contrary, the NRC established a process for differing professional opinions to encourage its employees to express their individual views so that potential safety issues would not be overlooked. The existence of differing professional opinions and individuals' concerns does not, in and of itself, constitute a safety issue. NRC management still has an obligation and responsibility to make decisions based on staff opinions. In this case, a Differing Professional Opinion panel was directed to review the concerns of the anonymous inspectors. The panel has completed its review and the resulting recommendations are currently being reviewed by senior NRC management. After action is taken on those recommendations, the results of the panel's review and related records will be made publically available. Similarly, the former NRC inspector's concerns, along with the results of the investigation that stemmed from those concerns, will be released to the public when the final reports are complete. It should also be noted that these staff opinions were considered in the staff's planning for the inspections related to operational readiness.

We recognize that CFUR's members are concerned about the safety of the Comanche Peak Steam Electric Station. While it is apparent that we do not agree on the significance or resolution of some issues, we have attempted to further explain the basis for our resolution of your concerns in the hope that, with that knowledge, you will understand how the NRC has discharged its responsibility to protect the public health and safety.

Sincerely,

J. E. Lyons

James E. Lyons, Chairman
Allegation Review Committee
Comanche Peak Project Division

Enclosure:
CFUR Issues

cc w/enclosure:
See next page

cc w/enclosure:
Mr. Robert F. Warnick
Assistant Director
for Inspection Programs
Comanche Peak Project Division
U. S. Nuclear Regulatory Commission
P. O. Box 1029
Granbury, Texas 76048

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Ms. Billie Pirner Garde, Esq.
Robinson, Robinson, et al.
103 East College Avenue
Appleton, Wisconsin 54911

Mrs. Juanita Ellis, President
Citizens Association for Sound Energy
1426 South Polk
Dallas, Texas 75224

E. F. Ottney
P. O. Box 1777
Glen Rose, Texas 76043

Mr. Roger D. Walker
Manager, Nuclear Licensing
Texas Utilities Electric Company
400 North Olive Street, L. B. 81
Dallas, Texas 75201

Texas Utilities Electric Company
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, Maryland 20814

William A. Burchette, Esq.
Counsel for Tex-La Electric
Cooperative of Texas
Heron, Burchette, Ruckert & Rothwell
1025 Thomas Jefferson Street, NW
Washington, D.C. 20007

GDS ASSOCIATES, INC.
Suite 720
1850 Parkway Place
Marietta, Georgia 30067-8237

Jack R. Newman, Esq.
Newman & Holtzinger
1615 L Street, NW
Suite 1000
Washington, D.C. 20036

Chief, Texas Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756

Honorable George Crump
County Judge
Glen Rose, Texas 76043

Mr. William J. Cahill, Jr.
Executive Vice President
TU Electric
400 North Olive Street, Lock Box 81
Dallas, Texas 75201

CFUR ISSUES

1. Issue

The risk of low power operation is exemplified by problems (including release of radioactive gases) at Ft. St. Vrain, which was also regulated by R-IV. Now, 10 years after startup, it is shutdown forever.

Evaluation

The NRC will not issue any license, not even a low-power license, without reasonable assurance that there is adequate protection of the public health and safety. Nevertheless, there are special considerations to low-power operation. Most importantly, the possible consequences of an accident during low-power operation are limited to a very small fraction of those possible at full power. Low-power operation would generate less than one-twentieth of the radioactive fission products which would be generated at full power. This decrease in fission products also dramatically reduces the amount of decay heat available to damage the core as compared to full power operation. Therefore, accidents at low-power operation would evolve over longer periods than at full-power operation and could be contained by equipment designed to cope with accidents occurring at full power.

CFUR's concern appears to stem from an OIA investigation of Region IV management in 1986 which raised issues related to the inspection activities at Ft. St. Vrain, in addition to Comanche Peak. CFUR implies that inspection policies during the construction of these plants had allowed inherent flaws to go undetected.

The concerns raised by OIA Report 86-10 were reviewed extensively by an NRC staff panel, referred to as the Comanche Peak Report Review Group (CPRRG), and subsequently in an independent investigation by David Williams, then with the Government Accounting Office and now the NRC's Inspector General. The results of those investigations, which were released to the public, concluded that the issues were primarily administrative and did not have any direct adverse impact on plant safety. Those issues were also investigated by Senator Glenn's committee.

The corrective actions that resulted from the CPRRG review, as described in the published report NUREG-1257, included several followup inspections which were assigned to the Comanche Peak Project Division when the Office of Special Projects assumed responsibility for the inspection activities at Comanche Peak in early 1987. The results of those followup inspections similarly did not reveal any evidence that any safety-significant deficiencies have gone undetected. Moreover, a comprehensive review of the design and construction of Comanche Peak has been conducted in conjunction with the Comanche Peak Response Team (CPRT) and Corrective Action Program since 1986. Based on extensive review and inspections, the staff concluded that the CPRT had adequately implemented its investigative activities related to the design, construction, construction quality assurance/quality control, and testing at CPSES. The staff further concluded that the CPRT evaluation of the results of its investigation was thorough and complete and its recommendations for corrective actions were sufficient to resolve identified deficiencies. The staff subsequently concluded in a variety of inspection reports that TU Electric had adequately implemented the hardware validation and final reconciliation portions of their Corrective Action Program.

Ft. St. Vrain is a high-temperature gas-cooled reactor (HTGR). Any difficulties that plant might have had during its startup are more likely due to the uniqueness of the HTGR technology than to NRC inspection practices. Ft. St. Vrain is the only commercial power HTGR in the United States. The decision by Public Service Company of Colorado to decommission Ft. St. Vrain is primarily due to the economics of the HTGR technology and has no bearing on the viability of Comanche Peak or any other light-water reactor as a safe energy source.

More generally, the potential for difficulties during the startup of a nuclear power reactor largely depend on the amount of effort the utility puts into preparedness for plant operation. Considerable attention has been focused on operational readiness because of the Augmented Inspection Team findings. The Operational Readiness Assessment Team inspection will be conducted during the period from January 22 through February 2, 1990 for the purpose of assessing whether Comanche Peak and TU Electric are adequately prepared for plant operation.

2. <u>Issue</u>

The potential for spent fuel accidents is more severe than previously thought, based on a study by BNL dated 2/5/87: BEYOND DESIGN-BASIS ACCIDENTS IN SPENT FUEL POOLS (GI-82). The lack of a high level waste repository will require long term storage of spent fuel at Comanche Peak.

<u>Evaluation</u>

Since the Brookhaven study was issued, the staff and its consultants have performed a more complete analysis of the risks of potential accidents in spent fuel pools and has concluded that the risks are acceptably small. In NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools," (April 1989), the NRC staff determined that the risks of accidents from spent fuel storage are dominated by seismic impacts on the structural integrity of the spent fuel pool, that the risks and consequences of such accidents appear to meet the Commission's Safety Goal Policy Statement, and that the risks are no greater than associated with the risks from core damage accidents. The NRC staff also concluded that the alternative measures for reducing the risks were not warranted in light of the costs of the alternatives and the large inherent safety margins in the design and construction of spent fuel pools.

The risk that the CPSES spent fuel pools will not have sufficient storage capacity is an economic risk only, not a safety risk. The CPSES spent fuel pools meet the minimum design capacity guidelines for a dual shared facility of one full core discharge plus two normal fuel discharge cycles (322 fuel assemblies for CPSES) as set forth in ANS 57.2. The CPSES Technical Specifications, which will be a part of the license, limit the storage capacity to no more than 1166 fuel assemblies as is currently designed. Any future changes to the storage capacity will require a license amendment and the attendant opportunity for a hearing. However, it should be noted that the Commission has determined that spent fuel pool modifications using previously approved methods involve a no significant hazard consideration as defined in 10 CFR 50.92 and, therefore, do not require that a hearing be held prior to issuance of the amendment.

The Commission addressed the issue of long term storage of spent fuel in its August 31, 1984 Waste Confidence Decision. Currently, 10 CFR 51.23 states in part:

> The Commission has made a generic determination that for at least 30 years beyond the expiration of reactor operating licenses no significant environmental impacts will result from the storage of spent fuel in reactor facility storage pools or independent spent fuel storage installations located at reactor or away-from-reactor sites.

The background discussion from the review and proposed revision of the Waste Confidence Decision and a conforming amendment to 10 CFR Part 51, which was published in the Federal Register on September 28, 1989, (Attachment 1) describes the actions taken to date by the Commission. The proposed revision to the Waste Confidence Decision reaffirms and supplements the 1984 findings and the environmental analyses supporting them.

3. Issue

Check Valve failures that occurred during hot functional testing in April and May 1989 were critical and would have contaminated systems outside containment. TU Electric's response to the check valve failures was inadequate, according to the NRC's July 10, 1989 report. Additional Borg-Warner check valve problems have been identified by the NRC since initial failures in April and May.

Evaluation

As stated in the December 7, 1989 meeting, CFUR's concerns were derived from the findings in the NRC's Augmented Inspection Team's (AIT) report and subsequent NRC inspection reports and letters regarding the check valve failures. The NRC review of Borg-Warner check valve issues is still in progress. Previous inspections related to this topic are documented in NRC Inspection Reports 50-445/89-30, 50-446/89-30; 50-445/89-52, 50-446/89-52; 50-445/89-64, 50-446/89-64; 50-445/89-71, 50-446/89-71; 50-445/89-73, 50-446/89-73; 50-445/89-84, 50-446/89-84; and 50-445/89-88, 50-446/89-88.

The NRC staff has concluded that the applicant's corrective action program to reset and control the bonnet elevation of Borg-Warner check valves will effectively prevent the previously observed phenomenon where the valve disk jammed under the seat ring. Although some problems have been encountered in the implementation of these corrective actions, the applicant's commitment to conduct a functional backflow test and/or radiographic examination for each valve will provide reasonable assurance that all Borg-Warner check valves are capable of performing their design function.

In NRC Inspection Report 50-445/89-73, 50-446/89-73 (Attachment 2), the NRC identified 14 open items regarding various issues stemming from the AFW backflow events. To date, two of these open items have been closed as documented in NRC Inspection Reports 50-445/89-84, 50-446/89-84 and 50-445/89-88, 50-446/89-88 (Attachments 3 and 4). All open items will be closed out prior to licensing and the closeouts will be documented in NRC Inspection Report 50-445/90-03, 50-446/90-03 and subsequent reports.

In addition to the open items, the NRC has issued an enforcement action, EA-89-219 dated January 25, 1990 (Attachment 5). That action is being taken to emphasize the importance of the lessons learned from the check valve failure events.

An issue not raised in the Stay Request, but in CFUR's subsequent November 8, 1989 letter to the NRC, was that the NRC had identified additional Borg-Warner check valve problems since the initial failures in April and May. TU Electric reported the failure of a swing arm in a Borg-Warner check valve installed in the service water system. As the result of discovering the failed swing arm, the NRC staff is reviewing the service suitability of the Borg-Warner check valve swing arms. The applicant, along with its consultant, Aptech, conducted an extensive series of nondestructive tests on the swing arms to identify and replace the discrepant swing arms. An extensive engineering analysis was performed to demonstrate the acceptability of those swing arms which were not replaced. That analysis is now under review and the NRC will ensure that the check valves operate properly prior to making a decision on a Unit 1 fuel load license.

The AIT report indicated that, during the check valve failure events, operations personnel failed to effectively recognize and act on conditions adverse to quality. The staff's concerns regarding those findings are described in the subsequent enforcement action (EA-89-219). However, we consider the significance of these findings related to TU Electric's transition from construction activities to an operational environment.

In that regard, we will rely on the NRC's Operational Readiness Assessment Team to assess whether TU Electric's corrective actions, in response to the AIT findings, have been effective.

4. Issue

Counterfeit bolts have been used throughout the plant. Substandard material may also have been procured from the Meredith Company.

CFUR requested information regarding Meredith Company.

Evaluation

As discussed in the December 7, 1989 meeting, CFUR's concerns were derived from the findings in NRC inspection reports and letters on counterfeit materials. The NRC has taken a number of generic short-term and long-term measures to provide assurance that NRC licensees do not install counterfeit equipment and materials in their plants. In May 1989, the NRC issued Generic Letter 89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products, which described to the nuclear industry, those characteristics of effective procurement and dedication programs. Generic Letter 89-02 provided NRC's conditional endorsement of an industry standard for dedication programs which evaluate the suitability of commercial grade products for use in safety-related applications. Also in March 1989, the NRC issued an Advance Notice of Proposed Rulemaking soliciting public comment on whether or how NRC regulations should be revised to provide increased assurance that counterfeit or misrepresented vendor products are not installed in nuclear plants. Over 60 commenters provided responses to the NRC on the proposed rulemaking and the staff is currently evaluating the public comments.

In addition to the short-term measure (Generic Letter) and the long-term measure (rulemaking), the NRC inspection and investigative staff have been very aggressive in pursuing instances of suspected counterfeit or misrepresentation by vendors. These efforts are directed to keep the industry fully informed so that appropriate licensee corrective actions can be taken and to assure that appropriate enforcement and investigative actions against the vendors are also taken.

During the past two years, the NRC has issued over 25 Bulletins, Information Notices and Supplements to alert the nuclear industry of suspected misrepresentation by vendors and the staff has provided support to the Department of Justice's review of vendors suspected of wrongdoing.

The NRC recognizes that vendor misrepresentation is not a problem unique to the nuclear industry in that counterfeiting and fraud can and do occur in other industries. To assure that other Federal agencies are informed of instances of vendor misrepresentation identified by the NRC, copies of NRC's Bulletins and Information Notices are forwarded to other

Attachment H

primarily located at the plant site. In addition, most of the senior management personnel, including the Chief Engineer and the Directors of Quality Assurance, Construction and Management Services, as well as all of the managers and supervisors in Nuclear Operations are located at the plant. By being at the plant site, we are directly involved in the day-to-day management of plant activities and are able to implement the hands on management approach. In addition, we're readily available to our managers and supervisors to address any issues or concerns as well as to provide a visible leadership.

As you are aware, during hot functional testing, deficiencies were identified related to check valve backflow and out of sequence performance of a step in a test. TU Electric, as well as the NRC, conducted extensive evaluation to determine the causes and corrective action to resolve these deficiencies.

(Slide) We are implementing the corrective actions and the post modification testing which assure us that these check valves function as designed. In addition, maintenance procedures have been modified and personnel have received additional training to preclude recurrence. Administrative procedures have also been revised to clearly state that the tasks in

Attachment
I

TU ELECTRIC

Log # TXX-89596
File # 10130
IR 89-30
IR 89-30

August 18, 1989

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 NOS. 50-445 AND 50-446
RESPONSE TO NRC INSPECTION REPORT NOS.
50-445/89-30; 50-446/89-30
AUGMENTED INSPECTION TEAM (AIT) INSPECTION
OF CHECK VALVE FAILURES

REF: 1) Letter from R. F. Warnick, USNRC, to W. J. Cahill, TU Electric dated July 10, 1989

2) TU Electric letter logged TXX-89492, W. J. Cahill to USNRC dated July 24, 1989

Gentlemen:

Reference 1 requested that TU Electric submit a report summarizing the lessons learned from the Auxiliary Feedwater (AFW) backflow events on April 23 and May 5, 1989 and the corrective actions TU Electric planned to take. Reference 2 acknowledged the Reference 1 request and stated a report would be submitted by August 18, 1989. The report is attached.

As this report discusses, the cause of the backflow events was backflow through hung open Borg-Warner/International Pump Inc. (BW/IP) pressure seal check valves coincident with the failure of Auxiliary Operators to operate valves in the sequence specified by procedures. In response to these events, TU Electric is taking corrective action for the affected hardware, including inspection and, as necessary, rework of BW/IP check valves at CPSES. Additionally, TU Electric is taking action to address the cause of the events and prevent recurrence of similar events. These actions include the following:

- The reassembly procedure for the BW/IP check valves has been revised to ensure that the valve disc will properly seat.
- Administrative procedures have been revised to clearly state that the tasks in a procedure are to be performed in the sequence specified unless certain exceptions are satisfied.

IE01

~~8908240338~~

- o Operations personnel are receiving and will continue to receive training in the revised administrative procedures, in the need to comply with procedures in general, and in avoidance of the type of noncompliances with procedures that occurred during the April 23 and May 5 events.

TU Electric has also evaluated the backflow events on April 23 and May 5, the precursors to these events, and the Company's response to these events to determine lessons learned and identify corresponding improvements. In performing this evaluation, TU Electric also accounted for the conclusions and recommendations of the AIT, together with the weaknesses identified by the NRC at a meeting on the CPSES power ascension program on July 17, 1989.

Based upon its evaluation, TU Electric has concluded that improvements are warranted in four general areas before fuel load. These areas, and the corresponding improvements that TU Electric is making, are discussed below. The attached report provides a more detailed description of the improvements.

- o <u>Management and Supervision of Operations</u> - TU Electric is taking action to expedite the transition from a construction to an operating attitude, to provide Operations with greater control of the project, to improve the reporting of plant events and equipment failures to Operations management and supervision, and to enhance management's awareness of time and manpower needs for specific tasks.

- o <u>Corrective Actions and Evaluation of Plant Events and Equipment Failures</u> - TU Electric is taking action to improve the documentation and reporting of plant events and equipment failures, to increase the aggressiveness and timeliness of investigations of plant events and equipment failures, and to improve future team evaluations by TU Electric.

- o <u>Communications Among Operators and Shifts</u> - TU Electric is taking action to improve communications among operators and communications between shifts.

- o <u>Personnel Awareness of Operating Events and Equipment Failures and Their Implications for System Operability</u> - TU Electric is taking action to increase the awareness of Operations personnel concerning Work Requests and their implications for plant operability, and to improve the availability of information regarding plant events anu equipment failures to Operations personnel.

The improvements discussed above are only one part of a larger effort to ensure that TU Electric will be ready to operate CPSES Unit 1. For example, TU Electric has established an Operational Readiness Program and management of the transition from construction to operations has been placed under the direction of the Vice President, Nuclear Operations. These and other efforts, together with the improvements discussed in the attached report, will help ensure that TU Electric will be ready to operate upon completion of Unit 1 construction.

In summary, TU Electric has identified the root causes of the backflow events on April 23 and May 5, 1989, is taking corrective action for these events, including action to address root causes and prevent recurrence of similar events, and is implementing improvements based on lessons learned. Consequently, TU Electric believes that it is adequately addressing the events, and that upon completion of the corrective and preventive actions and implementation of referenced improvements, the events should not pose any impediment to the issuance of an operating license for CPSES Unit 1.

Sincerely,

William J Cahill Jr.

William J. Cahill, Jr.

TLH:daj

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

TU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION

DOCKET NOS. 50-445 AND 50-446

REPORT ON EVENTS OF APRIL 23 AND MAY 5, 1989
INVOLVING BACKFLOW THROUGH THE
AUXILIARY FEEDWATER SYSTEM

AUGUST 18, 1989

TABLE OF CONTENTS

I. EXECUTIVE SUMMARY

A. Introduction and Purpose

On April 23 and again on May 5, 1989, during hot functional testing of Comanche Peak Steam Electric Station (CPSES) Unit 1, backflow occurred from some of the steam generators through portions of the Auxiliary Feedwater (AFW) System to the Condensate Storage Tank (CST). This event happened because of hung open check valves and coincident failure of Auxiliary Operators (AOs) to operate manual valves in the sequence specified by procedures. Texas Utilities Electric Company (TU Electric) established a Task Team to investigate these events. The NRC also established an Augmented Inspection Team (AIT). This report discusses the TU Electric Task Team investigation results, and the operational weaknesses identified by the AIT Report and by the NRC in a meeting on July 17, 1989.

B. Description of the April 23 and May 5 Events

On April 23, 1989, a partial blowdown of steam generators 1, 2 and 4 occurred through AFW System lines to the CST. The event occurred while realigning valves following a preoperational test. An Auxiliary Operator (AO) began to open an AFW pump test isolation valve while another AO was closing the pump discharge valve. Operation of these two valves at the same time is not in accordance with the approved procedure which requires sequential valve operation. This operation coincident with hung open check valves created the backflow path from the steam generators to the CST. When the test isolation was fully closed, the backflow stopped and steam generator levels stabilized. Backflow occurred for approximately fifteen to twenty minutes.

On May 5, 1989, a partial blowdown of steam generators 1 and 3 also occurred through AFW System lines to the CST. This event occurred while aligning the system to perform an operability test. An AO began to close an AFW pump discharge valve while the pump test isolation valve was being opened. This violated the approved procedure which requires sequential valve operation. This operation coincident with check valve failures created a backflow path from the steam generators to the CST. Backflow occurred for approximately twelve minutes and was stopped when the pump discharge valve was closed.

Subsequently, the Reactor Operator (RO) directed an AO to close the pump test isolation valve; however, it was inadvertently left one-quarter turn open due to mechanical binding. This alignment re-initiated backflow through the system. Eventually, the pump test isolation valve was completely closed. Backflow occurred for approximately sixty-six minutes primarily due to the time required to identify that the test isolation valve was partially open.

C. TU Electric Investigation of Root Causes

To assure a thorough investigation of the April 23 event the Executive Vice President for Nuclear Engineering and Operations established a multi-disciplined Task Team. TU Electric senior management emphasized that the Task Team should concentrate on a thorough and deliberate determination of root causes. Based upon its reviews, the Task Team determined that certain check valves in the AFW System had become hung open due to the discs becoming lodged beneath the seat lip (see Figure 1). The condition resulted from an elevation difference between the valve seat and disc created by incorrect reassembly instructions.

The Task Team interviewed Operations personnel and reviewed available information to determine the root causes of the operator errors on April 23 and May 5. Investigation determined that valves were operated simultaneously due to a misunderstanding of the administrative controls governing the sequencing of procedure steps. In part, this lack of understanding was attributable to the absence of guidance in applicable Operations Department Administrative Procedures (ODAs).

The Task Team interviewed operators to determine why the isolation valve was not fully closed on May 5. The Team determined that the operators believed the valve to be closed because of the resistance felt in closing the valve. The AOs were unable to visually determine the degree of valve closure because of the location of the valve with respect to its operator.

D. Significance of the April 23 and May 5 Events

These events did not and could not have resulted in any radioactive release because they occurred during preoperational testing and prior to fuel load. Therefore, the events did not pose any threat to public health and safety.

The Task Team evaluated the impact of the backflow on piping and support integrity, containment penetrations, and instrumentation. Analysis identified several areas where piping Code allowable stresses were exceeded. Subsequent Ultrasonic Testing (UT) of the pipe verified that no plastic deformation had occurred. Thus, the stresses resulting from the elevated temperature were within the elastic range for the piping material and no piping needs to be replaced. One pipe support was visibly damaged and has been replaced. Additionally, analysis determined that ten supports were overloaded. Nonconformance Reports were written to require QC examination of the significant attributes of these supports and no deviations or deficiencies were found. Finally, containment penetrations were determined to be unaffected by the events. The impact on some flow transmitters is still under evaluation.

The TU Electric Task Team performed an evaluation of the potential effects of a similar malfunction of BW/IP check valves during plant operation. In the absence of a line break or a manual valve misoperation, the failure of the valves lacks significance because of the absence of a backflow path.

In the event of a loss of AFW flow to the steam generators for any reason the Emergency Response Guidelines would require operator actions to commence Reactor Coolant System (RCS) cooldown using systems other than the AFW and Feedwater Systems.

E. Precursors

The Task Team reviewed industry experience with check valves and previous check valve problem at CPSES to determine whether other check valves experienced the same failure mode as the check valves involved in the April 23 and May 5 events.

Although various concerns about the performance of check valves have been experienced by the industry, there was no generally available information prior to the April 23 event that the BW/IP check valves were likely to malfunction due to an elevation difference between the valve disc and seat.

The Task Team did identify check valves at CPSES that may have failed in a manner similar to those on April 23. These failures occurred in 1985, and on April 5 and 19, 1989. TU Electric has concluded that the existence of these failures indicate that improvements are warranted in the documentation, reporting, and evaluation of plant events and equipment failures. As discussed below, TU Electric is implementing improvements in these areas.

F. Corrective Actions, Preventive Actions, Lessons Learned, and Associated Improvements

TU Electric is taking corrective actions for affected hardware, including the following: 1) BW/IP pressure bonnet check valves are being inspected and the discs and seats aligned as necessary, and post modification operability tests are being performed to verify that the valves are fully closed; 2) affected piping is being repainted, the damaged pipe support has been replaced, and other pipe supports have been inspected with no deviations or deficiencies found; 3) potentially affected flow transmitters will be recalibrated and replaced if necessary; and 4) binding of the isolation valve will be evaluated and corrected.

TU Electric is taking actions to address the root causes and prevent recurrence of events similar to those on April 23 and May 5, including the following:

- o The valve reassembly procedure has been revised to include a requirement for determining the elevation adjustment necessary to avoid interference between the disc and the seat.

- o The Operations administrative procedure which provides guidelines on the use of procedures has been revised to emphasize the requirement that procedure steps are to be performed in sequence unless specific exceptions are satisfied.

- o The Shift Operations Manager has developed and is implementing an action plan to enhance procedural compliance. The need to complete procedural steps sequentially will continue to be emphasized and will become part of Operator Requalification and Replacement Training.

- o Reach rod operators for safety-related valves will be evaluated for proper operability and human factors considerations.

In addition, TU Electric has identified a number of areas where improvements could be made. These areas are discussed below.

- o Management and Supervision of Operations - TU Electric is taking action to expedite the transition from a construction to an operating attitude, to provide Operations with greater control of the project, to improve the reporting of plant events and equipment failures to Operations management and supervision, and to enhance management's awareness of manpower needs for specific tasks.

- o Corrective Actions and Evaluation of Plant Events and Equipment Failures - TU Electric is taking action to improve the documentation and reporting of plant events and equipment failures, to increase the aggressiveness and timeliness of investigations of plant events and equipment failures, and to improve future team evaluations by TU Electric.

- o Communications Among Operators and Shifts - TU Electric is taking action to improve communications among operators and communications between shifts.

- o Personnel Awareness of Operating Events and Equipment Failures and Their Implications for System Operability - TU Electric is taking action to increase the awareness of

Operations personnel to Work Requests and their implications for plant operability, and to improve the availability of information regarding plant events and equipment failures to Operations personnel.

The specific improvements that TU Electric is making in each of these areas is discussed in detail in Section VII.C of this report. NRC concerns as identified in the AIT Report and during the July 17, 1989 meeting are discussed in Appendix 1.

G. Summary and Conclusions

The April 23 and May 5 events were of no immediate safety significance because there was no fuel in the reactor and Unit 1 was not radioactive. A similar event during operation coupled with a steam line or AFW line break could have resulted in loss of AFW. Operator action in accordance with procedures would have maintained the reactor in a safe condition. TU Electric is taking corrective action for the deficiencies in the hardware identified by these events. Additionally, TU Electric is taking action to address the root causes of the events and to prevent recurrence of similar events. Finally, TU Electric has identified lessons learned from these events and is taking actions to improve the management and supervision of Operations personnel, to improve corrective actions for plant events and equipment failures, to improve communications among Operations personnel and between shifts, and to improve personnel awareness of operating events and equipment failures and their implications for system operability.

II. INTRODUCTION AND PURPOSE

Comanche Peak Steam Electric Station (CPSES) is a two-unit Westinghouse pressurized water reactor (PWR) owned by Texas Utilities Electric Company (TU Electric). During hot functional testing of CPSES Unit 1 on April 23 and May 5, 1989, backflow occurred from some of the steam generators through portions of the Auxiliary Feedwater (AFW) System to the Condensate Storage Tank (CST) because of hung open check valves, coincident operator error and, on May 5, mechanical binding of an isolation valve.

The NRC issued a Notice of Violation to TU Electric on May 18, 1989, based in part on the April 23 event. Additionally, on May 5, 1989, the NRC issued a Confirmation of Action Letter (CAL) which confirmed that certain actions would be taken by TU Electric in response to the events and which provided for an NRC investigation of these events by an Augmented Inspection Team (AIT).

The results of the AIT investigation were provided in a letter to TU Electric on July 10, 1989. The letter described several operational weaknesses identified by the AIT during its investigation. Additionally, in a meeting at Rockville, MD on July 17, 1989, the NRC identif[illegible] similar [illegible]perational weaknesses resulting from the backflow events a[illegible] recent NRC violations.

TU Electric informed the Nuclear Regulatory Commission (NRC) of these events on April 24 and May 6, respectively. Additionally, TU Electric established a Task Team on May 1, 1989 to investigate the causes and significance of these events and to recommend corrective actions. Based on the results of those investigations, TU Electric determined the events were potentially reportable under 10CFR50.55(e), notified the NRC on May 19, 1989, and provided an interim report to the NRC on June 19, 1989 which categorized the events as reportable (see SDAR CP-89-15, TXX-89429). Two INPO Nuclear Network Notices were issued by TU Electric on May 17 and May 24, 1989. The May 17 Notice generally described check valve backleakage. The May 24 Notice questioned industry contacts concerning check valve backleakage due to mechanisms other than valve distortion, debris or normal wear. To date no responses have been received. Additionally, on June 1, 1989, TU Electric notified BW/IP of the defects that existed in its check valves and indicated that they may be potentially reportable under 10CFR21.

This report discusses the results of the investigation of the April 23 and May 5 events by the TU Electric Task Team, responds to the NRC's July 10 letter, and addresses the operational weaknesses identified by TU Electric and by the NRC at the meeting on July 17, 1989. The remainder of this report is divided into the following sections:

- o Section III provides a description of the events on April 23 and May 5.

- Section IV describes the investigations performed by the TU Electric Task Team and summarizes the results of the investigations, including identification of the causes of the events on April 23 and May 5.

- Section V discusses the significance of the events on April 23 and May 5.

- Section VI describes prior deficiencies involving BW/IP check valves at CPSES and other plants, and discusses the relevance of these deficiencies to the events on April 23 and May 5.

- Section VII discusses TU Electric's corrective and preventive actions for the April 23 and May 5 events, the lessons learned from these events, and improvements being made by TU Electric. This section also addresses the weaknesses identified by the NRC in its July 10 letter and at the meeting on July 17, 1989.

- Section VIII presents TU Electric's conclusions as a result of these events.

- The six appendices provide additional information on TU Electric's response to NRC concerns; check valve backleakage testing; check valve maintenance history; two unrelated material deficiencies relevant to BW/IP check valves; Task Team inspection techniques; and evaluation of AFW check valves against EPRI Guidelines, respectively.

III. DESCRIPTION OF THE APRIL 23 AND MAY 5 EVENTS

A. Description of the April 23 Event

On April 23, 1989, a partial blowdown of steam generators 1, 2 and 4 occurred through AFW System lines to the CST. This blowdown created abnormally high temperatures in system piping (greater than 200°F in AFW System piping and approximately 500°F in Feedwater System piping) and reduced water levels in the three steam generators approximately 12% of the narrow range indication in 15 to 20 minutes. The event caused blistering and discoloration of the paint on the TDAFWP discharge piping.

Prior to the event the plant conditions were as follows:

1. Reactor Coolant System (RCS) pressure control was in automatic
2. RCS pressure was 2235 psig
3. RCS temperature was 557°F
4. Steam Dump control was in automatic
5. Steam generator pressure was 1100 psig
6. All Main Steam Isolation Valves (MSIVs) were open
7. Total steam generator blowdown flow was 45 gpm
8. Motor Driven Auxiliary Feedwater Pump (MDAFWP) 2 was in operation with a flowrate of 120 gpm
9. No fuel was in the reactor

The event occurred while realigning Turbine Driven Auxiliary Feedwater Pump (TDAFWP) valves following a preoperational test. The TDAFWP flow control valves were fully open and the motor operated isolation valves were throttled and deenergized. The TDAFWP was started to provide flow to the steam generators for three minutes and was then tripped from the Control Room in anticipation of realigning it to the test header for a three hour run to perform a hot alignment check.

The Reactor Operator (RO) used approved procedures to realign and run the TDAFWP to the test header. He briefed the Safeguards Building Auxiliary Operator (AO) and then sent him to close valve 1AF-041 (TDAFWP DISCH ISOL), and open valve 1AF-042 (TDAFWP TST ISOL). Upon reaching the TDAFWP room the AO first opened valve 1AF-042 approximately 1/4 of a turn. He then proceeded to close 1AF-041. The AO turned the valve operator on 1AF-042 one-quarter turn because the open/close direction tags were missing on valve 1AF-041, and he wanted to verify the proper rotation to open the valve. He did not realize that turning the operator this small amount could unseat the valve. The AO then requested and was provided assistance to operate these valves. When three other AOs arrived at the TDAFWP room, one AO began to fully open 1AF-042 and another AO relieved the Safeguards Building AO and continued to close 1AF-041.

Operation of these two valves at the same time is not in accordance with the approved procedure which requires that 1AF-041 be closed before 1AF-042 is opened. This operation, coincident with hung open check valves and the unseating of a Feedwater Isolation Bypass Valve, FIBV, (which is not intended to prevent backflow at pressures greater than containment design pressure) created an open backflow path from the steam generators to the CST (see Figure 2).

The RO noticed that steam generator water levels were decreasing as the valves were being operated. The RO increased MDAFWP 2 discharge flow to 400 gpm and noticed that only steam generator 3 was receiving flow at approximately 20 gpm. Recognizing that a potential backflow condition may exist, the RO directed the Safeguards Building AO to verify that valve 1AF-055 (MDAFWP 02 TST ISOL) was closed. The Safeguards Building AO reported back that 1AF-055 was closed, but also stated that the paint on the TDAFWP discharge piping was bubbling. The RO then told the AO to close 1AF-042. When 1AF-042 was fully closed, the backflow stopped and steam generator levels stabilized. Backflow occurred for approximately fifteen to twenty minutes.

B. Description of the May 5, 1989 Event

On May 5, 1989, a partial blowdown of steam generators 1 and 3 occurred through AFW check valves and lines to the CST. The blowdown caused paint discoloration of the MDAFWP 1 discharge piping to steam generator 1 and TDAFWP discharge lines to steam generators 1 and 4. An estimated 20% of narrow range level in steam generator 1 was displaced through the lower feedwater nozzle into the AFW System, while an estimated 11% of narrow range level in steam generator 3 was displaced from the lower feedwater nozzle. Steam generator 3 did not blowdown sufficiently to cause hot water to reach AFW piping and discolor paint on the AFW line to steam generator 3.

Prior to the event, the plant conditions were as follows:

1. RCS pressure control was in automatic
2. RCS pressure was 2235 psig
3. RCS temperature was 557°F
4. Steam Dump control was in automatic
5. Steam generator pressure was 1100 psig
6. Steam generator blowdown was isolated
7. All MSIVs were open
8. All AFW pumps were shutdown
9. No fuel was in the reactor

The event occurred while aligning the system to perform an Auxiliary Feedwater operability test to familiarize Operations personnel with Operation Test Procedures (OPT). Both of the MDAFWPs had been stopped for system realignment. An AO and the Shift Technical Advisor (STA) were sent from the Control Room to begin the system alignment. After arriving at the MDAFWP 2 room, the AO began to close 1AF-054 (MDAFWP 2 ISOL), and requested assistance in manipulating the remainder of the AFW valves. including valve 1AF-055. When an AO arrived to provide the requested assistance, the valves were operated at the same time. This violated approved procedures, which required that 1AF-054 be closed before 1AF-055 is opened. This operation, coincident with hung open check valves and the unseating of the FIBV, created an open backflow path from the steam generators to the CST (see Figure 3). Backflow occurred for approximately twelve minutes until 1AF-054 was closed.

After closing 1AF-054, the AO verified the pump cross connect valves were closed. MDAFWP 2 was then started and data collected in accordance with the OPT. The pump was stopped and the AO was instructed to realign the AFW System to increase steam generator levels. The AO opened cross-connect valves 1AF-090 and 1AF-091 and attempted to close valve 1AF-055. However, 1AF-055 was inadvertently left one-quarter turn open (the AO thought it was fully closed, but the valve was mechanically bound). This alignment re-initiated backflow through the system when MDAFWP 2 discharge valve 1AF-054 was re-opened. When the AO notified the RO that the lineup was restored, the RO started MDAFWP 2 to supply flow to all steam generators. The RO watched water levels in the steam generators for approximately twenty minutes and determined that levels were not responding correctly for a pump discharge flow rate of 300 gpm. Suspecting a problem with MDAFWP 2, the RO stopped MDAFWP 2 and directed the AO to prepare MDAFWP 1 for starting. MDAFWP 1 was started and the identical response was observed for pump flow rate and steam generator levels. The RO stopped MDAFWP 1 and told the AO to close the AFW cross connect valves, 1AF-090 and 1AF-091. The RO then started MDAFWP 1 and 2 to supply the steam generators and observed that levels were not increasing as he expected. Suspecting a backflow path existed, the Unit Supervisor went to the MDAFWP 2 room and helped the AO fully close valve 1AF-055. The RO was then able to restore normal flow to the steam generators and observed the correct level response. Backflow for this portion of the event occurred for approximately sixty-six minutes.

program, in conjunction with a review of operator logs, established the backflow paths for the April 23 and May 5 events. Identification of these flow paths provided a basis for identification and subsequent engineering evaluation of potentially overstressed piping, supports, containment penetrations and instrumentation. Appendix 2 discusses the results of this testing program. As this Appendix indicates, numerous check valves were determined to be hung open, indicating a generic problem.

The Task Team conducted reviews to determine the cause of the backflow through the AFW System. The results of the reviews were as follows:

- The Task Team reviewed the maintenance and modification histories of the check valves to determine if any shortcomings could have resulted in the check valve failures. The results of this review are presented in Appendix 3. As this Appendix discusses, prior disassembly and reassembly of various BW/IP check valves in 1983 produced an elevation difference between the valve seat and disc due to incorrect reassembly instructions. The instruction stated that the valve retainer, which locates the disc assembly, was to be bottomed out. This technique created the aforementioned elevation difference. As discussed below, the inadequacy in the reassembly instruction only pertains to pressure seal check valves, of which there are fifty-seven in Unit 1 and 2. One-hundred-three bolted bonnet valves were unaffected because their design is such that a fixed vertical relationship exists between the seat/disc assembly and seat ring.

- The Task Team used radiography to determine disc position prior to disassembly and the Computer Assisted Drawing (CAD) program to determine the actual measurements of critical valve internal components. The results of the radiographs, the CAD program, and inspections of BW/IP valves are discussed in Appendix 5. Based upon these results, the Task Team determined that the valve discs for pressure seal check valves had become hung open due to the discs becoming lodged beneath the seat lip. In addition, the Task Team learned that available vendor information did not specify maximum disc axial play. Excessive axial play coincident with seat/disc elevation differences is viewed as a contributory factor to valve failure.

- The Task Team reviewed available industry experience with check valves in other nuclear plants to determine whether these plants may have identified a problem with BW/IP check valves that could have caused the backleakage on April 23 and

May 5. As discussed in Section VI.A, the Task Team did not identify problems related to the backflow events at CPSES from available industry information. Related information obtained during the course of the investigation of the events had not previously been identified to industry

- The Task Team reviewed previous problems with check valves at CPSES to determine whether these problems and any common causes were present. As discussed in Section VI.B previous failures of check valves at CPSES had occurred, which may have failed in a manner similar to those of April 23 and May 5, indicating a need for improvement in the documentation, reporting, and evaluation of plant events and equipment failures.

- Coincident with the Task Team investigation, two potentially significant material conditions in BW/IP check valves were identified during Station Service Water System testing. These conditions are unrelated to check valve backleakage, but are discussed in Appendix 4.

- The Task Team evaluated the design of the AFW check valves using guidance issued by the Electric Power Research Institute (EPRI). As discussed in Appendix 6, this evaluation did not identify any factor that would relate to the cause of the hang up of the AFW check valves on April 23 and May 5.

D. Investigation of the Root Causes of the Operator Errors

Operations personnel under direction of the Manager, Operations conducted interviews with shift crews to determine the root cause of operator errors made during the April 23 event.

As discussed in Section III, the event occurred following the simultaneous operation of valves 1AF-041 and 1AF-042. Investigation determined that valves 1AF-041 and 1AF-042 were operated simultaneously due to a misunderstanding of the administrative controls governing the sequencing of procedure steps. In addition, the valve operator arrangements are unique.

The RO referenced the approved procedure and correct section for instructions on realigning the TDAFWP to the test header. This section of the procedure indicates that closure of 1AF-041 is Task 1 and opening 1AF-042 is Task 2. These tasks are numbered in sequence which requires they be performed in sequence. The requirement that Tasks 1 and 2 be performed in sequence was not fully understood or followed by the Auxiliary Operators. In part, this lack of understanding is attributable to the absence of guidance in applicable Operations Department Administrative Procedures (ODAs). The ODA that provides guidelines for the preparation and review of operations procedures states that mandatory sequence of steps is assumed unless the steps are identified by bullets or unless the procedure states otherwise; however, another ODA that describes guidelines on use of procedures does not identify this rule. The Control Room and Auxiliary Operators are not responsible for the preparation of Operating Procedures and therefore are not as familiar with the requirements of the former procedure as they are of the requirements of the latter procedure.

The valve operator for 1AF-041 is mounted on the TDAFWP room floor and is connected to the valve stem by a series of reach rods and gear boxes. The arrangement of the valve operator and gears causes 1AF-041 to be a reverse operating valve, clockwise to open and counter-clockwise to close. Due to the uniqueness of the valve operator, the handwheel for 1AF-041 is normally labeled to identify the closed direction. Upon arriving at the TDAFWP room, the AO discovered that the direction label was missing and he was unsure of the proper rotation for closing the valve. To determine the proper rotation, the AO took the valve operator for 1AF-042, which was labeled in the open direction, approximately 1/4 turn and observed the movement of its gear box. This allowed the AO to determine the proper direction of handwheel movement for 1AF-041. The AO knew that 1AF-041 required approximately 40 minutes to operate due to the number of turns to full stroke and requested assistance. The AO believed that 1AF-042 would also require approximately 40 minutes to full stroke and did not think that the 1/4 turn on the handwheel would have unseated the valve. Three additional AOs were dispatched to the TDAFWP room to assist. The dispatched AOs were not adequately briefed on the evolution in progress and upon arrival, one relieved the operator closing 1AF-041 and another began to fully open 1AF-042. This resulted in both valves being open at the same time.

The Task Team also conducted interviews to determine the cause of the personnel errors related to the May 5 event. As in the April 23 event, the investigation revealed that the system was aligned by AOs using the approved procedure and correct section. However, Task 1 and Task 2 were not performed in the sequence specified in the procedure because the AOs did not fully understand the requirement to perform these tasks in sequence.

The Task Team interviewed operators to determine why 1AF-055 was not fully closed on May 5. The Team determined that valve 1AF-055 was a remote manually operated valve, that the AOs believed the valve to be closed because of the resistance felt in trying to close the valve, and that the AOs were unable to visually determine the degree of valve closure because of the location of the valve with respect to the operator. This valve binding caused the overall duration of the May 5 event to be much greater than the April 23 event.

The Task Team determined that the operators quickly identified the cause of the April 23 event to be parallel operation of valves. Considering that the operators on shift during the May 5 event were unaware of the April 23 event and were also dealing with an unknown problem, valve binding, the actions were considered timely and investigated logically.

In summary, based upon its interviews of Operations personnel and review of documents, the Task Team determined that operators aligned the AFW valves in the wrong sequence on both April 23 and May 5. This was due to the failure to follow procedures, caused by a lack of understanding of the administrative requirements to perform procedure tasks in the sequence specified in the procedure. Contributing to this problem was confusion over which way the valve was to be turned. Additionally, the failure of the operators to completely close valve 1AF-055 on May 5 was due to its mechanical binding and the inability to readily verify closure. Operator actions and investigations for both events were considered timely and logical.

In its report (pp. 31,35), the AIT states that the AOs did not comply with procedures governing manipulation of the valves because they believed that they could rely upon the check valves to prevent backflow through the AFW System, and because they were under time constraints to complete the valve alignment prior to the end of the shift. The Task Team determined that the AOs involved in the April 23 and May 5 events were not specifically provided with any directions to complete the valve alignment by the end of shift, but they may have taken it upon themselves to do so. The Task Team could not confirm that the operators relied on the check valves to seat when operating valves in parallel. The Task Team concluded that the primary cause of procedural noncompliance was a lack of understanding of the requirement to perform procedure steps in the sequence specified in the procedure.

E. Summary

Based on the above investigation and actions, the Task Team identified the following root causes of the April 23 and May 5 events:

o Check valve failure occurred because of incorrect instructions for reassembly. The incorrect instructions, derived from vendor information, are applicable to pressure seal type valves only. When followed, these instructions created the potential for an unacceptable elevation difference between the valve seat and disc which caused the valve disc to become lodged beneath the seat lip. In addition, disc axial play had not been previously specified

Although proximity between the check valves and upstream fittings and devices was not a cause of the AFW check valve leakage on April 23 and May 5, it was a factor in the backleakage through valve 1AF-069 on April 19, 1989. As discussed in Section VII.B.1, TU Electric will evaluate whether to increase the distance between check valves and upstream orifices based upon an evaluation being performed by Kalsi, Inc.

[3]The valve was not radiographed prior to disassembly, but other indications, such as the pump suction relief valve lifting, indicated the disc was most likely open.

by the vendor. Valve inspections done by the Task Team and discussed with BW/IP indicated that disc axial play was, in some cases, excessive. This aggravated the problem created by the elevation difference. Fifty-seven pressure seal check valves in Unit 1 and 2 are potentially affected. One-hundred-three bolted bonnet valves are unaffected because the valve design prevents an elevation difference during reassembly. However, all bolted bonnet valves are being examined to assure the amount of axial play is within the design requirements. Corrective actions for the affected valves are described further in Section VII.B.1.

- Backflow on April 23 and May 5 was initiated because of hung open check valves coincident with the failure of Auxiliary Operators to operate manual valves in the sequence specified by the procedures. A remote isolation valve, which operators thought was shut, also contributed to the May 5 backflow initiation and caused a delay in stopping the event.

V. SIGNIFICANCE OF THE APRIL 23 AND MAY 5 EVENTS

This section evaluates the safety significance of the April 23 and May 5 events. The evaluation is divided into three parts: 1) evaluation of the actual significance of the events; 2) evaluation of the impact of the events on the integrity of the piping system and pipe supports, containment penetrations, and instrumentation; and 3) evaluation of the significance of the events if they had occurred during operation of the plant. Each of these is discussed separately below.

A. Actual Significance

The April 23 and May 5 events did not and could not have resulted in any radioactive release because they occurred during preoperational testing prior to fuel load. Therefore, these events posed no threat to public health and safety. Furthermore, although water was diverted from the steam generators via the AFW System to the CST, the decrease in the steam generator level was detected and AFW flow was restored prior to excessive loss of steam generator level.

While these events did have a potentially significant impact on certain plant components, such as piping, supports, instrumentation and containment penetrations, these potential problems were identified, the hardware was thoroughly evaluated, and necessary corrective actions have been or will be taken by fuel load.

B. Piping and Support Integrity

The Task Team performed a preliminary thermal blowdown analysis on the piping and pipe supports affected by the backflow from the steam generators for both events. This analysis included a correlation of the level changes in the steam generators to backflow rates and mass/energy balances at piping junctures to determine piping temperatures. These temperatures were compared to the amount of coating discoloration observed on the piping. The coatings manufacturer (Carboline) performed a type test in which similarly painted metal coupons were heated at several different temperatures in an oven. The resulting discoloration of these coupons was then compared to the discoloration of the actual piping which provided support for the temperatures calculated in the mass/energy balances.

After the preliminary temperature distributions for the affected piping were determined, a piping and pipe support stress analysis was completed. This analysis identified several areas where piping Code allowable stress was exceeded. Subsequent ultrasonic testing of the pipe verified that no plastic deformation had occurred. Thus, the stresses resulting from the elevated temperature were within the elastic range for the piping material and no piping needs to be replaced.

Sixty-four supports were preliminarily identified as being loaded beyond their current design load. A more detailed analysis, accounting for actual installation tolerances and actual material allowables based on certified material test reports (CMTRs), determined that only ten supports were overloaded. Nonconformance Reports were written for QC examination of the significant attributes of these supports and no deviations or deficiencies were found.

The results of the completed blowdown thermal analysis agree with those of the preliminary analysis with the exception of a small segment of pipe (approximately twenty feet) on the MDAFWP discharge whose temperature was increased by 30°F over that of the preliminary analysis. This will be factored into a final stress and support analysis. Should the final stress analysis indicate pipe stress over that allowed or any increased support loading beyond that which was previously evaluated, the affected piping and supports will be re-evaluated and reworked as necessary.

C. Impact on Containment Penetrations

The Task Team also evaluated whether the feedwater containment penetrations could have been degraded as a result of the backflow events given that exposure to the temperatures associated with hot water for a sufficient period of time could result in concrete damage. Physical inspection revealed no damage to the penetrations. Additionally, analysis and UT inspection of the piping concluded that the penetrations were not affected by loads created by piping that was in the backflow path.

D. Impact on Instrumentation

Flow Elements (FE), Flow Transmitters (FT) and Temperature Elements (TE) could have experienced high temperatures as a result of the backflow. A review of the instruments' design against a maximum possible temperature of 557°F was performed. The FEs are metallic plates and hence unaffected. The TEs' qualification temperature is 2000°F and therefore acceptable. The FTs, per discussion with the manufacturer, may lose calibration and may be damaged as a result of high temperatures. The FTs will be re-calibrated if possible and replaced if necessary prior to fuel load.

E. Potential Significance of the Events if they had Occurred During Operation

The TU Electric Task Team performed an evaluation of the potential effects of malfunctions of BW/IP check valves during plant operations. The check valve disc hang up condition occurred in the AFW check valves and Main Steam (MS) check valves (steam supply lines). In the absence of a line break or manual valve misoperation, the failure of the valves would lack significance because of the absence of a backflow path.

In the event of a loss of AFW flow to the steam generators for any reason the Emergency Response Guidelines would require operator actions to commence Reactor Coolant System (RCS) cooldown using systems other than the AFW and Feedwater Systems.

VI. PRECURSORS

The Task Team investigated several CPSES check valve failures from 1983 to just prior to the April 23 event. In addition, previously received industry information was reviewed to determine if CPSES had properly reacted to that information. Finally, the Task Team contacted a number of sites who were thought to have purchased BW/IP check valves.

The Task Team conclusions are described below.

A. Nuclear Industry Experience with BW/IP Check Valves

The Task Team investigated the nuclear industry's experience with BW/IP check valves to determine whether there was any indication that BW/IP check valves were prone to failure due to excessive valve disc elevation or axial disc play. Although various concerns about the performance of check valves were identified, there was no indication from industry that the BW/IP check valves were likely to malfunction from these causes.

There have been a number of NRC Notices and Bulletins that raised concerns about the malfunction of check valves through various failure modes. There were no cases of check valve failure identified from NRC correspondence similar to that of the failure modes experienced at CPSES. Additionally, the Task Team determined that NRC Bulletins on check valves had been reviewed by CPSES and corrective actions taken as applicable. Similarly, the Task Team determined that NRC Notices on check valves had been reviewed for applicability and appropriate actions taken.

The Task Team performed a search of the INPO Nuclear Plant Reliability Data System (NPDRS) to determine if BW/IP check valves had failed at other plants. A total of thirty-eight BW/IP check valve failures were identified; twenty-three of these failures were related to disc seating. Of these twenty-three, approximately seventy-five percent were caused by foreign material caught between the disc and seat, disc distortion, improper installation of the disc-stud-hinge arm assembly, or corrosion of materials. None of these BW/IP check valve malfunctions were identified as occurring through the failure modes experienced at CPSES.

The Task Team contacted four plants to discuss problems with BW/IP valves. Three of the four plants had experienced backleakage and all expressed concerns with the general quality of their BW/IP valves. The three affected plants provided the following information regarding backleakage through their BW/IP check valves:

1. A unique procedure had been supplied to St. Lucie for check valve assembly. The procedure applies to 12 inch pressure seal bonnet model 73060 check valves and is used for clevis, bonnet arm and disc assembly replacement. This procedure was designed to make up for variations in tolerances applied during body/neck fabrication. These variations in tolerance resulted in an unacceptable difference in elevation between the centerline of the disc and the centerline of the seat. The need for this procedure was recognized by BW/IP before the valves were shipped to St. Lucie.

2. Diablo Canyon experienced seat leakage problems with BW/IP pressure seal bonnet check valves. Diablo Canyon attributed its problems to uncertainty involved in aligning the disc parallel to the seat during assembly, although no non-intrusive techniques (radiography, ultra-sonics, fiber-optics) were used to verify that rotational misalignment was the sole cause of their seat leakage problems. These uncertainties existed because there are no dowel pins or other type of positive positioning mechanisms designed into the valve to ensure disc/seat parallelism. This problem is unrelated to the check valve failures that occurred at CPSES on April 23 and May 5.

3. McGuire also experienced problems with BW/IP check valves. These problems include significant bonnet leakage and three instances of greater than design leakage past the seat. The valves experiencing backleakage were replaced before the exact cause of the malfunction was determined. McGuire assumed that, because of the magnitude of the backflow, the disc was stuck in the neck of the check valve. TU Electric has not experienced a similar check valve failure. McGuire also modeled a BW/IP valve in a test loop and determined after experimentation that the bonnet should be raised to ensure proper seating. The assembly procedures at the McGuire plant have been revised accordingly.

The Task Team's review of available industry experience with BW/IP check valves did not identify any problems that were related to the CPSES failure due to excessive valve disc elevation or axial disc play. Check valve leakage has been observed; however, this leakage was generally attributed to causes unrelated to valve reassembly, such as foreign material between the seat and disc or disc distortion. Based on discussions with McGuire it was determined that a similar failure mechanism had been identified; however, this information was not disseminated to the industry.

B. <u>Previous Occurrences at CPSES</u>

The Task Team conclusions pertaining to previous CPSES check valve failures are discussed below.

Prior to April 1989

In 1983, check valve parts were found in the Component Cooling Water (CCW) heat exchanger. A valve disc had become detached because a weld which held the disc retaining nut had cracked allowing the retainer nut to back off. Further investigation found that the failed weld was a tack weld instead of the specified fillet weld. A modification recommended by BW/IP was made to replace tack welds with fillet welds holding the disc nut to disc stem. In addition, during this same time frame, various check valves within AFW and other systems had been disassembled for flushing and draining operations and then reassembled. The modification and reassembly after flushing and draining are germane to the backflow events only because they most likely caused the check valves to become hung open. At the time, the incorrect instructions for reassembly were apparently followed, which created the elevation difference between the seat and disc.

In May 1985, a CPSES Problem Report documented that damaged snubbers along with a cracked disc seat and bent stud on 1MS-042 (steam inlet check valve to the TDAFWP) had been found in the AFW turbine steam supply line. Revision 0 of the failure analysis stated that the cause was the bonnet being too low in the body. The report was later revised based upon input provided by BW/IP to state that the damage was caused by unusual flow conditions in the piping system (water hammer) coupled with the bonnet being installed crooked. Through discussion with BW/IP, TU Electric personnel agreed that the failure was not due to the bonnet being too low in the valve. The corrective action included modification to the valve to accept the stated flow conditions by lengthening the disc stop. In addition, the piping and supports were modified to minimize the consequences of water hammer upon turbine pump start.

In retrospect, the problem with valve 1MS-042 may have been attributable in part to the incorrect assembly instruction. A more thorough discussion with BW/IP and a more in-depth investigation by TU Electric in 1985 might have confirmed that the reassembly procedure was incorrect.

Both TU Electric and the AIT noted that post assembly backleakage testing had not been specified or performed for any of the aforementioned valve disassembly operations. In addition, the AIT noted that no post maintenance test or surveillance requirements were specified from 1985 to the recent hot functionals.

It is TU Electric's position that applicable provisions in Section XI of the ASME Code do not require that check valves be tested other than in the forward direction. However, it should be noted that in 1988, TU Electric revised its post-modification test procedures to require post work testing for backleakage. Therefore, TU Electric had taken action to procedurally address this issue prior to the backflow events.

Surveillance testing was not performed on the AFW System after 1985 because operability requirements in accordance with Technical Specifications were not applicable. The AFW System was, however, included in the plant layup program and was maintained in wet layup with hydrazine-treated water for most of that period. There was no evidence of corrosion contributing to the failures experienced in the AFW System.

Backleakage on April 5

On April 5, 1989, while filling steam generators following draining to attain in-specification chemistry, a report of water flowing into the TDAFWP Room was received by the Control Room. Investigation revealed that the source of the water was backleakage through the TDAFWP piping. A flowpath was found from Steam Generator 4 AFW supply line to the TDAFWP room through a clearance-tagged open vent valve. This flowpath indicated that check valve 1AF-106 in TDAFWP supply line to Steam Generator 4 was not seating properly.

At the time, an instruction was being written to forward flush the TDAFWP supply lines to the steam generators with Reactor Makeup Water, and it was decided to add a section to this instruction in order to determine if the check valves in the remaining TDAFWP supply lines were seating. The flush identified that two of the remaining check valves were not properly seating. Work Requests were written and a post Hot Functional (HFT) due date of May 26, 1989 was assigned. Testing, radiography and CAD techniques performed after the April 23 and May 5 events determined that the failure of these valves to seat was due to an elevation difference between the valve disc and seat.

The Work Requests for these valves did not quantify the amount of leakage. Therefore, the organizations which review procedures and Work Requests were not alerted, nor did they pursue the severity of the problem. As a result, the backleakage was not documented on a higher-visibility document such as a PIR or Nonconformance Report (NCR).

Corrective action for this above concern is described in Section VII.C.

April 19, 1989 Event

On April 19, 1989 a miniflow check valve, 1AF-069, for MDAFW Pump 2 was identified as deficient after the pump's suction relief was noted to be lifting. Two Nonconformance Reports were written to document the condition of 1AF-069 and the valve was disassembled and inspected. As a result of this inspection, a deformation of the disc stem and face of the stop was identified. This damage was caused by tapping of the valve disc against the stop as a result of turbulence produced by an upstream orifice. To correct this problem, the disc stop was built up an additional .125" to help keep the disc more in the flow stream when the valve is open, thereby reducing tapping.

In addition to the deformation, approximately .175" of axial play was noted in the valve disc. At that time, acceptance criteria for axial play were not available at CPSES. However, after discussions with BW/IP, the disc stud bushing was trimmed which resulted in reduced axial play. Following the April 23 and May 5 events, testing revealed that the rework was successful.

The Task Team concluded that although the April 19 deficiencies, tapping and axial play, were not the primary cause of the April 23 failure, the relationship between the two failures was not identified. Design engineering personnel were not aware of the April 5 check valve failures. They therefore believed that the 1AF-069 failure was isolated. Consequently, engineering concentrated on the readily identifiable deficiencies in 1AF-069 of tapping and axial play. Extensive investigatory techniques such as radiography and CAD were not thought to be needed, and were therefore not developed or used.

The Task Team could not positively conclude that the discs were hung open in the check valves involved in the April 5 and April 19 failures. The valves involved in the April 5 failures were not opened or inspected prior to the April 23 and May 5 events. The valve involved in the April 19 failure was not radiographed prior to disassembly. Therefore, it was not possible to determine the pre-existing disc/seat relationship. However other indications such as the observed amounts of water on April 5 and the lifting of a pump suction relief on April 19, indicate that the discs were most likely hung open.

<u>April 23, 1989 Event</u>

The April 23 event is described in detail in Section III.A, above.

PIR-110 was written to investigate this event. The Manager, Operations recognized that simultaneous opening of 1AF-041 and 1AF-042 was incorrect and that it had initiated the event. However, he felt that the error was isolated to the shift in question. Therefore, procedure noncompliance was not pursued with other shifts.

The AFW operability test was performed on May 5 with Operations department management's knowledge of the April 23 event. The potential for creating another backflow condition had been recognized by Operations management; however, it was concluded that the test procedures placed sufficient controls on the operation of manual valves so that backflow would not occur. Several similar tests had been conducted between April 23 and May 5 without difficulty. Operations management believed that it was appropriate to proceed with the test on May 5 because even if a generic problem existed in the BW/IP check valves, isolation valves in the AFW System would provide adequate protection against backflow and as stated before, at that time, Operations management felt that the operator error on April 23 was an isolated event.

In retrospect, the operator errors could have been more fully investigated and investigation results provided to all Operations personnel.

C. Conclusions

Documentation of available industry experience with check valves did not identify any concern with the elevation of the valve disc related to the valve seat for BW/IP check valves. However, several precursor events at CPSES such as the 1985 failure of check valve 1MS-142 and the April 5 and April 19 failures did involve BW/IP check valves that may have had the same failure mechanism as the check valve failures which occurred on April 23 and May 5. In addition, Operations management could have more thoroughly investigated the April 23 operator error prior to permitting the performance of additional testing on the system.

Based upon these precursors, TU Electric is making improvements in the thoroughness of its evaluations of the causes of equipment failures, the documentation and reporting of equipment failures, and the evaluation of the effect of equipment failures on the operability of plant systems. These improvements are described in Section VII.C.

VII. CORRECTIVE ACTIONS, PREVENTIVE ACTIONS, LESSONS LEARNED, AND ASSOCIATED IMPROVEMENTS

A. Introduction

The events on April 23 and May 5 involved deficiencies in the BW/IP check valves. Additionally, the events themselves adversely affected certain hardware at CPSES. Subsection B below identifies the corrective actions that TU Electric is taking for this hardware. In addition, Subsection B describes the actions being taken by TU Electric to address these root causes and prevent recurrence of events similar to those on April 23 and May 5.

Lessons learned, together with associated improvements made by TU Electric, are discussed in Subsection C, below. This Subsection also addresses the weaknesses discussed by the AIT in its July 10 letter and by the NRC at the meeting on July 17. Appendix 1 lists the conclusions and recommendations identified by the NRC and states how TU Electric has addressed each one.

All actions described in the following Subsections will be completed by fuel load of the respective units unless otherwise noted.

B. Corrective and Preventive Actions

1. Corrective Actions for Hardware

TU Electric is taking the following corrective actions for the hardware at CPSES:

Discs for the BW/IP check valves - As discussed in Section IV above, backflow occurred through the BW/IP pressure seal check valves on April 23 and May 5 when the valve disc hung open. This failure was caused by an elevation difference between the disc and seat during valve reassembly, resulting in the disc lodging under the seat lip. A potential contributing factor was an unspecified and, in some cases, excessive axial play in the valve disc. To address these problems, TU Electric is taking the following actions:

a. Unit 1 BW/IP pressure seal check valves are being inspected and/or reworked by eliminating the elevation difference between the valve disc and seat. The rework is being accomplished by taking critical dimensions and using these dimensions to establish the amount of the retainer ring backout for the valve bonnet. The amount of permissible backout is being specified by engineering.

b. Pressure seal and bolted bonnet BW/IP check valves will be inspected to determine disc axial play. Based on the results of these inspections, the valves will be reworked as necessary in accordance with vendor specified tolerances.

c. Rework performed in accordance with either action a. or b. above, is being inspected by QC personnel and post-modification operability tests are being performed, including verification that the valve fully closes.

d. Inservice testing requirements are being established to ensure closure of BW/IP and other check valves for which backflow is a safety function. This review will identify which valves will be tested prior to fuel load and will also note exceptions and their milestones.

Damage to BW/IP Check Valves - The backleakage through BW/IP check valve 1AF-069 on April 19, 1989, was caused by damage to the valve disc and body due to turbulence produced by an upstream orifice. To address this problem, TU Electric is taking the following actions:

a. A full inspection of 1AF-069 was performed. Evidence of tapping on the valve stop was noticed and a deformation of the disc stem and face of the stop was present. Additionally, approximately .175" of axial play was noticed (this amount of axial play would be acceptable under the subsequently-developed BW/IP installation tolerances). It appeared that the axial play resulted from fabrication and was not the result of operation. In order to properly seat the valve, the disc assembly was taken apart and the axial play was reduced from approximately .175" to .060"- .075". Also, the disc stop was built up an additional .125" to repair the deformed area as well as to help keep the disc more in the flow stream when the valve is open. The valve has received post-repair tests and is now operating properly.

b. Kalsi, Inc. is performing an evaluation of CPSES check valves in response to INPO Significant Operational Event Report (SOER) 86-03. Following receipt of this evaluation, TU Electric will determine whether to increase the distance between orifices and check valves. If it is determined that check valve failure is not imminent, implementation of the design changes may be deferred until after fuel load. Additionally, periodic post fuel load internal inspection of check valves will be performed to monitor and trend wear in the check valves. If these inspections reveal excessive wear TU Electric will initiate design changes to increase the distance between the orifices and check valves.

Damage to Pipe Components - The events on April 23 and May 5 caused visible damage to a piping support, caused paint to blister or discolor on certain AFW piping, and resulted in stresses in AFW piping that exceeded Code allowable limits. To address these problems, TU Electric is taking the following actions:

a. TU Electric is repainting the affected piping.

b. TU Electric has replaced the damaged pipe support.

c. TU Electric has inspected piping and supports based on preliminary blowdown thermal analysis. Upon completion of the final stress and support analysis (using the finalized blowdown thermal analysis temperatures), they will be reevaluated and reworked as necessary. These and any other follow-up actions will be discussed further as part of SDAR CP-89-15.

2. Preventive Actions

TU Electric is taking the following actions to address the root causes of the events on April 23 and May 5 and to prevent recurrence of similar events:

Installation Procedures for the BW/IP Check Valves - As discussed in Section IV above, backleakage occurred through the BW/IP check valves on April 23 and May 5 because of an elevation difference between the valve disc and seat due to inadequate reassembly instructions. Additionally, the lack of criteria governing axial play in the valve disc may have contributed to the backleakage. To address this problem, TU Electric is taking the following actions:

a. The onsite valve assembly procedure has been revised to include a requirement for determining the elevation adjustment necessary to avoid interference between the disc and the seat.

b. The acceptable range for the axial play dimension has been determined by BW/IP and will be included in its instruction and site procedures.

c. A 4" BW/IP check valve was bench tested following adjustment in accordance with the revised procedure discussed above to verify the adequacy of the procedures. The test showed the procedure was adequate.

d. The Quality Assurance department has taken additional action to assure that components and material procured from BW/IP will meet quality requirements. BW/IP has been placed on "Special Status" on the Approved Vendors List. This "Special Status" requires Engineering to develop critical characteristics on safety-related parts and components purchased from this vendor. These critical characteristics will be checked in the shop and during receipt inspection activities. This "Special "Status will be maintained until sufficient confidence has been reestablished in the quality of material supplied by BW/IP.

Adherence to Procedures - As discussed in Section IV above, the failure to follow procedures on April 23 and May 5 was caused by a lack of understanding of the need to perform the steps in the sequence written in the procedure. To address this problem, TU Electric is taking the following actions:

a. The Manager, Operations met with the personnel involved in these events and counseled them on procedure usage and procedure compliance.

b. The administrative procedure which provides guidelines on the use of procedures has been revised to emphasize the requirement that procedure steps are to be performed in the sequence specified in the procedure, except as otherwise stated in the procedure, allowed by emergency operations procedure rules of usage, or permitted by the Shift Supervisor with appropriate documentation of the deviation.

c. Administrative procedures have been revised to add applicable procedures for the AFW System to the list of procedures required to be available and referenced when performing field work.

d. The Shift Operations Manager has developed and is implementing an action plan to enhance procedural compliance. As part of this plan, a memorandum on procedure compliance was provided to the Shift Supervisors, who in turn discussed the memorandum with their respective crews. The Manager, Operations and/or Shift Operations Manager also met and discussed the memorandum with each crew. Additionally, a workshop was held by the Manager, Operations with Operations Department Senior Reactor Operators (Shift Supervisors, Unit Supervisors, Shift Technical Advisors and Staff), including Training and Plant Evaluation personnel, to discuss the April 23 and May 5 events and procedure compliance.

e. Quality Assurance is performing an overview of implementation of selected procedures that control operation of selected systems necessary for safe shutdown. This overview will be performed by personnel who possess technical experience in operation, maintenance and testing. These overviews will continue until they are deemed unnecessary by the Director of Quality Assurance.

f. In November 1988, TU Electric instituted a program for performance-based audits and surveillances. This approach to audits and surveillances emphasizes direct observation of plant activities in-progress by personnel who are qualified in the activities being observed. It also stresses the technical adequacy of the procedures being used, as well as the performance of the personnel who are implementing them. The Quality Assurance department will re-evaluate this program based on identified lessons learned from the April 23 and May 5 events. Any necessary program enhancements will be made after completion of this evaluation. In addition, compliance-based verifications will continue to be performed to assure personnel are adequately implementing program requirements (i.e., procedures and instructions) which govern their activities until they are deemed unnecessary by the Director of Quality Assurance.

Manipulation of Remote-Operated Valves - As discussed in Section IV above, the event on May 5 was caused, in part, by a mechanically bound isolation valve. To address this concern, TU Electric is taking the following actions:

a. Reach rod operators for other safety-related valves will be evaluated for proper operability and human factors considerations. This evaluation will include consideration of factors such as whether a valve is operated in a direction that is opposite to the usual direction for valve operation, whether the valve is operable, the ease of operation of the valve, and the gear ratio and time required to operate the valve.

b. Safety-related reverse-operated valves documented in the above evaluation will be marked to indicate the direction of operation.

c. The cause of valve 1AF-055 binding will be determined and corrected.

<u>Other Actions to Preclude Recurrence</u> - In addition to the actions discussed above, TU Electric is taking the following actions to help prevent recurrence of events similar to those which occurred on April 23 and May 5:

a. The events on April 23 and May 5 were documented on PIR-110 and PIR-129, respectively, for purposes of obtaining corrective action. A discussion of these PIRs and associated issues such as industry experience with check valves is being added to the licensed and non-licensed operator requalification and replacement training programs. Operations personnel will receive training in this part of the requalification program prior to fuel load.

b. Technical Specification surveillance test procedures for the AFW pumps are being revised. The revision will require the discharge pipe downstream of the test loop to be checked for elevated temperatures that would indicate backleakage through check valves 30 minutes after the test .

c. When requesting personnel to provide assistance in performing a plant evolution, reactor operators and auxiliary operators have been directed to brief the personnel on the evolution and applicable procedures prior to performing them.

d. The Shift Operations Manager has counseled the Senior Reactor Operators (SROs) on the importance of maintaining proper system status and the risks involved in leaving a valve lineup in an indeterminate condition.

C. <u>Lessons Learned and Associated Improvements</u>

In addition to the corrective actions and preventive actions discussed in the preceding section, TU Electric has evaluated the events on April 23 and May 5, the precursors to these events and the response to these events to determine lessons learned and identify corresponding improvements. In performing this evaluation, TU Electric has considered the findings and recommendations in the NRC's AIT Report and the weaknesses identified by the NRC in the meeting on July 17, 1989.

TU Electric has identified a number of areas where improvements could be made. In some cases, the areas overlap, and some improvements are common to more than one area. The areas, together with TU Electric's corresponding improvements, are discussed below.

1. Management and Supervision of Operations

Several of the circumstances discussed in this report indicate that improvements can be made in the management and supervision of operations. These improvements are as follows:

- Transition from a Construction to an Operations Attitude - To further instill an operating attitude in all Operations personnel prior to fuel load, TU Electric is taking several actions, including: 1) directing personnel to immediately evaluate the impact of events and equipment failures on the operability of components and systems; 2) directing personnel to evaluate events and equipment failures for reportability under 10CFR50.72 and 50.73 and the Technical Specifications; 3) deleting the provision in ODA-408 which allows procedures for off-normal evolutions to be issued without review by the Station Operations Review Committee (SORC), and requiring test procedures issued after September 2, 1989 to be reviewed and approved through post operating license processes; and 4) eliminating temporary programs and more fully implementing permanent operational programs.

- Greater Control of the Project by Operations -TU Electric is taking several steps to provide Operations with greater control of the project, including: 1) reassignment of management of the Transition Team from the Projects organization to the Vice President, Nuclear Operations; 2) development of an integrated schedule by Operations; 3) completion of system and area turnovers to Operations; 4) reassignment of the responsibility for the power ascension program from Startup to Operations; and 5) requiring Operations approval for scheduling incomplete construction items to be completed after fuel load.

- Improvements in Notification of Operations Management and Supervision of Events and Equipment Failures - TU Electric is taking several steps to improve notifications to Operations management and supervision of events and equipment failures, including: 1) Operations personnel have been instructed to provide greater detail in problem descriptions on Work Requests to alert management to the severity of problems; 2) SROs are now reviewing Work Requests for potentially significant multiple equipment failures and are notifying management of such failures; 3) operators have been directed to request assistance from systems engineers to help evaluate problems involving plant systems; and 4) the CPSES morning meetings on operation and plant events have been enhanced through greater participation by all project organizations.

- <u>Time and Manpower Needs for Specific Tasks</u> - TU Electric is taking several steps to provide additional assurance that Operations management and supervision are aware of the time and manpower requirements for specific activities, including: 1) workshop training has been provided to Shift Supervisors on planning and controlling plant evolutions, including ensuring that manpower levels are adequate for routine evolutions; 2) AOs have been instructed to identify any need for additional manpower and to identify any problems with access, work conditions, etc. during pre-evolution briefings; and 3) activities performed near the end of a shift will be planned to ensure that the activities can be performed prior to the end of shift or that relief will be available for the personnel performing the activities at the end of the shift.

2. <u>Corrective Actions</u>

Several of the circumstances discussed in this report indicate that improvements can be made in the corrective actions for plant events and equipment failures. These improvements are as follows:

- <u>Documentation and Reporting of Events and Equipment Problems</u> - TU Electric is taking several steps to enhance the documentation and reporting of plant events and equipment failures, including: 1) Operations procedures have been revised to encourage Operations personnel to document human factors concerns inside or outside the control room; 2) Operations personnel have been instructed to document the significance of problems, including leakage amounts, on Work Requests; and 3) the Condition Report (CR) program has been initiated for the documentation of non-hardware problems.

- <u>Aggressiveness and Timeliness of Investigations of Plant Events and Equipment Failures</u> - TU Electric is taking several steps to increase the aggressiveness and to improve the timeliness of investigations of plant events and equipment failures, including: 1) the PIR program will be refined to include provisions for failure mode analyses and human performance evaluations; 2) PIRs are being discussed in the CPSES morning meetings on operations and plant events to provide for immediate management review and determination of whether multi-discipline evaluations are warranted; 3) operators have been directed to request assistance from systems engineers to help evaluate problems involving plant systems.

- Improvements in Task Team Evaluations - Based upon the experience with the Task Team investigation of the April 23 and May 5 events, TU Electric has learned several lessons that will be applied as appropriate to any similar investigations in the future including: 1) establishing the team promptly after the event; 2) utilizing a multi-discipline team; 3) having dedicated, full-time team members; 4) designating a single point of contact with the NRC to ensure that the NRC is provided with complete, consistent, and timely information; and 5) establishing a clear line of communication and direction from management to the Task Team. These lessons will be formalized in an incident investigation procedure.

3. Communications Among Operators and Shifts

Several of the circumstances discussed in this report indicate that improvements can be made in communications among operators and between shifts. These improvements are as follows:

- Communications Among Operators - Administrative procedures will be revised to provide for the prompt transmission of plant incident information to Operations personnel.

- Communications Between Shifts - Administrative procedures have been revised to require that an oncoming shift be notified of the "Lessons Learned" by the preceding shifts, including plant events, significant PIRs involving operator error or involvement, and unexpected system or component responses. Shift Supervisors are now required to brief the oncoming crew on plant status, upcoming evolutions on the next shift, and current lessons learned or PIRs. The Manager, Operations now briefs a crew returning to shift work after a long period off-shift to notify them of events during this period and scheduled events during the next week. Shift Orders have been enhanced by including policy changes, corrective actions due for PIRs, and other general information. A working copy of evolutions is now maintained by Operations until the evolution is completed or terminated by the Shift Supervisor.

4. Personnel Awareness of Operating Events and Equipment Failures and of Their Implications for System Operability

Several of the circumstances discussed in this report indicate that improvements can be made in awareness by Operations personnel of operational events and equipment failures and of their implications for system operability. These improvements are as follows:

- Impact of Work Requests - TU Electric is taking several steps to provide additional assurance that Operations personnel are aware of Work Requests and their implications for plant operability, including: 1) open corrective and preventive maintenance Work Requests have been reviewed, and any operability concerns and mode restraints have been identified; 2) control room operators have begun to review Work Requests generated during the previous 24 hours to identify significant failures, potential impacts on plant operability, reportability of the Work Requests, and the priority of the Work Requests; and 3) operators have been directed to request assistance from systems engineers to help evaluate problems involving plant systems.

- Availability of Information - TU Electric has taken or is taking the following actions to improve the availability of information regarding plant events and equipment failures to Operations personnel including: 1) making current PIRs available in the control room and referencing the PIRs in the station log; 2) discussion of PIRs at the CPSES morning meetings on operations and plant events; and 3) implementation of a system status program that may include, for example, the use of laminated prints that can be marked to indicate system or component status.

- Shift Log Information - TU Electric will take the following actions to improve the documentation of equipment problems in shift logs: 1) problems causing initiation of a PIR will be referenced in the Station Log with its PIR number; 2) Technical Specification Limiting Conditions for Operation (LCO) will be tracked in the Unit Log and will be discussed during the shift turnover process.

D. Conclusion

TU Electric has evaluated the events on April 23 and May 5, their impact on hardware, and implications for operation. Based upon this evaluation, TU Electric is taking corrective action for the affected hardware, has taken corrective action to address the root causes of the events and to prevent recurrence of similar events, and is making improvements to address the lessons learned from the events.

VIII. SUMMARY AND CONCLUSIONS

TU Electric performed an investigation of the root causes and significance of the April 23 and May 5 events. The events were caused by defects in BW/IP check valves as a result of an inadequate reassembly procedure, and by a failure of Operations personnel to follow procedures while manipulating isolation valves. Additionally, the event of May 5 was caused in part by a mechanically bound isolation valve. These events had no actual safety significance because there was no fuel in the reactor and Unit 1 was not radioactive. If a similar event occurred during operation, operator action would have maintained the reactor in a safe condition.

TU Electric has taken corrective action for the deficiencies in the hardware identified by these events, including inspection and modification of BW/IP check valves. Additionally, TU Electric is taking action to address the root causes of the events and to prevent recurrence of similar events, including revision of the assembly procedure for the BW/IP check valves, providing additional training on compliance with procedures and clarification of the procedure governing manipulation of the AFW isolation valves. Finally, TU Electric has identified lessons learned from these events and is taking actions to improve the management and supervision of Operations personnel, to improve corrective actions for plant events and equipment failures, to improve communications among Operations personnel and between shifts, and to improve awareness of operating events and equipment failures and their implications for system operability.

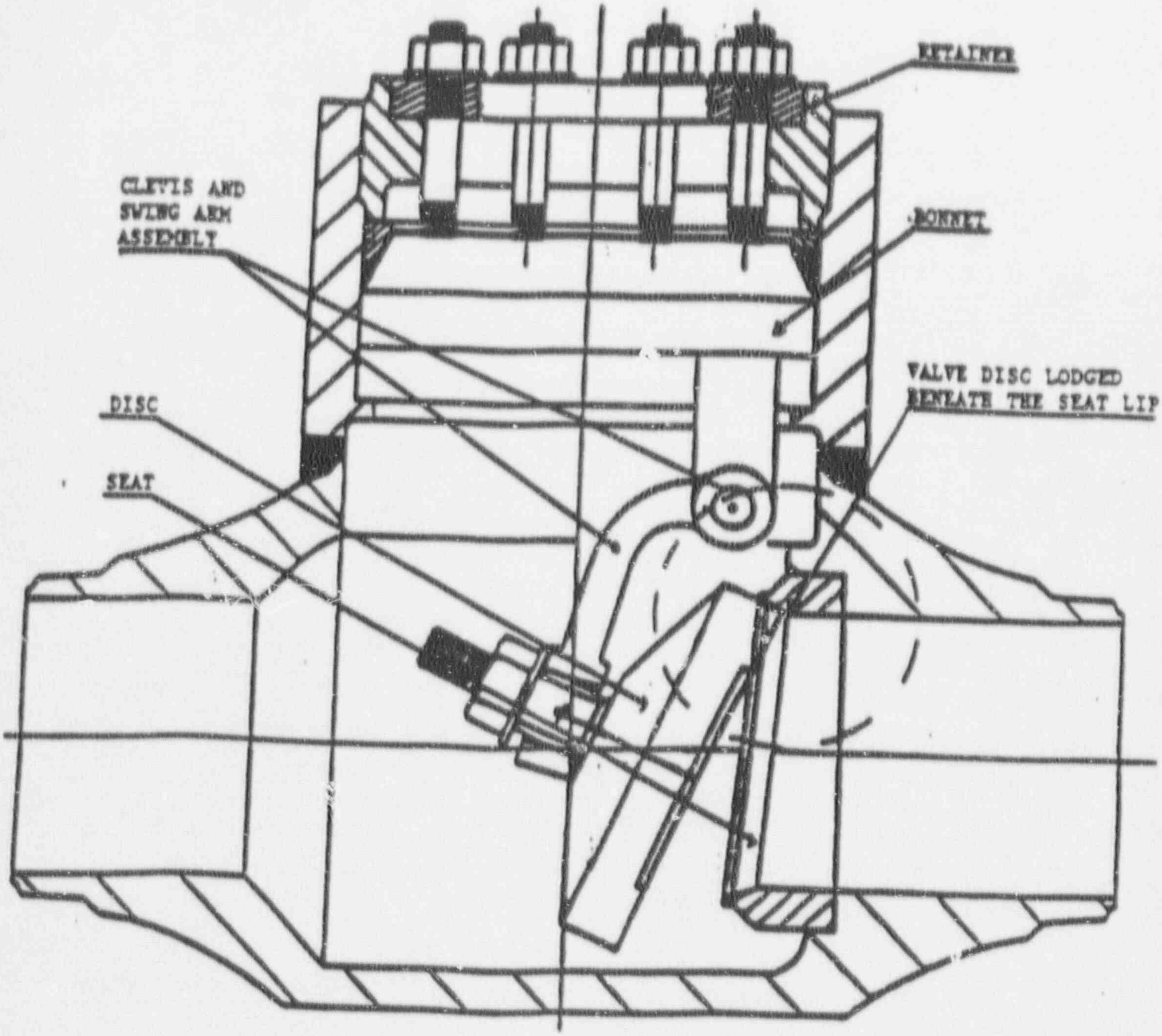

RETAINER
CLEVIS AND
SWING ARM
ASSEMBLY
BONNET
VALVE DISC LODGED
BENEATH THE SEAT LIP
DISC
SEAT

FIGURE 1

APRIL 23,1989 EVENT

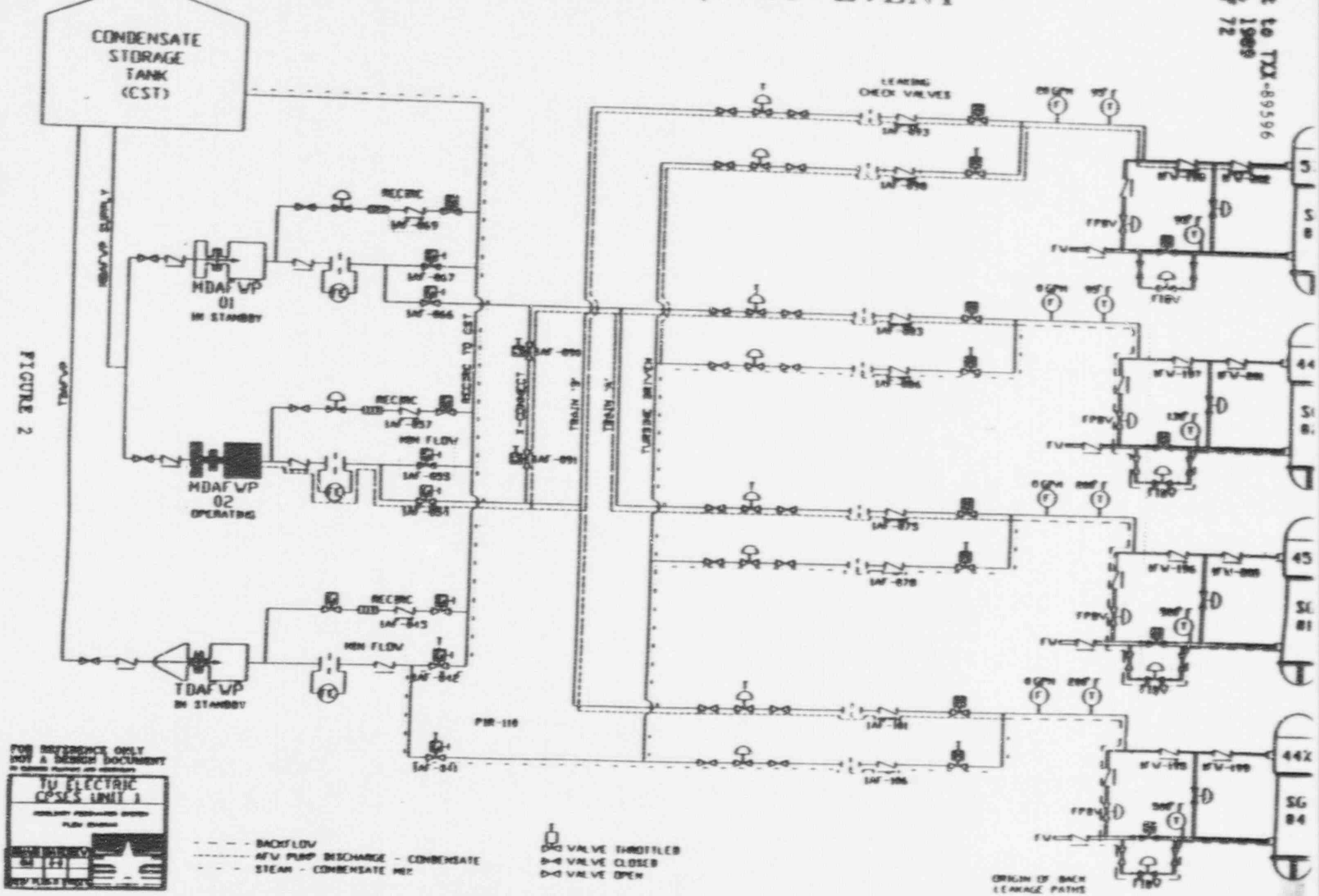


FIGURE 2

Attachment to TXX-89596
August 18, 1989
Page 43 of 72

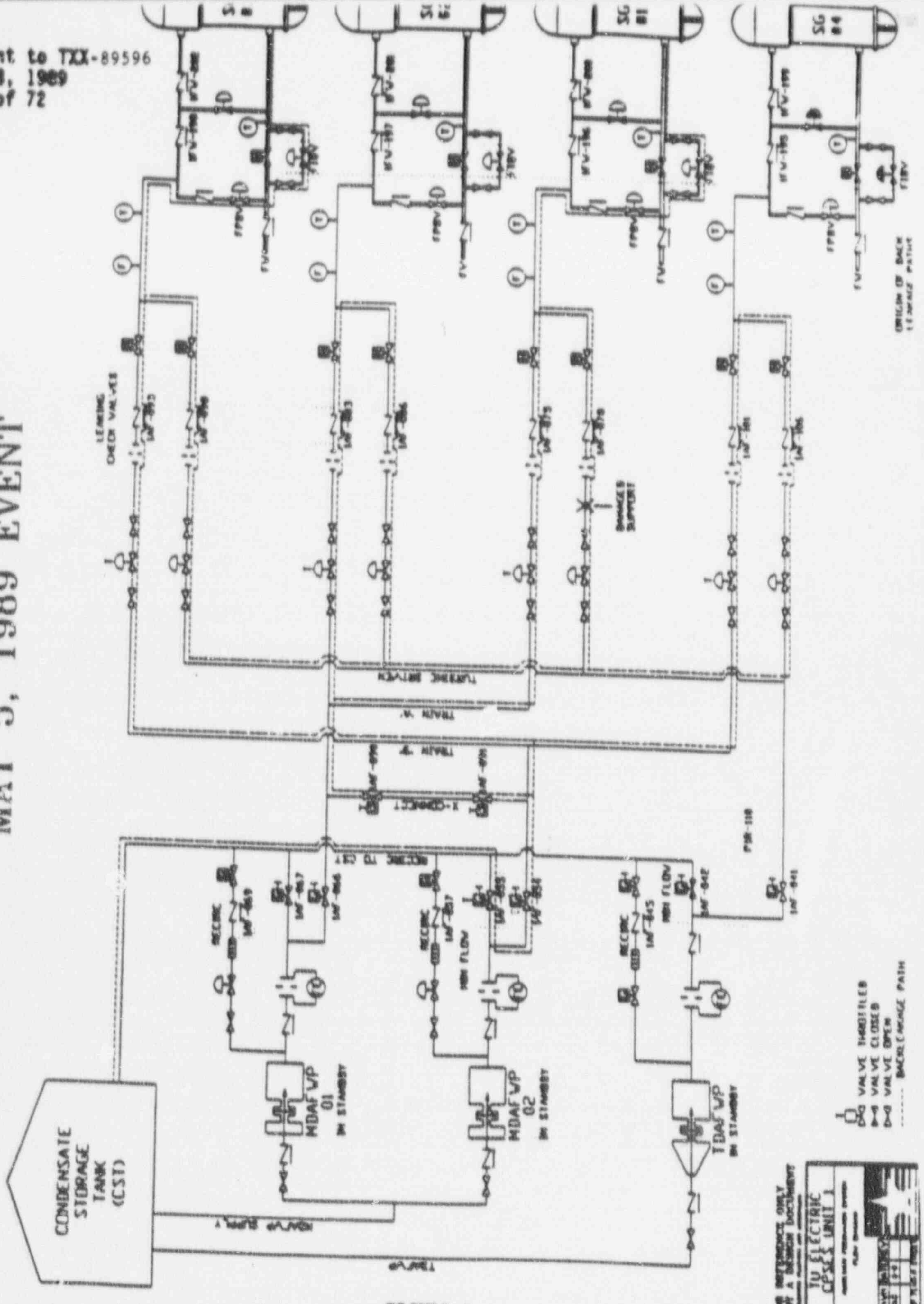


FIGURE 3.

APPENDIX 1

RESPONSE TO NRC CONCERNS

In addition to the actions taken to correct the deficiencies identified as a result of the April 23 and May 5 events, TU Electric has taken a number of actions to address the root causes and prevent recurrence of these events. Furthermore, TU Electric has implemented improvements to address a number of lessons learned as a result of the investigation of these events and the possible precursors to the check valve failure. These actions are described in Section VII of this report.

The NRC staff made a number of conclusions and recommendations in the AIT Report. Additionally, on July 17, 1989 Mr. Warnick, the Assistant Director for Inspection Programs of the Office of Special Projects of the NRC, enumerated similar weaknesses to senior TU Electric management during a meeting in Rockville, MD. TU Electric either has or will address the NRC staff's concerns and recommendations as set forth below.

A. Mr. Warnick's July 17, 1989 Concerns About CPSES Operation and Corrective Actions Implemented by TU Electric

1. NRC Concern

Operators and Startup personnel failure to follow procedures. Valving errors to start the 2 backflow events, PT-01-02, PT-37-01, and PT-64-03.

TU Electric Action

TU Electric is taking a number of corrective actions to improve future compliance with procedures (see Section VII.B.2).

2. NRC Concern

Operators' lack of sensitivity to the position of valves. Changing the AFW valves out of the proper order of sequence.

TU Electric Action

In addition to placing increased emphasis on compliance with procedures, TU Electric has provided training/workshops on avoidance of the April 23 and May 5 events and the risks associated with improper valve line-ups (see Section VII.B.2).

3. NRC Concern

Operators' failure to recognize the significance of check valve backleakage during the precursor event.

TU Electric Action

TU Electric is taking a number of steps to ensure that the significance of equipment failures is documented and that Operators are aware of the impacts of equipment failures (see Sections VII.C.2 and VII.C.4).

4. NRC Concern

Operators' failure to make sure supervision was aware of the three check valves that had significant backleakage (precursor event).

TU Electric Action

TU Electric is taking several corrective actions to ensure that Operations management is aware of future events and equipment failures (see Section VII.C.1).

5. NRC Concern

Supervisors' failure to stay informed of plant evolutions and problems (the system flushing to solve the chemistry problem and the RHR valving problem during the remote shutdown test. If check valve had failed, it would have put RCS water to the RWST.).

TU Electric Action

TU Electric is implementing several corrective actions to improve reporting of equipment failures and plant events to management and supervision of Operations, and to improve the documentation and reporting of events and equipment problems (see Sections VII.C.1 and VII.C.2 and TXX-89430 dated June 26, 1989).

6. NRC Concern

Failure to accurately and adequately document the extent of a problem (the precursor event Work Request said, "repair check valve leakage"). No TDR on RHR event. No TDR on PT 44-01 and QA person doing surveillance did not issue a surveillance deficiency.

TU Electric Action

TU Electric is taking a number of actions to enhance documentation and reporting of future events and equipment failures (see Section VII.C.2 and TXX-89430 dated June 26, 1989).

7. NRC Concern

Weakness in the documentation of equipment problems in the shift log.

TU Electric Action

TU Electric is implementing a number of actions to improve communications on equipment problems and events between operators and shifts (see Section VII.C.3).

8. NRC Concern

Failure to recognize inoperable equipment.

TU Electric Action

TU Electric is taking a number of steps to enhance the timeliness and aggressiveness of corrective action and to enhance the awareness and impact of operating events and equipment failures on system operability (see Sections VII.C.2 and VII.C.4).

9. NRC Concern

Failure to recognize and document equipment out-of-service.

TU Electric Action

A number of steps are being taken to enhance evaluation, documentation, and investigation of equipment failures and work requests (see Sections VII.C.2 and VII.C.4).

10. NRC Concern

Lack of adequate communications between the operating shifts.

TU Electric Action

TU Electric is taking a number of steps to enhance communications within Operations and between shifts (see Section VII.C.3).

11. NRC Concern

Weakness in the exchange of information at shift turnover (Precursor event and April 23 event).

TU Electric Action

TU Electric is taking several actions to enhance communications between shifts (see Section VII.C.3).

12. NRC Concern

Supervision/Management review of problems documented on work requests (Precursor event).

TU Electric Action

TU Electric is taking a number of steps to enhance the documentation, investigation and reporting of events and equipment problems and improve reporting of events and equipment failures to Operations management and supervision (see Sections VII.C.1, and VII.C.2 and VII.C.4).

13. NRC Concern

Failure of persons with knowledge of the precursor check valve problems to raise the information to management.

TU Electric Action

TU Electric is taking a number of actions to improve documentation of events and equipment problems, and to improve the reporting of such events and problems to Operations management and supervision (see Sections VII.C.1 and VII.C.2).

14. NRC Concern

The slowness and lack of direction initially demonstrated by TU Electric following the April 23 event.

TU Electric Action

TU Electric is taking action to improve the aggressiveness of investigation of events and equipment failures and to enhance future Task Team investigations (see Section VII.C.2).

15. NRC Concern

The perception that "Projects and the Schedule" were driving decisions at the time of the precursor event and the start of HFT.

TU Electric Action

TU Electric is taking several actions to improve the control of the project by Operations (see Section VII.C.1).

16. NRC Concern

The perception that the Operations staff are not in control of the plant.

TU Electric Action

TU Electric is taking several steps to increase Operations control over the project (see Section VII.C.1).

B. Conclusions in the NRC's AIT Report on the April 23 and May 5 Events and Corrective Actions Taken by TU Electric

1. NRC Conclusions (4.1.1)

The identification of three inoperable check valves in the TDAFWP supply lines on April 5 should have been aggressively pursued. Instead, it was assigned a normal work request priority. This event reflects a lack of understanding of the system operability implications of failed components and a lack of aggressiveness of Operations management to follow-up on the results of the system flush they had specifically scheduled to determine the scope of the original identified check valve problem. This event was clearly a missed opportunity to discover the full extent of the check valve problem in time to prevent the April 23 and May 5 events from occurring.

TU Electric Actions

TU Electric is taking a number of actions to ensure timely and aggressive investigation and corrective action for future events and equipment failure. Furthermore a number of actions were taken to enhance documentation and reporting of events and equipment failures (see Sections VII.C.1, and VII.C.2 and VII.C.4).

2. NRC Conclusions (4.1.2)

The overall response by control room personnel to both events (falling steam generator levels) was weak (see paragraph 2.1.2).

TU Electric Actions

TU Electric has implemented a number of corrective actions to preclude the reccurrence of similar events which, including training on the April 23 and May 5 events, will improve response of control room personnel to events of this type (see Section VII.B.2).

3. NRC Conclusions (4.1.3)

Continuing to test the AFW system after the April 23, 1989 event with known multiple failures of check valves without taking appropriate precautions shows a potential lack of respect for degraded plant conditions. It also shows lack of communications between shifts.

TU Electric Actions

Operations management did consider the degraded condition of the check valves before concurring that testing activities could proceed. It was concluded that administrative controls in place would compensate for the problems identified if properly implemented. Notwithstanding, TU Electric is taking several steps to ensure that Operations personnel are aware of operation events and equipment failures and of their impact on plant operability, and to enhance communications (see Sections VII.C.3 and VII.C.4).

4. NRC Conclusions (4.1.4)

It took an inordinately long period of time for Operations to adequately identify the May 5 event and to report it as such, especially considering that it had a greater magnitude of severity than the April 23 event. The applicant's originally stated intent of including this event within the first PIR (110) appeared to be slow. In fact, PIR-89-129 was only written at the NRC's AIT insistence.

TU Electric Actions

TU Electric is taking actions to enhance the timeliness, reporting and evaluation of future events and equipment failures (see Section VII.C.2).

5. NRC Conclusions (4.1.5)

The out-of-sequence operation of valves in the May 5 event, occurring 12 days after a fundamentally identical out-of-sequence valve operation in the April 23 event, reflects a significant weakness in the applicant's ability to prevent an operational error from recurring.

TU Electric Actions

TU Electric is taking actions to improve adherence to plant procedures; aggressive documentation, reporting, and evaluation of events and equipment failures; and communications between shifts (see Sections VII.B.2, VII.C.2 and VII.C.3).

6. NRC Conclusions (4.1.6)

Sending only one auxiliary operator near the end of shift to operate valves 1AF-041 and 1AF-042 reflects a lack of understanding in the control room regarding task manpower requirements.

TU Electric Actions

TU Electric is taking several steps to provide additional assurance that Operations management and supervision are aware of manpower requirements for specific plant activities (see Section VII.C.1).

7. NRC Conclusions (4.1.7)

The AIT considers the difficulty of operation of valves 1AF-041 and 1AF-054 to be a contributing cause to the April 23 and May 5 events, but of minor safety significance. The AIT supports the applicant's intent to make these valves easier to operate.

TU Electric Actions

Actions are being taken to facilitate manipulation of remote-operated valves (see Section VII.B.2).

8. NRC Conclusions (4.1.8)

The evaluative process, which ultimately determined the root cause for the check valve failures appeared to be unnecessarily protracted in that it required almost six weeks from the inception of the AFW Task Team until the development of a definitive root cause and corrective action program. This protracted process, although not directly related to any regulatory requirement, is an example of the applicant's lack of management aggressiveness in the resolution of a safety-significant issue. This issue involved the multiple failures of passive components in a system intended to mitigate the consequences of an accident. For an NTOL plant, the applicant's response did not reflect the style of proactive Operations management philosophy normally associated with safe reactor plant operation. The AIT notes that when the applicant's Project Management took charge of the Task Team on May 26, 1989, efforts were significantly more timely and reflected a stronger commitment to corrective action. The applicant's Task Team went to the vendor Borg-Warner and made things happen. This aggressive attitude by management brought to light the root cause and brought about a corrective action plan in a timely manner.

TU Electric Action

TU Electric is taking action to improve the aggressiveness and timeliness of investigation of plant events and equipment failures and to improve future Task Team evaluations (see Section VII.C.2).

C. TU Electric Implementation of NRC Staff Recommendation

1. NRC Recommendation (4.2.1)

Create a minimum equipment list that would aid Operations personnel to make judgements regarding the effect of failed components on system operability.

TU Electric Action

Due to the number of modes of equipment failure and the fact that the significance of the failure of a specific piece of equipment is dependent on plant configuration and what other equipment remains operable, TU Electric does not believe that a reliable minimum equipment list can be created. Furthermore, because a minimum equipment list would not be comprehensive (anticipating the significance of every piece of equipment in every plant configuration), plant operators might place undue reliance on such a list and fail to perform probative analysis of the significance of equipment failure not on the minimum equipment list. TU Electric believes that equipment failures must be evaluated on a case-by-case basis. TU Electric is upgrading its program for the evaluation of equipment failure by requiring prompt review of the impact of maintenance work requests and additional engineering support for operation (see Sections VII.C.1, VII.C.2 and VII.C.4).

2. NRC Recommendation (4.2.2)

Assign system engineers the in-line task of reviewing all work requests related to a given system. The engineer would evaluate the impact of all component failures in regard to system operability.

TU Electric Action

Operators are being directed to request assistance from system engineers to help evaluate problems involving plant systems. Other actions are also being taken to enhance evaluations of Work Requests and impacts of equipment failures on operability (see Sections VII.C.2 and VII.C.4).

3. NRC Recommendation (4.2.3)

Provide training to control room personnel and supervisors regarding manpower requirements for certain types of plant evolutions.

TU Electric Action

Workshop training is being provided to Shift Supervisors on planning and controlling plant evolutions, including ensuring that manpower levels are adequate for routine evolutions (see Section VII.C.1).

4. NRC Recommendation

Provide continued emphasis on training plant personnel to comply with procedures. Steps are to be performed in sequence unless otherwise specifically approved.

TU Electric Action

The Shift Operations Manager has developed and implemented an action plan to enhance procedural compliance. As part of this plan, a memorandum on procedure compliance was provided to the Shift Supervisors, who in turn discussed the memorandum with their respective crews. The Manager, Operations and/or Shift Operations Manager also met and discussed the memorandum with each crew. Additionally, a workshop was held by the Manager, Operations with Operations Department Senior Reactor Operators (Shift Supervisors, Unit Supervisors, Shift Technical Advisors and Staff), including Training and Plant Evaluation personnel, to discuss the April 23 and May 5 events and procedure compliance. Emphasis on procedural compliance will continue to be emphasized in recurrent replacement training for operators (see Section VII.B.2).

5. NRC Recommendation (4.2.5)

Provide better communications between Operations staff, especially during shift changes.

TU Electric Action

TU Electric is taking several actions to enhance communications between operators and shifts (see Section VII.C.3).

6. NRC Recommendation (4.2.6)

Provide a large and conspicuous plant status board in the control room, sufficient to provide significant "night order" information and to facilitate the transfer of information between shifts.

TU Electric Action

TU Electric is implementing a system status program that may include the use, for example, of laminated prints that can be marked to indicate system or component status (see Section VII.C.4).

7. NRC Recommendation (4.2.7)

Initiate an immediate design revision to separate the 3-inch miniflow check valves from their associated orifices. The present configuration, if not corrected, lends itself to an exceptionally short lifespan for the check valves due to flow turbulence and valve tapping damage (see paragraph 2.3.3).

TU Electric Action

TU Electric is conducting evaluations to determine the effect of flow turbulence and valve tapping on the 3-inch miniflow check valves. Appropriate corrective action will be taken.

APPENDIX 2

CHECK VALVE BACKLEAKAGE TESTING

Results Engineering developed a specific test procedure to determine which check valves in the Feedwater and Auxiliary Feedwater Systems leak past their seats. The testing was initiated on April 28, 1989 and concluded that the check valves in the TDAFWP (1AF-078, 86, 98 and 106) and MDAFWP (1AF-075, 083, 093, 101) supply lines failed under backflow conditions. The check valves in the main feedwater upper penetration (1FW-195, 196, 197, 198, 199, 200, 201, 202) did not leak past their seats.

Performance and Test personnel tested the check valves in the TDAFW and MDAFW supply lines using ODA-408, "Nonstandard Alignments and Evolutions," procedures 1-89-053 and 1-89-055. The testing consisted of isolating the valve, connecting the upstream side of the valve to the nearest drain, pressurizing the downstream side of the check valve and measuring the decrease in pressure and flow across the valves after the upstream connections were opened. Results are as follows:

Test No.	Valve	GPM Leakage	Test Pressure (PSIG)
1-89-055	1AF-075	5.32	99
1-89-055	1AF-078	5.47	100
1-89-055	1AF-083	5.42	98
1-89-055	1AF-086	5.52	100
1-89-055	1AF-093	5.42	96
1-89-055	1AF-098	5.47	100
1-89-055	1AF-101	5.42	95
1-89-053	1AF-106	5.01	95

The AFW Pump Discharge Check Valves were tested by ODA-408 Procedure 1-89-058. The tests for the MDAFWP check valves 1AF-051 and 1AF-055 were performed by isolating the valves and pressurizing the downstream side. When the upstream test connection was opened, no leakage was detected. The TDAFWP check valve (1AF-038) was tested in a similar fashion except that the upstream test connection is on top of the pipe, so the vent was cracked open while covered with a soapy film to detect air displacement with the upstream pipe pressurized. Pressure on the upstream side could not be stabilized, although no air leakage was detected. The pressure problem was attributed to boundary valve leakage from a valve other than the check valve, and a radiograph (RT) performed on 1AF-038 and confirmed that it was closed. Results of the test of AFW pump discharge check valve are as follows.

Test No.	Valve	GPM Leakage	Test Pressure (PSIG)
1-89-058	1AF-038	0	50
	1AF-051	0	78
	1AF-065	0	71

The AFW miniflow recirculation check valves were tested by ODA-408 procedure 1-89-060. The test was performed by crosstieing the recirculation header to the pump discharge header to provide CST head pressure against the downstream side of the check valves. Leakage was collected at the upstream drain valve. Results are as follows.

Test No.	Valve	GPM Leakage	Test Pressure (PSIG)
1-89-060	1AF-057	7.81	21.5
1-89-060	1AF-069	0.0185	21.5

Because of inconvenient test connections, the recirculation check valve for the Turbine Driven Pump, 1AF-045, was not tested; instead radiography was used to determine the status of the valve. RT indicated that the valve was hung open. The low leakage rate through 1AF-069 is attributed to the reworking of the valve internals that was performed in response to the April 19, 1989 event.

The Main Feedwater pump discharge check valves were tested by ODA-408, "Non Standard Alignments and Evolutions," Procedure 1-89-059. The test was performed by isolating the valve and pressurizing the downstream side. When the upstream test connection was opened, leakage was collected. Results are as follows.

Test No.	Valve	GPM Leakage	Test Pressure (PSIG)
1-89-059	1FW-006	0.817	120
1-89-059	1FW-013	8.62	120

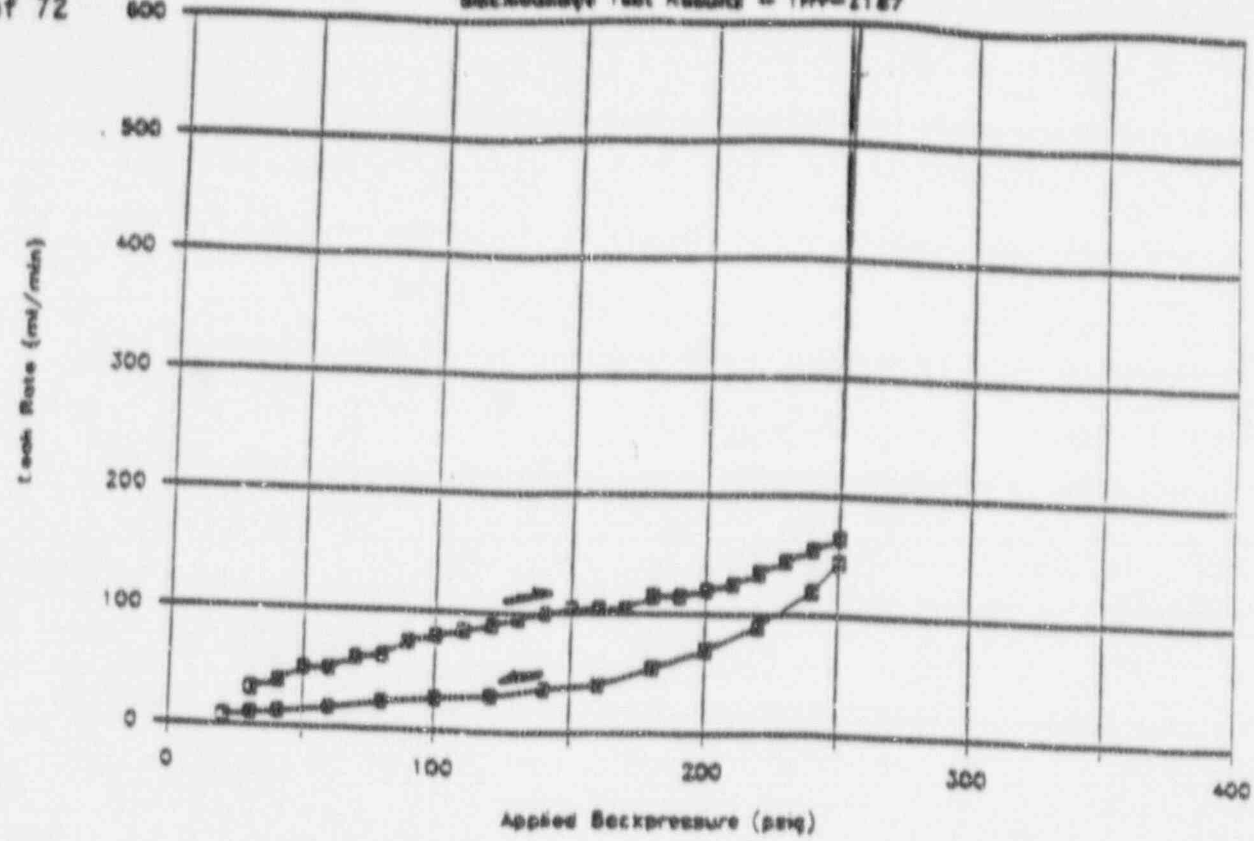
The FIBVs were tested by ODA-408 Procedure 1-89-068. The test was performed by applying pressure beneath the air operated bypass valve seat and charting the leakage as pressure was increased. The attached charts show leak rate through the FIBVs as a function of applied pressure. As these charts demonstrate, leak rates for each of the FIBVs (except the FIBV for SG 4) increased sharply when back pressures reached approximately 100 to 300 psi. From these charts, it was concluded that the FIBVs would have isolated against containment atmospheric design pressure as required by the design,[1] but that they were not sufficient to prevent backflow from the steam generators into the AFW System during conditions involving the higher pressures on April 23 and May 5. Therefore, the Task Team determined that the path of the backflow included the FIBVs.

In conclusion, the Task Team was able to determine which check valves in the Feedwater and Auxiliary Feedwater Systems failed where subject to backflow conditions. This determination was useful in establishing the backflow paths during the April 23 and May 5 events. Additionally, the Task Team determined that a number of check valves were subjected to backleakage.

[1]Although the FIBVs satisfied their design performance requirements, TU Electric is revising its procedures to require isolation of these valves when the main feedwater system is not in operation supplying flow to the steam generators. Additionally, TU Electric is conducting a review to determine whether similar valves exist in safety-related systems and whether additional protection would be provided by requiring isolation valves upstream of such valves to remain closed during particular plant conditions.

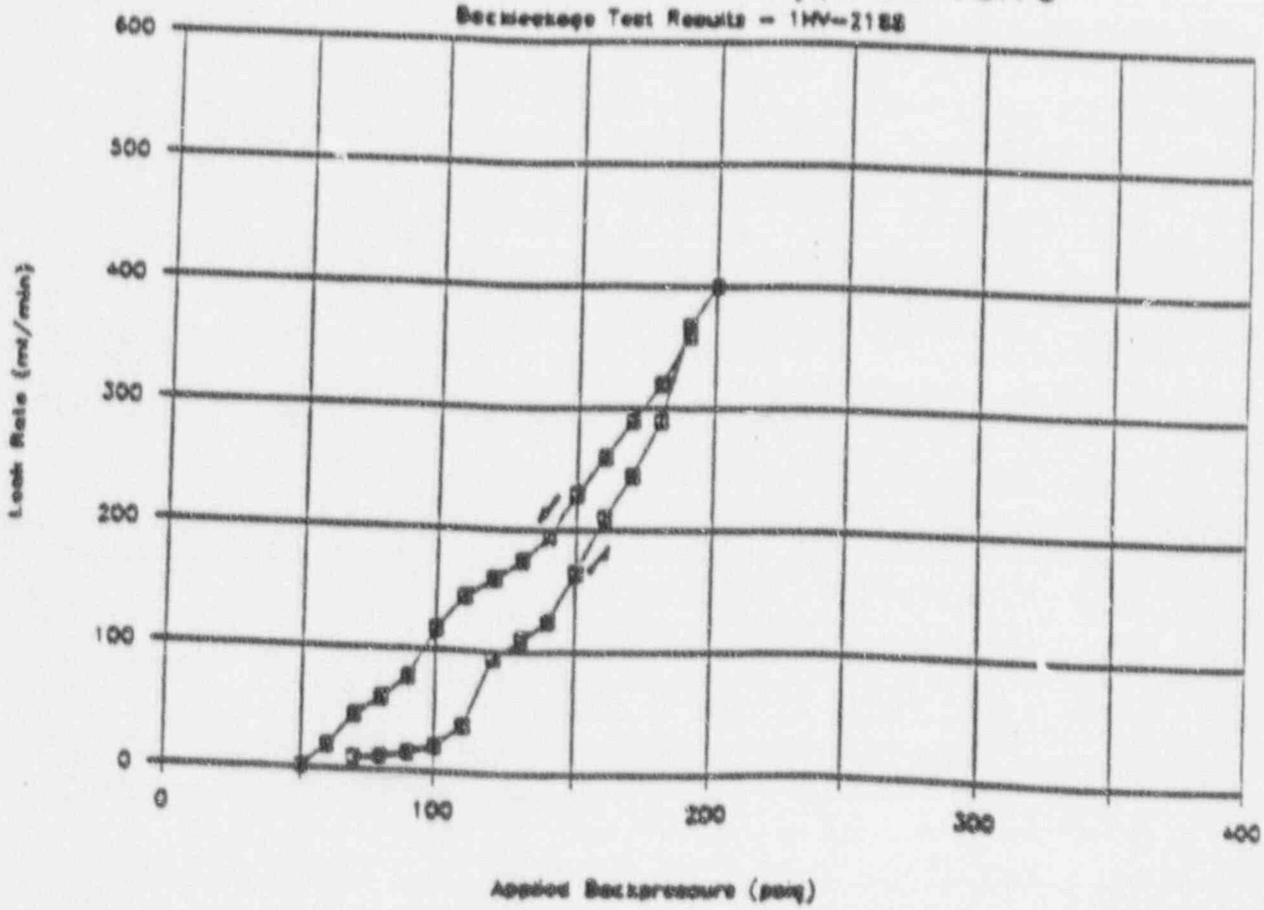
Feedwater Isolation Bypass Valve

Backleakage Test Results — 1HV-2187

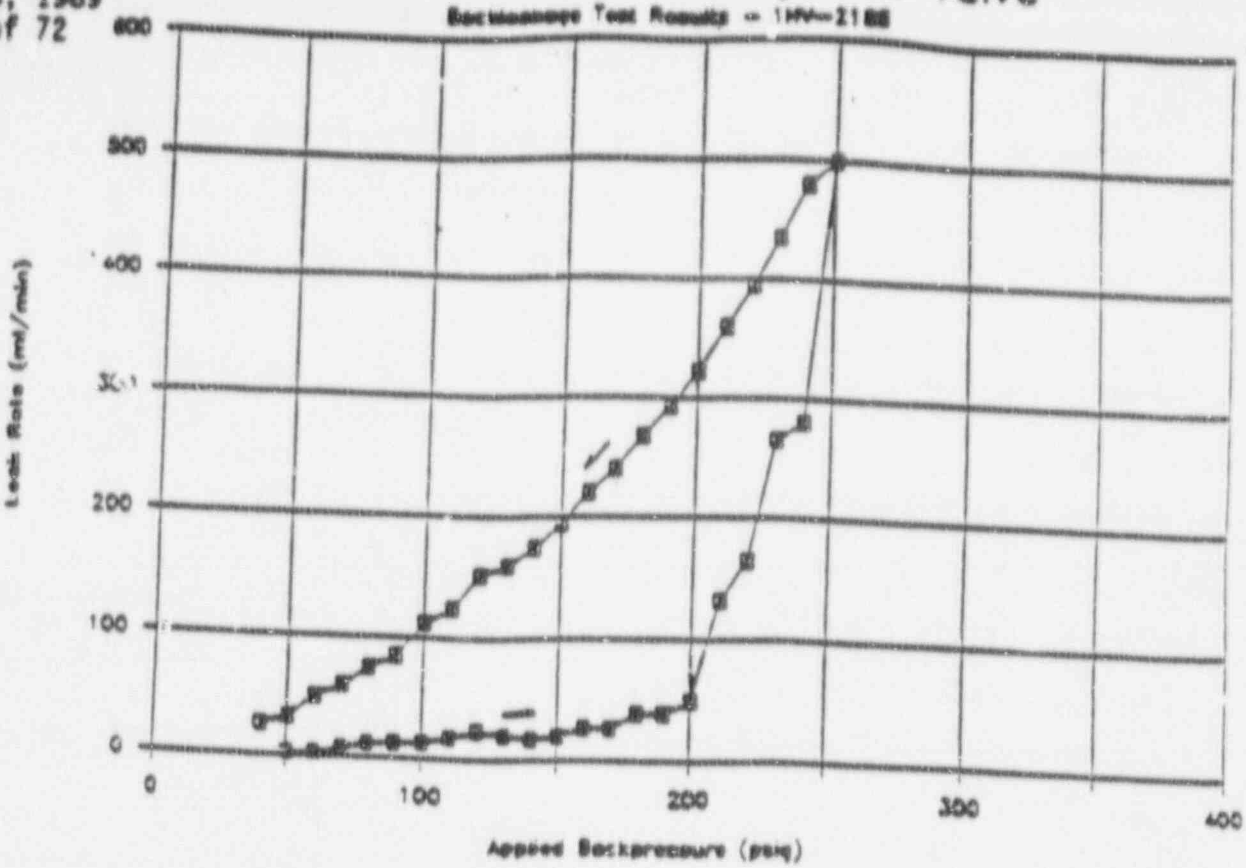


Feedwater Isolation Bypass Valve

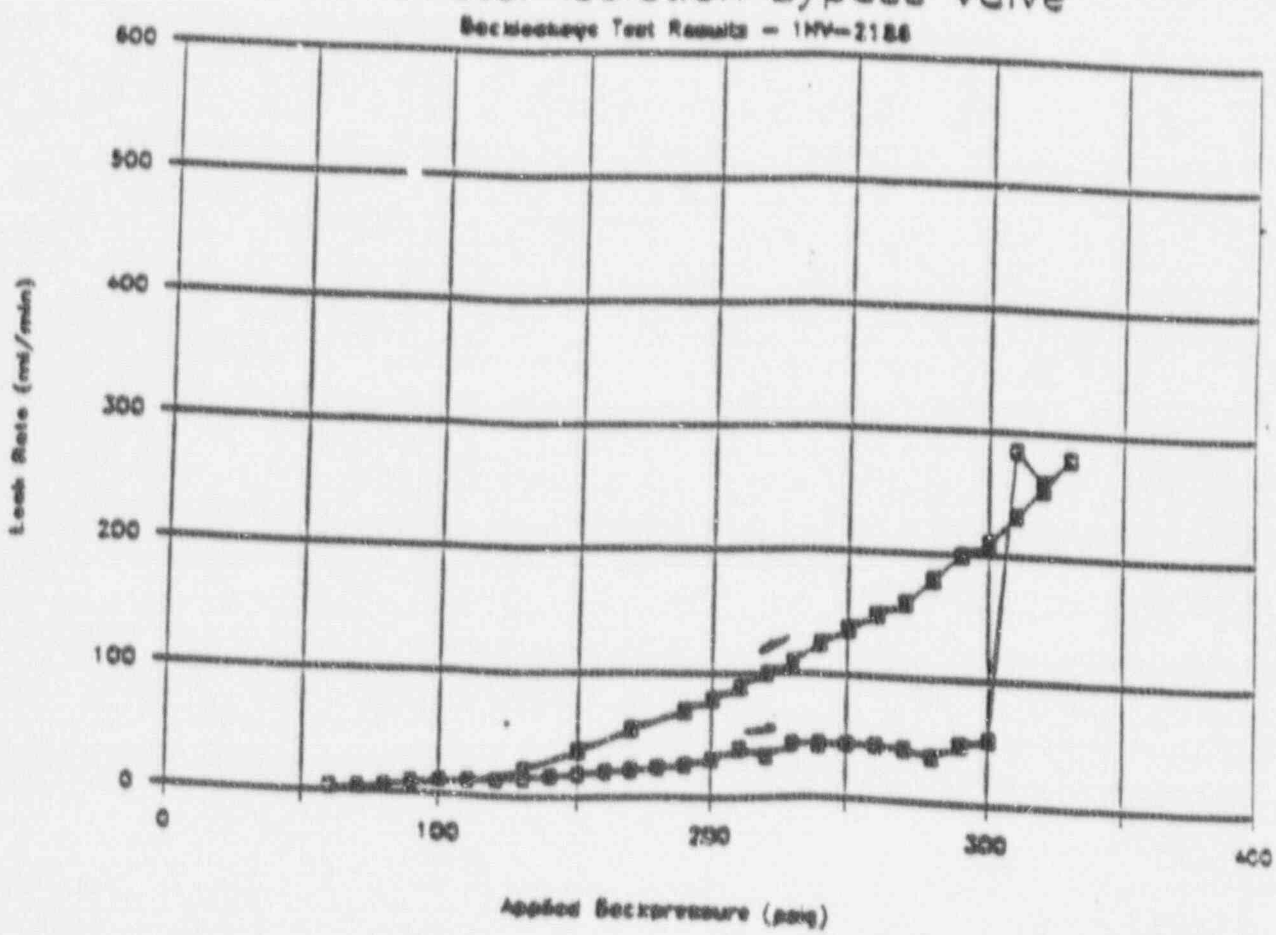
Backleakage Test Results — 1HV-2188



Feedwater Isolation Bypass Valve



Feedwater Isolation Bypass Valve



APPENDIX 3

CHECK VALVE MODIFICATION AND MAINTENANCE HISTORY

A search of the historical files was performed to determine if any onsite modification or maintenance performed on the valves could have been responsible for their recent failure.

A repair program of 1983 was of special interest. A modification had been made to replace tack welds holding the disc to disc stem, and disc stem to disc nut. This modification was made because of the potential for valve internals to come apart during operation, and as a result of a recommendation by BW/IP. Only three of the valves that failed tests in 1989 were modified on site during 1983. However, the valve internals had all been removed at one time or another in order to perform the necessary inspection during the 1983 modification. Additionally, during the past years, internals of some valves have been removed for routine system flushing. Valve internals, which were removed in 1983 or for subsequent flushing, were reassembled in accordance with CP-CPM-9.18 and CP-20B-1, "Borg Warner Maintenance Manual," the BW/IP Inspection Plan for Check Valves and MMI-1002, "Borg Warner Check Valve Inspection."

Review of documents indicated that the reassembly of the valves was performed in accordance with approved procedures. In addition, records show that QA control and QC verification was properly applied to each activity. Also, a representative of BW/IP was present during the 1983 modification. There is no evidence of noncompliance to these procedures. However, the procedures lacked adequate detailed instructions to ensure proper reassembly because they did not provide instructions for aligning the valve disc and seat. Therefore, the Task Team concluded that the BW/IP check valves were improperly reassembled due to the inadequate assembly instructions based on vendor information.

In addition to the work performed in 1983, check valves have been subject to other maintenance and modification activities. The attached table lists each Unit 1 and Common BW/IP check valve and its maintenance and modification history. The information in the attached table was compiled from a review of work travelers, Inspection Removal Notices and QC inspections associated with these valves. Additionally, previous work documents, including Nonconformance Reports (NCRs), Problem Reports (PRs) and PIRs were reviewed for any unusual trends or noncompliances with specifications. In order to determine whether any trends existed, characteristics of each valve and its associated maintenance and modification activities were identified and placed into one or more categories. These categories included:

o Size - nominal pipe size.

o Rating - the pressure rating of the valve.

- o Internals removed for inspection - if valve internals were removed for inspection during the 1983 overall repair program.

- o Disc Assembly Modification - if full fillet welds were not present, and the disc assembly was taken apart or modified in any way to make the recommended repair.

- o Owners form NIS-2 - if this form was present, it ensured that the repair was performed on site by Brown & Root.

- o Bushing Modifications - if the axial clearance was changed at any time for any reason other than the 1983 modification.

- o Internals removed for flushing - if the internals had ever been removed by operators and/or maintenance activities other than the above.

- o Downstream of an orifice - if the check valve is operating in the area of an orifice.

- o Post work inspection by BW/IP - if any indication was given in the modification documents that the vendor was present for the modification work or made a separate inspection at a later date.

- o Separate passivitization - if the internals were ever removed strictly for removal of rust. Note that rust removal was performed in conjunction with the 1983 modification.

- o Internals transferred - if the internals of the valve as it is now installed differ from those originally shipped with the valve.

- o Valve failing - if the valve was shown through testing not to hold back pressure, or if, through radiograph, it was determined to be restricted from closing.

As the attached table demonstrates, most of the categories do not exhibit any correlation with valve failures. For example, none of the valves that failed during testing had been subject to separate passivitization, transfer of valve internals, or bushing modifications to adjust clearances, and only three of the thirteen failed valves had been subject to modification. Therefore, the Task Team concluded that these maintenance and modification activities were not the cause of the backleakage.

The table also identifies a correlation, in either whole or part, between valve failure and four categories; 1) valve size of 3" or 4" inches; 2) internals removed for inspection; 3) internals removed for flushing[2] and 4) valve downstream of orifice/turbulence. The first three of these factors all indicate that inadequate vendor assembly instructions were the cause of the valve failure; i.e., the inadequate instructions only pertained to 3" and 4" inch valves, and the inadequate instructions were used during reassembly following removal of valve internals. With respect to the last category, valve downstream of orifice/turbulence, TU Electric is considering an inspection program for valves near orifices and is evaluating the need to move these orifices, as is discussed in Section VII.B.1 and in Appendix 6.

[2] In some cases, it was necessary (and common practice in the industry) to remove the internals of a check valve in order to perform high velocity flushing of a piping system. In addition, in a few cases, the internals of a check valve were removed to facilitate draining of a system. The occasional removal and reassembly of valve internals does not adversely affect the function of a valve, provided that these activities are performed properly.

VALVE	SIZE	RATING #	INTERNALS REMOVED FOR INSPECTION	DISK MODIFIED BY WELDING/MACHINING	OWNERS FORM NIS-2 ON FILE	BUSHING MODIFICATION TO ADJUST CLEARANCES	INTERNALS REMOVED FOR FLUSH/BONNET LEAKAGE	DOWNSTREAM OF ORIFICE/TURBULENCE	POST WORK INSPECTION BY BW/IP	SEPARATE PASSIVITIZATION	VALVE FAILING	
1AF-009	3"	150	X	X	X	X	X					
-014	6"	150	X	X	X		X		X			
-024	6"	150	X	X	X		X		X			
-032	8"	150	X	X	X		X		X			
-038	8"	900	X	X	X		X					
-045	3"	900	X					X	X		X	
-051	6"	900	X	X	X				X			
-057	3"	900	X	X	X			X	X		X	
-065	6"	900	X	X	X		X					
-069	3"	900	X	X	X			X	X		X	
-075	4"	900	X					X	X		X	
-078	4"	900	X	X	X		X	X	X		X	
-083	4"	900	X					X	X		X	
-086	4"	900	X				X	X	X		X	
-093	4"	900	X					X	X		X	
-098	4"	900	X				X	X	X		X	
-101	4"	900	X					X	X		X	
-106	4"	900	X				X	X	X		X	
-167	8"	150	X	X	X		X					

VALVE	SIZE	RAT.	INTERNALS REMOVED FOR INSPECTION	DISK MODIFIED BY WELDING/MACHINING	OWNERS FORM NIS-2 ON FILE	BUSHING MODIFICATION TO ADJUST CLEARANCES	INTERNALS REMOVED FOR FLUSH/BONNET LEAKAGE	DOWNSTREAM OF ORIFICE/TURBULENCE	POST WORK INSPECTION BY BW/IP	SEPARATE PASSIVITIZATION	VALVE [illegible]ILING	
1CC-0003	3"	150	X	X	X							
-0004	3"	150	X									
-0317	10"	150	X	X	X							
-0602	4"	150	X									VENDOR WELDED
-0651	4"	150	X									VENDOR WELDED
-0690	4"	150	X	X	X	X						
-0693	4"	150	X	X	X	X						
-0697	4"	150	X									
-0713	8"	150	X	X	X							
1DD-006	3"	150	X	X	X							
-018	3"	150	X	X	X	X						
-021	3"	150	X	X	X	X						
XDD-048	3"	150	X									
1DD-065	3"	150	X									
XDD-104	3"	150	X				X					
XVA-001	6"	150	X									
-002	6"	150	X									
1CA-0016	3"	150	X	X	X		X					

VALVE	SIZE	RATING #	INTERNALS REMOVED FOR INSPECTION	DISK MODIFIED BY WELDING/MACHINING	OWNERS FORM NIS-2 ON FILE	BUSHING MODIFICATION TO ADJUST CLEARANCES	INTERNALS REMOVED FOR FLUSH/BONNET LEAKAGE	DOWNSTREAM OF ORIFICE/TURBULENCE	POST WORK INSPECTION BY BW/IP	SEPARATE	PASSIVITIZATION	VALVE FAILING	
1CT-013	10"	150	X							X			
-025	16"	150	X	X						X			
-042	10"	150	X	X	X		X			X			INTERNALS TRANSFERRED
-047	4"	300	X				X						
-048	4"	300	X				X						
-063	4"	300	X				X						
-064	4"	300	X				X						
-065	10"	150	X	X	X		X						
-077	16"	150	X	X	X								
-094	10"	150	X	X	X		X						
-142	16"	150	X	X	X					X			
-145	16"	150	X	X	X					X			
-148	16"	300	X	X	X					X			
-149	16"	300	X	X	X					X			
1CH-024	6"	150	X	X	X		X						LEAKED AT ILRT EVEN POST REWORK
-343	3"	150	X	X	X		X						
-346	3"	150	X				X						VENDOR WELDED
-403	3"	150	X				X						VENDOR WELDED
-406	3"	150	X	X	X	X							
-453	3"	1	X										
-463		1	X										

VALVE	SIZE	RATING #	INTERNALS REMOVED FOR INSPECTION	DISK MODIFIED BY WELDING/MACHINING	OWNERS FORM NIS-2 ON FILE	BUSHING MODIFICATION TO ADJUST CLEARANCES	INTERNALS REMOVED FOR FLUSH/BONNET LEAKAGE	DOWNSTREAM OF ORIFICE/TURBULENCE	POST WORK INSPECTION BY BW/IP	SEPARATE PASSIVITIZATION	VALVE FAILING
1FW-191	6"	900	X								
-192	6"	900	X								
-193	6"	900	X								
-194	6"	900	X								
-195	6"	900	X								
-196	6"	900	X								
-197	6"	900	X								
-198	6"	900	X								
-199	6"	900	X								
-200	6"	900	X								
-201	6"	900	X								
-202	6"	900	X								
1MS-142	4"	900	X				X				X
-143	4"	900	X	X	X		X				X
XSF-003	10"	150	X	X	X						
-004	10"	150	X	X	X						
-160	3"	150	X	X	X						
-180	3"	150	X	X							

VALVE	SIZE	RATING #	INTERNALS REMOVED FOR INSPECTION	DISK MODIFIED BY WELDING/MACHINING	OWNERS FORM NIS-2 ON FILE	BUSHING MODIFICATION TO ADJUST CLEARANCES	INTERNALS REMOVED FOR FLUSH/BONNET LEAKAGE	DOWNSTREAM OF ORIFICE/TURBULENCE	POST WORK INSPECTION BY BW/IP	SEPARATE PASSIVITIZATION	VALVE FAILING	
1SV-014	4"	150	X	X	X	X	X					
-016	10"	150	X	X								
-017	10"	150	X									
-048	4"	150	X	X								
-388	10"	150	X	X	X							
-389	10"	150	X									VENDOR WELDED
1CT-030	3"	150	X				X					

APPENDIX 4

IDENTIFIED MATERIAL CONCERNS

An unrelated deficiency pertaining to BW/IP check valves that occurred during Station Service Water System testing was identified during the Task Team investigation. A swing arm on check valve 1SW-048 failed because of a pre-existing flaw. The preliminary indication of the failure mode on 1SW-048 was the presence of a pre-existing flaws and hot cracking resulting from improper casting and/or heat treatment coupled with the aggressive chemistry of the Service Water System. Analysis of two other swing arms from Unit 2 (2SW-0048 and 2CT-0149) did not reveal the same type of flaws that were present in the failed swing arm but did suggest a potentially insufficient heat treatment. The two swing arms destructively examined were subject to a relatively low service stress. In addition three more intact swing arms (which have seen varying degrees of service) have been destructively examined and show no signs of any preexisting flaw.

In order to firmly establish the condition of all the Unit 1 swing arms, all the valves will be non-destructively examined. The examination will consist of:

1. Visual 10x inspection

2. Wet fluorescent penetrant particle testing

3. Replication[1] on two zones of each arm

In addition, an evaluation of the porosity observed in the clevis of a spent fuel valve (1XSF-004) was performed by the manufacturer. This evaluation consisted of an x-ray to determine extent of porosity and a review of the design calculations. The review concluded that the part was satisfactory for its intended service.

In addition, an engineering evaluation is being performed to determine the maximum amount (size) of porosity which could be accepted without exceeding allowable stress in the remaining cross-section. A preliminary review of the stress in the clevis indicated an extremely low service stress (approximately 6 ksi is imposed) when compared to the allowable stress (approximately 34 ksi).

The material deficiencies were reported to the NRC on June 26, 1989 as potentially reportable under 50.55(e). Safety significance of these deficiencies are still under evaluation and will be reported to the NRC as part of SDAR CP-89-19.

[1]Replication is a process by which a surface is polished and an acetate tape is applied, peeled off and microscopically examined. This provides a topological examination in which hot cracks can be detected.

APPENDIX 5

RADIOGRAPHY, INSPECTIONS, AND COMPUTER ASSISTED
DRAWINGS FOR BW/IP CHECK VALVES

The Task Team utilized radiography (RT), inspections, and Computer Assisted Drawings to help determine the cause of the backleakage through the BW/IP check valves. The results of these activities are discussed below.

Radiography

Twenty-one check valves were radiographed. Ten of these valves appeared to be hung open (i.e., the top of the disc hung up under the seat lip at the 12 o' clock position). Of the ten open valves, eight were four-inch AFW valves and two were three-inch AFW pump recirculation valves. Two other four-inch valves (1MS-142 and 1MS-143) appeared to be seated improperly. Although the disc in these two valves did not appear to be lodged under the seat, the discs were not in contact with the seats over the lower halves. The remaining nine check valves appeared to be properly closed. The attached table provides specific valve radiograph results.

Radiographing these valves played a key role in the identification of the root cause of the backflow. This technique showed that there was a difference between seat/disc elevation and that the disc was lodged beneath the seat lip.

Inspections

Fourteen of the radiographed valves were disassembled and inspected. The fourteen inspected valves included the twelve valves that were determined to be open as a result of the radiographs. The attributes subject to inspection included axial play, seat angle, proper alignment, machining of the disc edge, and retainer position. The attached table shows the results of these inspections. As this table demonstrates, there does not appear to be any correlation between the inspected attributes and the valves that were determined to be open. For example, the inspected valve with the largest amount of axial play in the disc (valve 1FW-198) was closed, while other valves with less axial play were open. Therefore, the Task Team concluded that none of these attributes, in and of its' self, was the root cause of the hung open valve discs.

Computer Assisted Drawings

Using CADs, 2D and 3D drawing models were created for the as-found condition of 1AF-106 (4" 900# Pressure Bonnet Swing Check Valve) and 1FW-198 (6" 900# Pressure Bonnet Swing Check Valve).

These drawing models simulated the potential for hang up and improper closure of the check valves. The models were prepared with dimensions obtained from manufacturing drawings and dimensions taken from disassembled valves. Also, input was obtained from the BW/IP representative onsite. The models demonstrated that variation in disc elevation (and to some extent disc stud axial play) affects valve operation.

Conclusions

Based upon the radiographs and the CADS, the Task Team determined that the backleakage through the check valves was caused by hung open valve discs due to an elevation difference between the valve disc and seat (and, to a much lesser extent, excessive axial play). Using this information, the Task Team reviewed the vendor manual for the BW/IP check valves and determined that the manual did not provide adequate instructions for ensuring that the valve disc is at the same elevation as the valve seat, and that BW/IP had not provided acceptance criteria for axial play.

VALVE		SIZE	RATING	RT COMPLET	RT RESULTS	PROPER ALIG	RETAINER POS	SEAT ANGLE	DISC EDGE MACHINED	FILLET WEL MACHINED	AXIAL PLAY
1AF-009	3"	150									
1AF-014	6"	150									
1AF-024	6"	150									
1AF-032	8"	150									
1AF-038	8"	900									
1AF-045	3"	900	X	[illegible]	N	0	7.7	N	N	.223	
1AF-051	6"	900									
1AF-057	3"	900	X	OPN	Y	0	5	Y	N	.142	
1AF-065	6"	900									
1AF-069	3"	900	X	CLSD	Y	0	83	Y	N	.078	
1AF-075	4"	900	X	OPN	Y	0	12	N	N	.165	
1AF-078	4"	900	X	OPN	N	0	5	Y	N	.180	
1AF-083	4"	900	Y	OPN	Y	0	[illegible]	[illegible]	N	.206	
1AF-086	4"	900	[illegible]	OPN	Y	[illegible]	5	Y	N	.193	
1AF-093	4"	900	X	OPN	Y	0	12	N	N	.265	
1AF-098	4"	900	X	OPN	N	0	5	Y	Y	.147	
1AF-101	4"	900	Y	OPN	Y	0	5	N	N	.210	
1AF-106	4"	900	X	OPN	N	0	5	N	N	.197	
1AF-167	8"	150									
1FW-191	6"	600									
1FW-192	6"	600									
1FW-193	6"	600									
1FW-194	6"	600									
1FW-195	6"	600	X	CLSD							
1FW-196	6"	600	X	CLSD							
1FW-197	[illegible]	600	X	CLSD							
1FW-198	[illegible]	600	X	CLSD	Y	.150	5	Y		.315	
1FW-199	6"	600	X	CLSD							
1FW-200	6"	600	X	CLSD							
1FW-201	6"	600	X	CLSD							
1FW-202	6"	600	X	CLSD							
1MS-142	4"	900	X	OPN	Y	0	3.2	Y	Y	.194	
1MS-143	4"	900	X	OPN	Y	0	3.8	Y	N	.124	
1CT-048	4"	300	X	*							*RT UNCLEAR
1CT-094	6"	300	X	*							*RT UNCLEAR

APPENDIX 6

EVALUATION OF AFW CHECK VALVES AGAINST EPRI GUIDELINES

The BW/IP check valves in the CPSES AFW System were evaluated against the criteria in EPRI Report NP-5479, "Application Guidelines for Check Valves in Nuclear Power Plants," to determine whether any inconsistencies between the CPSES check valves and the EPRI guidelines may have resulted in the backleakage through the CPSES check valves.

The EPRI Report states that the following six factors should be considered in determining the application of check valves: 1) valve sizing; 2) valve closure time; 3) structural compatibility; 4) valve seat leakage limits; 5) valve orientation; and 6) piping arrangement. The results of the Task Team's evaluation of the CPSES AFW check valves against each of these factors is discussed below.

1. Valve Sizing

   The 4" and 6" AFW check valves showed no sign of wear associated with improper sizing. Therefore, the Task Team concluded that valve sizing was not a cause of the BW/IP check valve failures.

2. Check Valve Closure Time

   AFW System design does not require any specific check valve closure times. Closure times were not a factor in the AFW check valve failures.

3. Structural Compatibility

   EPRI guidelines recommend a margin in pressure boundary thickness to account for wastage due to erosion/corrosion. The minimum valve thickness for this margin is dependent on design pressure and temperature, the ANSI pressure rating, and the ANSI body thickness of the valve. Under the EPRI guidelines, the minimum valve body thickness should be 0.411" for a typical 4" valve in an AFW supply line to the steam generators (conservatively assuming system design temperature is 200°F). The minimum body thickness for a 4" AFW valve at CPSES is 0.509". Therefore the BW/IP check valves that were evaluated conformed with the EPRI guidelines.

4. Seat Leakage Limits

   Backleakage through the AFW check valves on April 23 and May 5 was not caused by seat leakage, but instead by hung open discs. Therefore, this factor is not relevant to the root cause of the check valve backleakage.

5. Valve Orientation

EPRI guidelines state that swing check valves should be installed in horizontal runs. Check valves in the CPSES AFW System have been installed in horizontal runs. Therefore, orientation is not a consideration in the check valve failures.

6. Piping Arrangement

EPRI guidelines recommend that check valves be located at least 5 pipe diameters downstream of fittings such as elbows and tees and 10 diameters downstream of in-line disturbances such as pumps, control valves, and orifices. The following table lists all of the check valves in the AFW System and their proximity to upstream disturbances.

Valve	Nearest Upstream Component	# of Pipe Diameters In Between	Valve Failure	Disc Open
1AF-069	Breakdown Orifice	3	Yes	Yes[3]
1AF-057	Breakdown Orifice	3	Yes	Yes
1AF-045	Breakdown Orifice	6	Yes	Yes
1AF-075	Flow Orifice	7 1/2	Yes	Yes
1AF-083	Flow Orifice	7 1/2	Yes	Yes
1AF-093	Flow Orifice	6 1/8	Yes	Yes
1AF-101	Flow Orifice	6 1/4	Yes	Yes
1AF-078	Flow Orifice	4 1/2	Yes	Yes
1AF-086	45° Elbow	0	Yes	Yes
	Flow Orifice	4 3/4	Yes	Yes
1AF-098	Flow Orifice	4 3/8	Yes	Yes
1AF-106	Flow Orifice	4 1/2	Yes	Yes
1AF-024	Globe Valve	19	No	N/A
1AF-014	Globe Valve	19	No	N/A
1AF-051	90° EL	2 5/8	No	N/A
1AF-032	Globe Valve	2 1/3	No	N/A
1AF-065	90° EL	3 1/8	No	N/A
1AF-038	Enlarger 90°	2 3/4 - 2 1/2	No	N/A

As indicated above, many of the AFW check valves at CPSES are closer to upstream fittings and other devices than recommended by EPRI. The majority of these valves also exhibited backleakage under test conditions and were determined to be hung open as a result of radiographs. However, the Task Team concluded that the proximity of the check valves and upstream fittings and devices was not a factor in the backleakage through the check valves on April 23 and May 5. Although proximity between the valves and upstream fittings and devices might result in increased turbulence at the check valve, such turbulence would not cause the valve disc to hang up.

Attachment I

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

FEB 16 1990

In Reply Refer To:
Dockets: 50-445/90-03
50-446/90-03

Mr. W. J. Cahill, Jr.
Executive Vice President
TU Electric
400 North Olive Street, Lock Box 81
Dallas, Texas 75201

Dear Mr. Cahill:

This refers to the inspection conducted by Messrs. R. M. Latta, M. F. Runyan, and other NRC inspectors and consultants during the period January 3 through February 6, 1990, of activities authorized by NRC Construction Permits CPPR-126 and CPPR-127 for the Comanche Peak Steam Electric Station, Units 1 and 2, and to the discussion of our findings with you and members of your staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

During this inspection, it was found that certain of your activities were in violation of NRC requirements, as specified in the enclosed Notice of Violation. A written response to these violations is required.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC Public Document Room.

The response directed by this letter and the accompanying Notice is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Attachment H

RFWarnick

R. F. Warnick, Assistant Director
for Inspection Programs
Comanche Peak Project Division
Office of Nuclear Reactor Regulation

Enclosures: See next page.

~~9002270448~~

Calculation Change Notice 002 to Calculation 16345-CS(B)-178, Revision 3. The NRC inspector also witnessed a demonstration of the ultrasonic test technique used to determine weld penetrations. Based on the fact that the discrepant welds and poor fit-up inspections were shown to be isolated and that the affected platform was structurally adequate in spite of the discrepant welds, the NRC inspector concluded that the applicant had taken adequate action to resolve this item. This open item is closed.

f. (Closed) Unresolved Item (445/8965-U-04): During the NRC review of the applicant's room, area, and system turnover programs, several questions were raised concerning the overall adequacy of these programs to identify and correct hardware discrepancies which remained after the completion of construction. This unresolved item tracked the NRC's continuing assessment of these programs. Previous NRC inspection of this issue is documented in NRC Inspection Reports 50-445/89-65, 50-446/89-65; 50-445/89-76, 50-446/89-76; and 50-445/89-89, 50-446/89-89. NRC Inspection Report 50-445/89-89, 50-446/89-89 documents the NRC's final acceptance of the applicant's turnover programs. All issues associated with this unresolved item were resolved in this previous NRC review. Consequently, this unresolved item is closed.

g. (Open) Open Item (445/8973-O-04): Following the AFW check valve failures (NRC Inspection Report 50-445/89-30; 50-446/89-30), the applicant developed an inspection and reassembly procedure and post-installation test procedures to demonstrate the operability of Borg-Warner check valves. In several instances, the post-installation backflow tests failed to meet the acceptance criteria, revealing areas that had not been fully corrected by the original procedures. This open item addressed the root cause analyses and generic implications of these second-generation check valve failures.

A summary of the suspected root cause and the corrective action taken for each check valve failure is provided below:

<u>Valve 1AF-0083</u> (valve body/bonnet) was rotatively misaligned and the disc-stud was bent. A new disc-stud assembly was installed, the valve internals were reinstalled, and the reverse flow leak testing was satisfactory.

<u>Valve 1CA-0016</u> exhibited excessive seat leakage. The swing arm and bushing were replaced and the valve was blue

checked. The valve internals were reinstalled and the subsequent reverse flow leak testing was satisfactory.

Valve 1AF-0057 exhibited unacceptable valve body/bonnet rotational misalignment and incorrect bonnet elevation. The valve was disassembled and supplemental measurements were taken, the valve internals were reinstalled using the new height specification, and the valve was successfully tested in the reverse flow direction.

Valve 1SW-0048 was determined to have an excessively long swing arm bushing. The bushing length was reduced by 0.08" and replaced in the disc-stud assembly. The valve internals were reinstalled and the valve was successfully tested in the reverse flow direction.

Valve 1MS-143 was determined by radiography to have the disk lodged under the seat ring. The disk had apparently become lodged under the seat during the reassembly process. The valve did not experience forward flow after the reassembly process. The valve was disassembled and then reassembled taking care to ensure that the disk did not lodge under the seat. The reverse flow (air) test was then successful. A reverse flow steam test will be conducted in Mode 3.

In conjunction with the above documented activities, the applicant has revised the Borg-Warner check valve reassembly procedure and designed a specialized set of tools to allow for the establishment of more precise rotational alignment of the bonnet to the valve body. The NRC inspector witnessed a demonstration of the new tools and technique in the mechanical maintenance shop and the reassembly of valve 1AF-045 in the plant. The NRC inspector concluded that the new procedure will enhance the rotational alignment between the valve bonnet and body.

* Approximately 13 Borg-Warner check valves in the auxiliary feedwater and feedwater systems were identified by the applicant as having excessive body to bonnet external leakage. These valves were disassembled, honed to remove scratches in the valve body throat and provide better sealing surfaces, and reassembled. In most cases, this corrective action essentially stopped the leakage. Several check valves, including 1AF-038, which continue to leak, are scheduled to be "hot torqued" in Mode 3. The applicant anticipates that the extra pressure will seal the valve. Each valve that was disassembled was retested for backleakage upon reassembly with satisfactory results.

This open item will be left open pending successful Mode 3 testing of valve 1MS-143 and demonstration that the hot torquing referenced above corrects the remaining body to bonnet leakage problems.

h. (Closed) Open Item (445/8973-O-06): This item addressed the apparent lack of adequate flushing capability in the auxiliary feedwater (AFW) system using existing drains. This concern resulted from NRC interviews, conducted within the Augmented Inspection Team (AIT) inspection documented in NRC Inspection Report 50-445/89-30; 50-446/89-30, during which plant personnel stated that check valve internals were routinely removed to provide the appropriate drain paths. At the time, both NRC and the applicant speculated that the frequent disassembly and reassembly of check valves may have contributed to their eventual failure.

*

The applicant's response to this issue is documented in TU Electric memorandum CPSES-9001379, Davis to Guldemond. This document presents the following points:

(1) The startup practice of using check valves for flush exit/entrance points is an industry accepted evolution.

Not to provide drains until first fuel exchange?

*

(2) Check valve failure was due to inadequate installation procedures in the Borg-Warner instruction manual and was not related to the frequency with which these procedures were used.

(3) Additional drains and vents will be installed during the Unit 1 first refueling outage to facilitate the planned periodic inspections of Borg-Warner check valves.

The NRC inspector agreed that the frequency with which check valves were used as drain and vent points was not a contributing cause of the AFW backflow events. The applicant's intent to install new drains and vents and the fact that the plant is moving into the operations phase should greatly lessen the need in the future to utilize check valves in this manner. This open item is closed.

i. (Closed) Open Item (445/8973-O-07): This item identified the NRC's concern that no apparent provisions were made for continued maintenance and system preservation for the AFW system during the period from completion of preoperational testing in 1984 until completion of hot functional testing in 1989. This perception was based on NRC reviews of maintenance histories and discussions with personnel during the AIT inspection documented in NRC Inspection Report 50-445/89-30; 50-446/89-30.

The applicant stated that maintenance and preservation of the AFW system during this time period was controlled by Procedure MDA-301, "Protective Maintenance Program," and Procedure MEI-043, "Performance of Activities Required by ANSI N45.2.2." Procedure MEI-043 applies to equipment installed in the plant but not operational. The applicant provided a list of work orders on the AFW system covering late 1985 to late 1989 which included some preservation activities such as oil changes, filter examinations, inspection of bearings, "major" inspections, and "teardown" inspections. The applicant stated that the AFW system was in wet lay-up with adequate concentration of hydrazine to prevent corrosion until December 1986 when the system was placed in dry lay-up. Hydrazine was also used in dry lay-up for those areas which could not be drained.

The NRC inspector reviewed Procedures MDA-301 and MEI-043 and information regarding lay-up conditions of the AFW system. It appears that maintenance and preservation of the AFW system, though not extensive, was adequate to ensure the continued operability of the system. This open item is closed.

j. (Closed) Open Item (445/8973-O-08): During the auxiliary feedwater (AFW) backflow events (see NRC Inspection Report 50-445/89-30; 50-446/89-30), steam generator water flowed in the reverse direction through the feedwater isolation bypass valves and in the forward direction through the preheater bypass valves to the AFW piping. The applicant informed the NRC of their intent to administratively isolate the feedwater isolation bypass valves during startup and shutdown conditions to preclude the possibility for similar backflow events in the future. The applicant has revised Procedure IPO-004A (Revision 3), "Plant Shutdown from Minimum Load to Hot Standby," and Procedure IPO-002A (Revision 4), "Plant Startup from Hot Standby to Minimum Load," to require the feedwater isolation bypass valve downstream manual isolation valves to remain closed whenever the AFW system is being used to feed the steam generators. On startup these manual valves are opened upon transfer from the AFW system to the main feedwater system, and on shutdown the valves are closed on transfer back to the AFW system. If operators adhere to these administrative controls, backflow events similar to those experienced on April 23 and May 5, 1989, should not recur.

As a backup, the applicant has also revised Procedures IPO-004A and IPO-002A to require closure of the preheater bypass valves whenever the AFW system is providing feedwater to the steam generators. In order to effect this change, the applicant had to modify the interlock between the preheater bypass valves and the feedwater isolation valves. Design Change Authorization (DCA)-92571 was issued to reconfigure contacts to permit the preheater bypass valves to remain closed when its control switch is in the closed position regardless of the position of the feedwater isolation valves. The interlock between these two valves is restored when the preheater bypass valve control is returned to "AUTO." The preheater bypass valves will provide a redundant pressure boundary to prevent backflow from the steam generators to the AFW system.

?

The NRC inspector reviewed the revisions to Procedures IPO-004A and IPO-002A, DCA 92571, and relevant changes made to DBD-ME-203, "Feedwater System," and concluded that the applicant has taken sufficient action on this item. This open item is closed.

k. (Closed) Open Item (445/8973-O-10): This item addressed the applicant's evaluation of the human factors associated with remote valve operators. Valves 1AF-041 and 1AF-054 (AFW pump discharge isolation valves), due to the difficulty of operating their reach-rod valve operators, indirectly contributed to the AFW backflow events reported in NRC Inspection Report 50-445/89-30; 50-446/89-30. These valves required approximately 30 minutes to close from full open or to open from full closed. The applicant conducted a plant walkdown to locate and evaluate all valves operated with reach-rod operators. In Unit 1 and common, 398 valves were checked, of which 190 were safety related. Each valve was checked for labeling, stroke time, ease of operation, number of turns per stroke, accessibility, and direction of operation. Each valve checked was determined to be operable and the eight safety-related valves which could not be operated (due to plant conditions) were judged to be operable based on comparison with similar valves. However, 40 valves were classified as "difficult to operate" due mainly to long stroke times or difficulty in turning the valve operator. To date, the applicant has modified only one valve, 1AF-041 (see paragraph 1), reducing the gear ratio and the time to operate from 30 minutes to 2 minutes. The applicant intends to modify the other two AFW pump discharge isolation valves (1AF-054 and 1AF-066) during the first refueling outage and will schedule other valve modifications on a case-by-case basis. A list of difficult-to-operate valves has been included in Procedure OWI-206, "Guidelines for Operation of Manual and Power Operated Valves," to alert operators and control room personnel to the schedule and manpower requirements associated with these valves.

The NRC inspector reviewed data sheets from the plant walkdown, the revisions made to Procedure OWI-206, and a summary of the applicant's actions on this issue documented in memorandum CPSES-90001405. The NRC inspector concluded that the applicant has taken adequate action to address this issue. This open item is closed.

l. (Closed) Open Item (445/8973-O-12): This item addressed the applicant's actions to make valves 1AF-041 and 1AF-054 easier to operate. During NRC investigation of the April 23 and May 5, 1989, AFW events involving the failure of several Borg-Warner check valves, the NRC determined that the difficulty of operation of these two valves was a contributing cause.

For valve 1AF-041, the applicant issued DCA 91717, Revision 1, to modify the existing 24:1 ratio manual gear operator to a 6:1 ratio operator. This reduces the number of turns required to open the valve from approximately 404 turns to 89 turns. The valve rim-pull is still within the specification limit of 40 ft/lbs. This work was completed via Work Order C890015384. Design Modification (DM) 89-403 requires the reduction of the operator gear ratio for valve 1AF-054 (as well as valve 1AF-066). Valve 1AF-054 currently has a gear ratio of 18:1 and the difficulty of operation is not as great as that for valve 1AF-041. In addition, the applicant has developed an operator aid which contains information for operations personnel on the difficulty and length of time required to operate each valve (see the closure of 445/8973-O-11, NRC Inspection Report 50-445/89-88; 50-445/89-88). Based on the above applicant actions, this item is closed.

m. (Closed) Open Item (445/8973-O-13): This item addressed the applicant's review of check valve min/max axial gap (play) criteria developed by Borg-Warner in response to check valve failures associated with the AFW backflow events discussed in NRC (AIT) Inspection Report 50-445/89-30; 50-446/89-30. Early in the investigation of the check valve failures, axial gap was thought to have been a significant contributor to the failure mechanism. Later research established valve bonnet height as the primary cause with axial gap as a less important, secondary factor.

The applicant has completed review of Borg-Warner's axial gap criteria and has incorporated these values (with some conservative changes) into Procedure MSM-CO-8801, "Borg-Warner Check Valve Maintenance," Revision 2. Some of the Borg-Warner check valves currently installed have axial gaps outside the envelopes specified in Procedure MSM-CO-8801. Each of these valves have individual calculations verifying that the axial gap will not affect operability of the valve. Nonconformance Report (NCR) 89-7476 documents the axial gap range of the currently installed valves and functions (along with the calculations) as a use-as-is disposition where gap length does not conform to Procedure MSM-CO-8801. The applicant stated that any future modifications to the check valves would likely involve complete replacement of the bonnet-swing arm assembly at which time the axial gap criteria of Procedure MSM-CO-8801 would be fully incorporated.

The NRC inspector determined that the applicant has established adequate control of the axial gap dimension and that the operability of check valves with axial gaps outside the procedural envelope is adequately assured by both calculation and functional backflow tests. This open item is closed.

n. (Closed) Open Item (445/8973-O-14): Training to increase operator awareness. As previously documented in NRC Inspection Report 50-445/89-30; 50-446/89-30, this item was identified during the NRC AIT evaluation of multiple check valve failures in the AFW system experienced during hot functional testing. In particular, the AFW backleakage events reflected negatively on the quality of training received by the plant operations staff. The necessity of performing in-sequence valve operation was apparently not adequately emphasized. A second training-related concern was identified in that the failure of operations personnel to document the discovery of three failed AFW check valves on a Plant Incident Report (PIR) or on an NCR.

In response to these issues the applicant committed to enhancing the awareness of plant operations personnel to operability issues by conducting additional training in this area. This additional training encompassed the following elements: (1) an operations management and senior reactor operator workshop, (2) auxiliary operator requalifying course ("Plant Incident Reports"), and (3) auxiliary operator requalifying course ("Recent Plant Incidents").

The NRC inspector reviewed course outlines, lesson plans, and attendance verification records for the three training sessions referenced above and concluded that the applicant's retraining effort has fully addressed the personnel issues associated with the AFW backflow events. This open item is closed.

o. (Closed) Open Item (445/8973-O-15): This item addressed service life degradation of the AFW minimum flow recirculation check valves (1AF-045, -057, and -069) due to turbulent flow conditions resulting from proximity to breakdown flow orifices. This issue was raised during the

AIT inspection (NRC Inspection Report 50-445/89-30; 50-446/89-30) in association with NRC review of the applicant's action to address the failure of valve 1AF-069 which occurred on April 5, 1989. The failure of this valve was probably the result of bonnet height elevation discrepancies through flow turbulence downstream of the orifice causing the valve disk to slam repeatedly against the stop may have been a contributing cause.

At the time of the AFW backflow events, the applicant's consultant, Kalsi Engineering, Inc., was performing a comprehensive review of safety-related check valves in response to Significant Operating Event Report (SOER) 86-03, "Check Valve Failure or Degradation." Kalsi's final report, "SOER 86-03 Check Valve Application Review," dated November 30, 1989, recommended (for valves 1AF-045, -057, and -069) the replacement of the existing 3/8" x 5/8" (step) disk studs with 5/8" (straight) disk studs to reduce the probability of disk stud fatigue failure. The applicant adopted this recommendation in design modification (DM)-89-316 and Design Change Notice (DCN)-000103. The disk studs were modified under work orders C890014336, C890014469, and C890014470 for valves 1AF-045, -057, and -069 respectively. All three valves subsequently passed backflow tests conducted in accordance with Procedure EGT-328A, "Reverse Flow Operability Testing for Auxiliary Feedwater Check Valves."

The NRC inspector reviewed all of the documentation referenced above and concluded that, <u>for</u> at least the short term, the disk stud modification was a viable alternative to increasing the distance between the orifice and the check valve. The applicant plans to inspect the condition of the AFW minimum flow recirculation check valves during the first refueling outage and plans at some future date to relocate the check valves. This open item is closed.

p. (Closed) Open Item (445/8973-O-09): During a previous NRC inspection of the backflow events in the AFW system piping, the NRC had concerns relating to high stresses in an elbow west of support No. AF-1-096-023-S33R and in two instrumentation connections. These items of concern were in the pipe evaluated in Stress Problem 1-10C and determined to have been highly stressed during the events. During this inspection period, the inspector reviewed the analyses for Stress Problem 1-10C documented in Attachment 9 to Calculation 15454-NP(S)-GENX 343. This attachment documents the results of thermal expansion stress evaluations in accordance with ASME Code Section III, Class 2 and 3 criteria (except that ASME Code stress allowables were not used). The evaluations showed that: (1) the highest stresses due to thermal expansion effects (97.41 ksi), and to the combined effects of pressure, weight, and thermal expansion (104.0 ksi) during the events were attained in the subject elbow; and (2) the highest stress in the piping in the vicinity of the subject two instrumentation connections due to thermal expansion only was 47.0 ksi, and to the combination of sustained loads and thermal expansion was 52.9 ksi which exceeded the ASME Code allowable stresses. In addition, stresses in several other locations in the piping due to thermal expansion only and the combination of sustained loads and thermal expansion exceeded the ASME Code allowable stresses. Subsequently, the second event was reevaluated to account for as-built gaps in four supports in the vicinity of the piping adjacent to the subject instrumentation connections. The reevaluation demonstrated that the highest stress in this piping due to thermal expansion only was reduced to 8.0 ksi and to the combination of sustained loads and thermal expansion to 13.8 ksi which were less than the ASME Code stress allowables. Stresses in the piping, including the highly stressed subject elbow, remote from the supports where as-built gaps were included in the analysis, were unaffected in this reevaluation.

Subsequently, TU Electric performed radiographic and ultrasonic inspections of areas in the piping, including the piping in the vicinity of the subject instrumentation connections and elbow, and verified that no damage had been incurred during the events and the ASME Code minimum wall-thickness requirement not violated during the events.

Based on the preceding inspection results, the inspector found that the TU Electric evaluations and inspections described in the preceding were sufficient to resolve the previous NRC concerns. Although the ASME Code allowable stresses were exceeded during the events, measures are being instituted by TU Electric to prevent reoccurrence of backflow in the AFW piping system thereby limiting future stresses in the piping system to no more than in their design. Consequently, given that the number of load cycles during which some areas of the AFW piping systems have been exposed to the high stresses experienced during the events are few (no more than two) and no damage was found in these areas, the NRC inspector determined that the AFW piping system is adequate to serve its intended function during plant life. This open item is closed.

q. (Closed) Open Item (445/8975-O-01): As part of the evaluation of the impact on the integrity of the affected piping system, pipe supports, containment penetrations, and instrumentation due to the events of April 23 and May 5, 1989, events involving backflow through the AFW system, Attachment K

CEUR

TUELECTRIC

Log # TXX-90188
File # 903.9
910.4

May 18, 1990

W. J. Cahill
Executive Vice President

RECEIVED
MAY 22 1990

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
FOLLOW-UP TO NOTICED MEETING OF MAY 9, 1990

REF: TU Electric letter from W. J. Cahill, Jr. to the U.S. NRC dated April 27, 1990 (TXX-90172)

Gentlemen:

Reference 1 provided information requested by the NRC Staff concerning overheating of Auxiliary Feedwater (AFW) System discharge lines. The letter described the condition, its cause, and the venting methodology used to return the line temperatures to normal. Additionally, TU Electric stated that the details of and schedule for any proposed long term actions would be provided in a subsequent letter.

A Management Meeting was held on May 9, 1990, to discuss these conditions. During the meeting, the NRC Staff requested that TU Electric provide additional bases for continued operation with four conditions that were identified between April 24, and May 1, 1990. These conditions were: 1) overheating of AFW piping; 2) seat leakage across Feedwater Preheater Bypass Valves (FPBV); 3) sticking Feedwater Isolation Valves (FWIV); and 4) a decrease in FWIV body temperature below the specified 90°F setpoint with the valve pressurized.

This letter provides the details of and schedule for proposed long term corrective actions. In addition, the letter describes the bases for continued operations with the above described conditions.

<u>AFW Piping Overheating</u>

On April 24 and 25, 1990, AFW System piping reached a temperature of 165°F (25°F in excess of the specified design temperature of 140°F). This condition occurred in part because of backleakage across the seat of BW/IP International Inc. 4" pressure seal check valves which serve to isolate the AFW System from the Main Feedwater System (MFW). The leakage was identified during the transition from AFW to MFW at low power levels (less than 10%). A small amount of preheated feedwater was flowing through the open Feedwater Preheater Bypass Valves (FPBV) back through leaking AFW check valves.

~~9005250048~~ II D

30-826

Because upstream valves were not leaking, pressure equalized across the AFW check valves. This allowed the valve disc to open slightly, permitting backflow. Because of small pressure differentials between the MFW lines (~ 4 psid), a recirculation path was established through the AFW System discharge lines back to the MFW lines, allowing AFW discharge line temperatures to approach MFW temperature.

On April 30, 1990, AFW line temperatures increased to 235°F. The backflow path during this event was similar to that described above, however, in this case the FPBV's were closed. This event is described further in a subsequent section of this letter.

Immediate corrective actions for each of the events described above included forward flushing with AFW water to cool the lines and assist in seating the check valves, and manual venting upstream of the check valves to create and maintain a higher differential pressure across the valves, thereby assuring tighter seating. Additionally, the applicable operations procedure was changed to reflect manual venting. It is anticipated that the need for venting, which is presently used during the AFW/MFW transition during plant startup and shutdown, will be minimized after the check valve modification discussed below, is made.

Each of these conditions was immediately evaluated by a multi-disciplined task team and Operations management. Testing to quantify the leakage rates across the subject check valves indicated the valves had not hung open. Therefore, the check valves were capable of carrying out their primary safety function of stopping backflow in the event of an upstream pipe break. At no time were the AFW pumps in danger of becoming steam bound.

Engineering evaluated the effect of elevated temperatures on the AFW piping system and the impact of the elevated piping temperatures on the accident analyses. Based on this evaluation the maximum allowed temperature was increased from 140°F to 210°F. This evaluation applies to the piping from the AFW pumps discharge check valves to the MFW piping.

In addition, Engineering evaluated the effects on piping and supports and accident analyses for temperature excursions above 210°F, should they occur. This evaluation concluded that for reactor power levels less than 30% of rated thermal power, temperature excursions of up to 250°F for durations of less than 24 hours are acceptable.

Based on these evaluations and immediate corrective actions, it was determined that the operability of the AFW System was not affected by the backleakage and high temperatures.

Engineering has determined that moving the clevis slightly on the affected AFW check valves (8) will improve disc/seat surface contact. The internals of eight BW/IP check valves from Unit 2 will be so modified for installation into Unit 1. Prior to installation each set of internals will be bench tested to achieve maximum seat tightness. Seating surfaces will be lapped and blue checked as necessary. All modified valves will be leak tested after installation to assure positive seating. Modification and rework will be completed during the next cold shutdown period of sufficient duration.

In addition to the above actions, TU Electric is planning to order check valves of different design for this AFW application to cover the contingency that replacement of the present valves becomes appropriate. Any replacement of the check valves will take into account the lessons learned on the currently installed check valves.

Feedwater Preheater Bypass Valves Leakage

On April 28, 1990, with reactor power at approximately 20%, operators noted that AFW line temperatures were increasing with the FPBVs closed. It was suspected that leakage past these valves in series with minor AFW check valve leakage was enough to establish the recirculation path discussed above. Reactor power was subsequently reduced due to an unrelated event. Operations personnel initiated a procedure change which requires isolation of the FPBVs by closing an upstream manual valve when turbine load exceeds 30%. On April 30, 1990, following the shutdown of the Number 2 AFW motor driven pump, which was run to attempt to reduce the leakage on one of the leaking AFW check valves, one of the AFW line temperatures increased to 235°F with the FPBVs closed but not isolated. The operators isolated the FPBV within twenty-five minutes and restarted the AFW pump to reduce temperature.

As stated above, corrective action for this condition was to change the operational procedure to require isolation of the four FPBVs with upstream manual valves when turbine load exceeds 30%. This load was selected to allow for an orderly transition above the feed system water hammer interlocks and to transition to the Feedwater Control Valves. This action also stops the temperature increases in the AFW System and precludes the need for manual venting. TU Electric will overhaul these valves during the next cold shutdown period of sufficient duration.

As previously stated, the high temperatures in the AFW lines caused by leakage through the check valves and FPBVs were evaluated and found to be acceptable.

The safety function of the FPBV is to close on a feedwater isolation signal to preclude excessive mass and energy release to containment during a feedwater or steamline break. The assumptions in the analyses of these accidents were reviewed and found to remain bounding. For these analyses, the assumptions were selected to maximize the main feedwater and auxiliary feedwater flow delivered to the faulted steam generator. In addition, for these accidents, the function of feedwater isolation is accomplished by the redundant closure of the FWIVs and the main feedwater control valves upon receipt of a feedwater isolation signal and the trip of the main feedwater pumps on a low steamline pressure signal, thereby eliminating any adverse affects due to leaking FPBVs during a main feedline break or main steamline accident inside containment.

Sticking Feedwater Isolation Valves

On April 27, 1990, Operations personnel, as part of the normal startup sequence, attempted unsuccessfully to open the four Feedwater Isolation Valves using normal methods. After discussions internally, with other nuclear sites, and with the vendor, it was suspected that the valves may be binding because of differential thermal expansion.

This condition did not adversely affect the safe operation of the plant because the safety position of the valves is closed. The valves are required to be shut to isolate containment, to close to minimize mass and energy release inside containment and to minimize RCS cooldown during a feedwater line break event and to close on low feedwater temperature as part of steam generator water hammer prevention. In no case have the valves failed to close upon demand.

Based on preliminary evaluation and discussions with the vendor, a hydraulic lifting device was used to assist the operator in lifting the valve discs off of their seats. Further engineering analysis and vendor information confirmed that external hydraulic assistance will not overstress internal or external parts of the valves. This method has been proceduralized and will be used until Engineering personnel can determine the specific cause for the valves failing to open using the normal methods. Cause identification and implementation of corrective actions will be completed prior to the end of the first refueling outage.

Feedwater Isolation Valves, Reduced Materials Temperature

On April 28, 1990, following a turbine generator shutdown due to a steam leak, the temperature of one FWIV decreased to 88°F at a system pressure of approximately 1200 psig. The Technical Requirements Manual (TRM) requires that each FWIV be at 90°F or greater in Modes 1, 2, and 3. At the time of the temperature decrease, the plant was in Mode 1.

Immediately after the condition was identified the heat trace was energized to increase valve temperature. Temperature was within specification within four minutes after discovery. This action placed the valves in compliance with the TRM requirements while the engineering evaluation required as a TRM Compensatory Measure was initiated.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

The 90°F minimum temperature was based on meeting specific ASME Code acceptance criteria for impact testing. The structural integrity issue addressed in the TRM is related to the material's fracture toughness as measured by additional testing performed in conjunction with the impact testing and reported in Engineering Report ER-DBE-ME-045. Fracture toughness testing conducted at 40°F demonstrated the high resistance of this material to crack propagation under slow to moderate strain rate conditions such as occurred during the slow decline in feedwater and FWIV temperature at relatively constant pressure on April 28.

The primary question considered in the Engineering Evaluation concerned the possible propagation of any pre-existing flaws in the valve. Based on the highly tough nature of this material, demonstrated at substantially lower temperatures, structurally significant flaw propagation under the described conditions would not have occurred. The valves were therefore determined to be acceptable for continued operations.

Additional actions taken following this event included a procedure change to the operations surveillance logs requiring additional temperature monitoring in Mode 1 any time the FWIVs are closed. The plant shutdown procedure has been changed to place the FWIV heat tracing in service during plant shutdown. A revision to the system operating procedure will require the FWIV heat tracing breakers to remain closed at all times, and integrated plant procedures will have steps to verify the breakers are correctly aligned during startup and shutdown.

TU Electric intends to change the TRM to clarify action requirements for the FWIVs when the valve is pressurized and at reduced temperature conditions.

TU Electric management will ensure that members of your onsite staff are kept informed of the actions described above and the results of those actions. Please contact me if further details are needed.

Sincerely,

William J. Cahill

William J. Cahill, Jr.

TLH/d[illegible]

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

Attachment L

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

OCT 30 1989

In Reply Refer To:
Dockets: 50-445/89-73
50-446/89-73

Mr. W. J. Cahill, Jr.
Executive Vice President
TU Electric
400 North Olive Street, Lock Box 81
Dallas, Texas 75201

Dear Mr. Cahill:

This refers to the inspection conducted by Mr. R. Latta and NRC consultants during the period September 6 through October 3, 1989, of activities authorized by NRC Construction Permits CPPR-126 and CPPR-127 for the Comanche Peak Steam Electric Station, Units 1 and 2, and to the discussion of our findings with you and other members of your staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of selective examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

During this inspection, it was found that certain of your activities were in violation of NRC requirements, as specified in the enclosed Notice of Violation. A written response to these violations is required.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

~~8911090059~~

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

RF Warnick

R. F. Warnick, Assistant Director
for Inspection Programs
Comanche Peak Project Division
Office of Nuclear Reactor Regulation

Enclosures:
Appendix A - Notice of Violation
Appendix B - Inspection Report 50-445/89-73; 50-446/89-73

cc w/enclosures:
See next page

cc w/enclosure:
Roger D. Walker
Manager, Nuclear Licensing
TU Electric
Skyway Tower
400 North Olive Street, L.B. 81
Dallas, TX 75201

Juanita Ellis
President - CASE
1426 South Polk Street
Dallas, TX 75224

Texas Radiation Control
Program Director
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756

GDS Associates, Inc.
1850 Parkway Place, Suite 720
Marietta, GA 30067-8237

Honorable George Crump
County Judge
Glen Rose, Texas 76043

Ms. Billie Pirner Garde, Esq.
Robinson, Robinson, et al.
103 East College Avenue
Appleton, WI 54911

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

William A. Burchette, Esq.
Counsel for Tex-La Electric
Cooperative of Texas
Heron, Burchette, Ruckert & Rothwell
1025 Thomas Jefferson St., NW
Washington, DC 20007

TU Electric
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, Maryland 20814

E. F. Ottney
P. O. Box 1777
Glen Rose, Texas 76043

Jack R. Newman
Newman & Holtzinger
1615 L Street, NW
Suite 1000
Washington, DC 20036

George R. Bynog
Program Mgr./Chief Inspector
Texas Dept. of Labor & Standards
Boiler Division
P.O. Box 12157, Capitol Station
Austin, Texas 78711

APPENDIX A

NOTICE OF VIOLATION

TU Electric

Docket: 50-445/89-73

Comanche Peak Steam Electric Station
Unit 1, Glen Rose, Texas

Permit: CPPR-126

During an NRC inspection conducted on September 6 through October 3, 1989, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1989), the violations are listed below:

A. Criterion V of Appendix B to 10 CFR Part 50, as implemented by Section 5.0, Revision 1, of the TU Electric Quality Assurance Manual, requires that activities affecting quality shall be prescribed by and accomplished in accordance with documented instructions, procedures, or drawings.

   Paragraph 15.1 of TU Electric Specification 2323-MS-85 states, in part, "Welding and brazing procedures, welders, and welding operations shall be qualified in accordance with AWS D.1.1, Structural Welding Code," which requires shielded metal arc welding processes for joints classified as "structural steel" square groove butt welds.

   Contrary to the above:

   The square grove butt welds on the companion angle flanges of the heating, ventilation, and air-conditioning (HVAC) system which were required to be welded using the shielded metal arc welding process were determined to have been welded using the gas metal arc welding process.

   This is a Severity Level IV violation (Supplement II) (445/8973-V-01).

B. Criterion XVII of Appendix B to 10 CFR Part 50, as implemented by Section 17.0, Revision 1, of the TU Electric Quality Assurance Manual, requires that measures shall be established to assure that sufficient records to furnish evidence of the quality of items and of activities affecting quality are maintained.

8911090062

Paragraph 6.3.3 of TU Electric Procedure CHV-101 states, in part, "complete the applicable portions of the welding checklist in accordance with Figure 7.1, HVAC Welding Checklist Entry Instructions."

Contrary to the above:

The weld records for the companion angle flanges of the HVAC system which were required to provide evidence of activities affecting quality were determined to be inaccurate in that welders signed for shielded metal arc welds (SMAW) which they had not performed, as indicated by discrepancies in the applicant's welding checklist continuation sheets.

This is a Severity Level IV violation (Supplement II) (445/8973-V-02).

In responding to this violation, the applicant is requested to address the certification implications of welders utilizing the shielded metal arc welding (SMAW) process in that the inaccuracies of the applicant's weld records may have resulted in safety-related welds which utilize this process being performed by uncertified welders.

C. Criterion XVI of Appendix B to 10 CFR 50, as implemented by Section 16, Revision 1, of the TU Electric Quality Assurance Manual states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected . . . ."

Contrary to the above:

The applicant failed to take prompt corrective action in response to the identification of conditions adverse to quality subsequent to the determination that procedural noncompliances had occurred during the fabrication of HVAC duct flanges which were identified by TU Electric Corporate Security on July 18, 1989, but which were not acted upon expeditiously by TU Electric management until this issue was identified at the NRC exit on October 3, 1989.

This is a Severity Level IV violation (Supplement II) (50-445/8973-V-03).

Pursuant to the provisions of 10 CFR 2.201, TU Electric is hereby required to submit a written statement or explanation to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555, with a copy to the Assistant Director for Inspection Programs, Comanche Peak Project Division, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a

"Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation if admitted, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved.

If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

FOR THE NUCLEAR REGULATORY COMMISSION

RF Warnick

Dated at Comanche Peak Site
this 30th day of October 1989

Appendix B

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

NRC Inspection Report: 50-445/89-73
50-446/89-73

Permits: CPPR-126
CPPR-127

Dockets: 50-445
50-446

Construction Permit
Expiration Dates:
Unit 1: August 1, 1991
Unit 2: August 1, 1992

Applicant: TU Electric
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Facility Name: Comanche Peak Steam Electric Station (CPSES), Units 1 & 2

Inspection At: Comanche Peak Site, Glen Rose, Texas

Inspection Conducted: September 6 through October 3, 1989

Inspector: R. M. Latta 10/27/89
R. M. Latta, Resident Inspector Date
(paragraphs 2, 3, 4, 5, 6, 7, 8, 9, and 10)

Consultant: J. Dale - EG&G (paragraph 5)
W. D. Richins - Parameter (paragraph 7)
J. L. Taylor - Parameter (paragraphs 4, 6, and 8)

Reviewed by: H. H. Livermore 10/30/89
H. H. Livermore, Lead Senior Inspector Date

8911090072

Inspection Summary:

Inspection Conducted: September 6 through October 3, 1989 (Report 50-445/89-73; 50-446/89-73)

Areas Inspected: Unannounced, resident safety inspection of the applicant's actions on previous inspection findings; follow-up on violations/deviations; action on 10 CFR 50.55(e) deficiencies identified by the applicant; allegation follow-up; electrical components and systems; safety-related mechanical components; and general plant tours.

Results: Within the areas inspected no significant strengths or weaknesses were identified. One open item was identified regarding the failure of check valves during reverse flow operability testing. Eleven additional open items resulting from the NRC Augmented Inspection Team (AIT) evaluation of multiple check valve failures experienced during hot functional testing (HFT) are also identified in this report (paragraph 7). During this inspection period, three violations were identified concerning welding deficiencies and the applicant's failure to take prompt corrective action associated with HVAC system welding allegations (paragraph 5.b).

DETAILS

1. Per s Contacted

*J. L. Barker, Manager, ISEG, TU Electric
*D. P. Barry, Senior, Manager, Engineering, Stone and Webster Engineering Corporation (SWEC)
*O. Bhatty, Issue Interface Coordinator, TU Electric
*M. R. Blevins, Manager of Nuclear Operations Support, TU Electric
*R. C. Byrd, Manager, Quality Control (QC), TU Electric
*W. J. Cahill, Executive Vice President, Nuclear, TU Electric
*H. M. Carmichael, Senior Quality Assurance (QA) Program Manager, CECO
*J. T. Conly, APE-Licensing, SWEC
*W. G. Counsil, Vice Chairman, Nuclear, TU Electric
*B. S. Dacko, Licensing Engineer, TU Electric
*R. J. Daly, Manager, Startup, TU Electric
*G. G. Davis, Nuclear Operations Inspection Report Item Coordinator, TU Electric
*S. L. Ellis, Performance and Testing, TU Electric
*J. C. Finneran, Jr., Manager, Civil Engineering, TU Electric
*J. L. French, Independent Advisory Group
*W. G. Guldemond, Manager of Site Licensing, TU Electric
*T. L. Heatherly, Licensing Compliance Engineer, TU Electric
*J. C. Hicks, Licensing Compliance Manager, TU Electric
*A. Husain, Director, Reactor Engineering, TU Electric
*J. J. Kelley, Plant Manager, TU Electric
*J. E. Krechting, Director of Technical Interface, TU Electric
*O. W. Lowe, Director of Engineering, TU Electric
*D. M. McAfee, Manager, QA, TU Electric
*S. G. McBee, NRC Interface, TU Electric
*J. W. Muffett, Manager of Project Engineering, TU Electric
*E. F. Ottney, Program Manager, CASE
*S. S. Palmer, Project Manager, TU Electric
*P. Raysircar, Deputy Director/Senior Engineer Manager, CECO
*M. J. Riggs, Plant Evaluation Manager, Operations, TU Electric
*J. C. Smith, Plant Operations Staff, TU Electric
*R. L. Spence, TU/QA Senior Advisor, TU Electric
*P. B. Stevens, Manager of Operations Support, TU Electric
*J. F. Streeter, Director, QA, TU Electric
*C. L. Terry, Unit 1 Project Manager, TU Electric
*O. L. Thero, QTC Consultant to CASE
*R. G. Withrow, EA Manager, TU Electric

The NRC inspectors also interviewed other applicant employees during this inspection period.

*Denotes personnel present at the October 3, 1989, exit meeting.

2. Applicant's Action on Previous Inspection Findings (92701)

(Closed) Open Item (445/8632-O-01): Heat tracing on containment atmosphere sample line. This item was opened to track the inspection and rework of the electrical heat tracing on a 1-inch containment atmosphere sample line. The damaged heat tracing was determined to be loose and the covering tape had been pulled back to reveal adhesive material remaining on the pipe in Room 88 of the Unit 1 Safeguards building. The NRC inspector reviewed the associated closeout documentation which included: DMRC 87-1-049, Design Change Authorization (DCA) 61617, and several travelers including JB-1HT0313-153-T1. Based on these reviews and inspection of the repaired heat tracing in Room 88, the NRC inspector determined that the subject heat tracing had been replaced, that the installation appeared complete, and that the sample lines had been properly insulated. Therefore, this open item is closed.

3. Follow-up on Violations/Deviations (92702)

(Closed) Violation EA 86-09, Appendix A, Item 1.C.1: QC inspectors failed to witness butt splices of control and instrumentation connections. In particular, this violation involved the failure of specific QC inspectors to perform required observations specified in the controlling Procedure QI-QP-11.3-28. As documented in TU Electric's revised response to this violation contained in TXX-88792 dated November 30, 1988, the applicant's corrective actions included reinspections to ensure that all butt splices have been properly identified on the appropriate design drawings. The scope and methodology utilized by the applicant to verify that all splices were properly inspected and to insure that similar conditions did not reoccur were delineated in Issue-Specific Action Plans (ISAP) I.a.2 and VII.C respectively.

The NRC inspector reviewed the results of these ISAPs as well as Corrective Action Request (CAR) 50 which was issued to address the questionable performance of four QC inspectors. The NRC inspector also reviewed the completed training records for personnel working in accordance with Procedure QI-QP-11.3-28, Revision 24, "Class 1E Cable Terminations," as well as Procedure CP-QP-2.1, Revision 18, "Training of Inspection Personnel," and its current (replacement) Procedure NQA 1.16, "Introduction and Training of Quality Assurance Personnel." Based on these reviews and evaluations, the NRC inspector determined that the applicant's corrective actions which included retraining of all electrical QC personnel appeared adequate to prevent reoccurrence of this violation. This item is closed.

- Extract from IR 50-445/89-84, 50-446/89-84

Attachment

M

APPENDIX B

1485

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

NRC Inspection Report: 50-445/89-84
50-446/89-84

Permits: CPPR-126
CPPR-127

Dockets: 50-445
50-446

Construction Permit
Expiration Dates:
Unit 1: August 1, 1991
Unit 2: August 1, 1992

Applicant: TU Electric
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Facility Name: Comanche Peak Steam Electric Station (CPSES),
Units 1 & 2

Inspection At: Comanche Peak Site, Glen Rose, Texas

Inspection Conducted: November 8 through December 5, 1989

Inspector: R. M. Latta 21 DEC 89
R. M. Latta, Resident Inspector Date
(Electrical) (paragraphs 2, 3, 4, 5, 6, 7, 8, 9
and 10)

Consultants: J. L. Birmingham, RTS (paragraph 3)
W. D. Richins, Parameter (paragraph 7)
J. L. Taylor, Parameter (paragraphs 3, 4, 6 and 8)

Reviewed by: RF Warnick for 12/21/89
H. H. Livermore, Lead Senior Inspector Date

9001020302

c. (Closed) Open Item (445/8908-O-04): This open item identified concerns relative to the adequacy of temperature control verification by Quality Control (QC) during the welding process on the AFW rotor bar assembly. Specifically, the controlling maintenance instruction stated that <u>extreme caution</u> must be taken not to concentrate an excessive amount of heat on the rotor bar assembly during welding. The NRC inspector identified a concern that QC had not verified that this instruction was adhered to.

   Subsequently, TU Electric personnel met with the NRC on two separate occasions to provide additional information about this concern. During the second meeting, TU Electric indicated that an electrical engineer had inserted the caution about heat input. Additionally, a welding specialist identified the material as a low carbon steel and provided information regarding the energy input. Based on the supplemental information provided by the applicant, the identified concerns were adequately addressed. Therefore, this item is closed.

d. (Closed) Open Item (445/8973-O-05): Documentation for the failure of check valve 1MS-142 in 1985. This item was identified during the NRC Augmented Inspection Team evaluation of multiple check valve failures experienced during the April - May, 1989, hot functional testing. In particular, the applicant's Failure Analysis Report FA 85-001, Revision 0, had correctly identified the root cause of the failure of valve 1MS-142 as the valve bonnet and retainer ring being incorrectly placed too low in the valve body. Subsequent to contacting the supplier, Borg-Warner, the applicant revised the root cause stated in FA 85-001, replaced the valve disc and shortened the disc stud to reduce axial play. The applicant concluded that valve internals were correctly installed and that the root cause was actually unanticipated system transients as evidenced by failed system snubbers. The revised response was supported by analytical documentation regarding cold start system loads and vendor information pertaining to a similar incident at another nuclear facility. Apparently, no documentation of the applicant's discussion with Borg-Warner exists.

   The NRC inspector examined the applicant's supplementary documentation regarding the revised engineering decision. This documentation included a review of correspondence from maintenance engineering to licensing contained in TU Electric's memo TCF-891587 and TCF-891627 as well as Problem Report 85-297, Failure Analysis Report FA 85-005, and Test Deficiency Report CP-SAP-16. The NRC inspector concluded that as a result of not following up on the initially identified cause of this precursor event, the applicant failed to take adequate corrective action and similar valve failures due to improper bonnet retainer installation occurred in 1989. This issue is addressed by violation 445/8930-V-02, part B.1. Therefore, this open item is closed.

3. <u>Action on 10 CFR Part 50.55(e) Deficiencies Identified by the Applicant (92700)</u>

a. (Closed - Unit 1 only) Construction Deficiency (SDAR CP-87-21): "Effect of Thermolag on Derating Factors." This reportable deficiency involved the applicant's evaluations of thermolag derating factors which determined that the previously assumed value of 10% used on internal cable sizing calculations was nonconservative. Specifically, the derating factors of 31% for single trays and 20% for single conduits enclosed in thermolag were established.

As described in the applicant's interim report contained in TU Electric's letter TXX-7041, the failure to consider the increased derating of power cables due to thermolag installation could have caused the subject cables to exceed their design temperature rating resulting in the indeterminate status of associated Class 1E circuits. This condition reportedly was the result of evaluations performed by the vendor which altered the previously accepted cable derating factors.

The applicant's corrective actions included the identification of cables which would have exceeded the prescribed ampacity rating due to the thermolag and to either remove the thermolag from the raceways or increase the cable size. Additionally, the applicant revised the applicable Design Basis Document (DBD)-EE-052, "Cable Philosophy and Sizing Criteria," to establish the design considerations for cable ampacity derating.

The NRC inspector reviewed the results of the Consolidated Engineering Contractor Organization (CECO) response to this issue contained in CECO letter 1318 dated June 21, 1989. The actions documented in this letter included: the completion of design validation of all installed cables, the identification of cables which did not comply with DBD-EE-052, and a listing of the documents which implemented the corrective actions.

Based on the above inspection activities which included a review of a representative sample of the design change authorizations (DCAs) identified in the reference CECO correspondence, an examination of the controlling

Attachment #1

Log # TXX-90188
File # 903.9
910.4

TU

May 18, 1990

W. J. Cahill
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
FOLLOW-UP TO NOTICED MEETING OF MAY 9, 1990

REF: TU Electric letter from W. J. Cahill, Jr. to the U.S. NRC dated April 27, 1990 (TXX-90172)

Gentlemen:

Reference 1 provided information requested by the NRC Staff concerning overheating of Auxiliary Feedwater (AFW) System discharge lines. The letter described the condition, its cause, and the venting methodology used to return the line temperatures to normal. Additionally, TU Electric stated that the details of and schedule for any proposed long term actions would be provided in a subsequent letter.

A Management Meeting was held on May 9, 1990, to discuss these conditions. During the meeting, the NRC Staff requested that TU Electric provide additional bases for continued operation with four conditions that were identified between April 24, and May 1, 1990. These conditions were: 1) overheating of AFW piping; 2) seat leakage across Feedwater Preheater Bypass Valves (FPBV); 3) sticking Feedwater Isolation Valves (FWIV); and 4) a decrease in FWIV body temperature below the specified 90°F setpoint with the valve pressurized.

This letter provides the details of and schedule for proposed long term corrective actions. In addition, the letter describes the bases for continued operations with the above described conditions.

AFW Piping Overheating

On April 24 and 25, 1990, AFW System piping reached a temperature of 165°F (25°F in excess of the specified design temperature of 140°F). This condition occurred in part because of backleakage across the seat of BW/IP International Inc. 4" pressure seal check valves which serve to isolate the AFW System from the Main Feedwater System (MFW). The leakage was identified during the transition from AFW to MFW at low power levels (less than 10%). A small amount of preheated feedwater was flowing through the open Feedwater Preheater Bypass Valves (FPBV) back through leaking AFW check valves.

9005250048

D029
1/0

Because upstream valves were not leaking, pressure equalized across the AFW check valves. This allowed the valve disc to open slightly, permitting backflow. Because of small pressure differentials between the MFW lines (~ 4 psid), a recirculation path was established through the AFW System discharge lines back to the MFW lines, allowing AFW discharge line temperatures to approach MFW temperature.

On April 30, 1990, AFW line temperatures increased to 235°F. The backflow path during this event was similar to that described above, however, in this case the FPBV's were closed. This event is described further in a subsequent section of this letter.

Immediate corrective actions for each of the events described above included forward flushing with AFW water to cool the lines and assist in seating the check valves, and manual venting upstream of the check valves to create and maintain a higher differential pressure across the valves, thereby assuring tighter seating. Additionally, the applicable operations procedure was changed to reflect manual venting. It is anticipated that the need for venting, which is presently used during the AFW/MFW transition during plant startup and shutdown, will be minimized after the check valve modification discussed below, is made.

Each of these conditions was immediately evaluated by a multi-disciplined task team and Operations management. Testing to quantify the leakage rates across the subject check valves indicated the valves had not hung open. Therefore, the check valves were capable of carrying out their primary safety function of stopping backflow in the event of an upstream pipe break. At no time were the AFW pumps in danger of becoming steam bound.

Engineering evaluated the effect of elevated temperatures on the AFW piping system and the impact of the elevated piping temperatures on the accident analyses. Based on this evaluation the maximum allowed temperature was increased from 140°F to 210°F. This evaluation applies to the piping from the AFW pumps discharge check valves to the MFW piping.

In addition, Engineering evaluated the effects on piping and supports and accident analyses for temperature excursions above 210°F, should they occur. This evaluation concluded that for reactor power levels less than 30% of rated thermal power, temperature excursions of up to 250°F for durations of less than 24 hours are acceptable.

Based on these evaluations and immediate corrective actions, it was determined that the operability of the AFW System was not affected by the backleakage and high temperatures.

Engineering has determined that moving the clevis slightly on the affected AFW check valves (8) will improve disc/seat surface contact. The internals of eight BW/IP check valves from Unit 2 will be so modified for installation into Unit 1. Prior to installation each set of internals will be bench tested to achieve maximum seat tightness. Seating surfaces will be lapped and blue checked as necessary. All modified valves will be leak tested after installation to assure positive seating. Modification and rework will be completed during the next cold shutdown period of sufficient duration.

In addition to the above actions, TU Electric is planning to order check valves of different design for this AFW application to cover the contingency that replacement of the present valves becomes appropriate. Any replacement of the check valves will take into account the lessons learned on the currently installed check valves.

Feedwater Preheater Bypass Valves Leakage

On April 28, 1990, with reactor power at approximately 20%, operators noted that AFW line temperatures were increasing with the FPBVs closed. It was suspected that leakage past these valves in series with minor AFW check valve leakage was enough to establish the recirculation path discussed above. Reactor power was subsequently reduced due to an unrelated event. Operations personnel initiated a procedure change which requires isolation of the FPBVs by closing an upstream manual valve when turbine load exceeds 30%. On April 30, 1990, following the shutdown of the Number 2 AFW motor driven pump, which was run to attempt to reduce the leakage on one of the leaking AFW check valves, one of the AFW line temperatures increased to 235°F with the FPBVs closed but not isolated. The operators isolated the FPBV within twenty-five minutes and restarted the AFW pump to reduce temperature.

As stated above, corrective action for this condition was to change the operational procedure to require isolation of the four FPBVs with upstream manual valves when turbine load exceeds 30%. This load was selected to allow for an orderly transition above the feed system water hammer interlocks and to transition to the Feedwater Control Valves. This action also stops the temperature increases in the AFW System and precludes the need for manual venting. TU Electric will overhaul these valves during the next cold shutdown period of sufficient duration.

As previously stated, the high temperatures in the AFW lines caused by leakage through the check valves and FPBVs were evaluated and found to be acceptable.

The safety function of the FPBV is to close on a feedwater isolation signal to preclude excessive mass and energy release to containment during a feedwater or steamline break. The assumptions in the analyses of these accidents were reviewed and found to remain bounding. For these analyses, the assumptions were selected to maximize the main feedwater and auxiliary feedwater flow delivered to the faulted steam generator. In addition, for these accidents, the function of feedwater isolation is accomplished by the redundant closure of the FWIVs and the main feedwater control valves upon receipt of a feedwater isolation signal and the trip of the main feedwater pumps on a low steamline pressure signal, thereby eliminating any adverse affects due to leaking FPBVs during a main feedline break or main steamline accident inside containment.

Sticking Feedwater Isolation Valves

On April 27, 1990, Operations personnel, as part of the normal startup sequence, attempted unsuccessfully to open the four Feedwater Isolation Valves using normal methods. After discussions internally, with other nuclear sites, and with the vendor, it was suspected that the valves may be binding because of differential thermal expansion.

This condition did not adversely affect the safe operation of the plant because the safety position of the valves is closed. The valves are required to be shut to isolate containment, to close to minimize mass and energy release inside containment and to minimize RCS cooldown during a feedwater line break event and to close on low feedwater temperature as part of steam generator water hammer prevention. In no case have the valves failed to close upon demand.

Based on preliminary evaluation and discussions with the vendor, a hydraulic lifting device was used to assist the operator in lifting the valve discs off of their seats. Further engineering analysis and vendor information confirmed that external hydraulic assistance will not overstress internal or external parts of the valves. This method has been proceduralized and will be used until Engineering personnel can determine the specific cause for the valves failing to open using the normal methods. Cause identification and implementation of corrective actions will be completed prior to the end of the first refueling outage.

Feedwater Isolation Valves, Reduced Materials Temperature

On April 28, 1990, following a turbine generator shutdown due to a steam leak, the temperature of one FWIV decreased to 88°F at a system pressure of approximately 1200 psig. The Technical Requirements Manual (TRM) requires that each FWIV be at 90°F or greater in Modes 1, 2, and 3. At the time of the temperature decrease, the plant was in Mode 1.

Immediately after the condition was identified the heat trace was energized to increase valve temperature. Temperature was within specification within four minutes after discovery. This action placed the valves in compliance with the TRM requirements while the engineering evaluation required as a TRM Compensatory Measure was initiated.

The 90°F minimum temperature was based on meeting specific ASME Code acceptance criteria for impact testing. The structural integrity issue addressed in the TRM is related to the material's fracture toughness as measured by additional testing performed in conjunction with the impact testing and reported in Engineering Report ER-DBE-ME-045. Fracture toughness testing conducted at 40°F demonstrated the high resistance of this material to crack propagation under slow to moderate strain rate conditions such as occurred during the slow decline in feedwater and FWIV temperature at relatively constant pressure on April 28.

The primary question considered in the Engineering Evaluation concerned the possible propagation of any pre-existing flaws in the valve. Based on the highly tough nature of this material, demonstrated at substantially lower temperatures, structurally significant flaw propagation under the described conditions would not have occurred. The valves were therefore determined to be acceptable for continued operations.

Additional actions taken following this event included a procedure change to the operations surveillance logs requiring additional temperature monitoring in Mode 1 any time the FWIVs are closed. The plant shutdown procedure has been changed to place the FWIV heat tracing in service during plant shutdown. A revision to the system operating procedure will require the FWIV heat tracing breakers to remain closed at all times, and integrated plant procedures will have steps to verify the breakers are correctly aligned during startup and shutdown.

TU Electric intends to change the TRM to clarify action requirements for the FWIVs when the valve is pressurized and at reduced temperature conditions.

TU Electric management will ensure that members of your onsite staff are kept informed of the actions described above and the results of those actions. Please contact me if further details are needed.

Sincerely,

William J. Cahill

William J. Cahill, Jr

TLH/daj

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

Attachment

O

Log # TXX 90172
File # 10010
910.4
Ref. # 10CFR50.36

TU

April 27, 1990

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
REQUEST FOR INFORMATION REGARDING OPERATION OF THE
AUXILIARY FEEDWATER SYSTEM

Gentlemen:

On April 26 and 27, 1990, discussions were conducted with members of the NRC staff regarding a potential overtemperature condition in Auxiliary Feedwater (AFW) piping due to minor check valve leakage. It was identified that minor leakage through the AFW check valves from operation of main feedwater at low power levels resulted in excessive temperatures in the AFW piping on the upstream side of check valves. Continued minor leakage allows pressure equalization across these check valves, allowing them to unseat slightly and permit flow through the AFW lines from steam generator feedwater lines at a higher pressure to steam generator feedwater lines at a slightly lower pressure (~ 4psid). The slight pressure differential between feedlines is a result of the feedwater piping configuration.

During these discussions CPSES stated that it would vent the upstream side of check valves as necessary to seat the check valves tighter, allowing piping temperatures to stabilize at acceptable values. The controls implemented to perform this venting function have been reviewed by your onsite staff.

Subsequently, the NRC staff requested that TU Electric provide a letter committing to establish a schedule for any proposed long term actions for the above described condition and that TU Electric provide assurance that all other BW/IP check valves are capable of performing their intended safety function.

~~9005030234~~ D029 1/0

TU Electric will provide the details of and a schedule for any proposed long term actions and, if TU Electric elects to continue to use venting as the long term action, this decision will be discussed with the NRC staff. The Unit's transition from operational Modes 6 through 1, which required surveillance testing and rework with post work testing, has assured that all BW/IP check valves will perform their intended safety function.

Please contact me if further information is required.

Sincerely,

William J. Cahill, Jr.

William J. Cahill, Jr.

TLH/daj

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

4. Action on 10 CFR Part 50.55(e) Deficiencies Identified by the Applicant (92700)

a. (Closed) Construction Deficiency (SDAR CP-87-16): "Limit Switch Wiring." This deficiency, which was determined by the applicant to be reportable, involved the routing of cables/conduits to the wrong limit switches and the termination of cables to the wrong contacts on the limit switches. As a result of this issue, CAR-049 was initiated to disposition Unit 2 discrepancies. Field Verification Method (FVM)-089 as well as Startup Prerequisite Test Instruction XCP-EE-8 have been implemented to address Unit 1 deficiencies. FVM-089 has been reviewed and accepted as part of the Corrective Action Program (CAP) closeout as documented in previous NRC reports. Additional corrective actions, initiated by the applicant, included revising Design Basis Document (DBD)-EE-054 to incorporate terminal board identifications on controlled drawings and the development of a new drawing series, 2323-E1-0075, to provide specific limit switch identifications and orientations. The NRC inspector reviewed the above documentation as well as a sample of 5 out of approximately 55 DCAs/NCRs which had been issued to correct the subject deficiencies. Based on the above reviews and inspection activities, the NRC inspector determined that the applicant had taken adequate corrective measure for both Units 1 and 2. This construction deficiency is closed.

b. (Closed) Construction Deficiency (SDAR CP-87-85): "Degradation of Class 1E Circuits." This deficiency resulted from the direct connection of the Safety Systems Inoperable Indication (SSII) panel which is non-Class 1E, to Class 1E circuits. Additionally, the cables used to make these connections which had been routed with Class 1E cables did not have any supporting documentation which established their qualification to Class 1E standards. Subsequent justification for this condition was supported by an analysis which determined that the low energy instrumentation signals on the non-Class 1E cables would not have resulted in the degradation of Class 1E circuits. This analysis has been included in an advance FSAR change submitted to the NRC by letter TXX 89578 dated August 15, 1989. The NRC inspector reviewed the above documentation, as well as the National Electrical Code wiring ratings tables for the wiring sizes involved, and concluded that the justification for nonreportability was acceptable. This construction deficiency is closed.

c. (Closed) Construction Deficiency (SDAR CP-89-006): "6.9kv Breaker Charging Motor Linkage." This construction deficiency involved the applicant's reported loss of a connecting pin between the charging motor linkage and the breaker closing springs on several Brown Boveri breakers. The applicant determined that the deficiency was reportable and inspections of all affected breakers as well as revisions to maintenance procedures were initiated to address the deficiency and to prohibit the reuse of the connecting pin snap-ring retainers. The NRC inspector reviewed the associated nonconformance reports (NCRs) 89-01847, Revision 0, (Unit 1 breakers) and 89-02475, Revision 0, (Unit 2 breakers) and determined the disposition of the NCRs was acceptable. Additionally, the NRC inspector observed the applicant's inspections of several of the 1E breakers, as documented in NRC Inspection Report 50-445/89-64; 50-446/89-64. Based on the referenced nonconformance report (NCR) reviews and the inspections performed on these 6.9kv breakers, the NRC inspector determined that the applicant's corrective measures and maintenance practices including those proposed for Unit 2 were acceptable. This construction deficiency is closed for both Units 1 and 2.

d. (Closed - Unit 1 only) Construction Deficiency (SDAR CP-87-135): "Control Room Air Conditioning and Primary Plant Ventilation Systems." This issue involved inadequacies in the safety-related control circuits for the redundant trains associated with the control room HVAC system which were not designed to meet the single failure criteria. Specifically, as determined by the applicant, the control room HVAC system was susceptible to a single failure which could have prevented the automatic isolation of the system under accident conditions. Additionally, the auxiliary, safeguard, and fuel building ventilation supply fans were powered from a non-Class 1E power supply and were automatically tripped by nonsafety-related pressure switches. The significance of these inadequacies was that the capability to limit the radiation dose received by control room operators during postulated accident conditions to within FSAR limits was compromised and that the post-LOCA offsite dose could have been increased above the dose levels specified in the FSAR.

The NRC inspector reviewed the applicant's corrective actions stated in TU letter TXX 88013 dated January 29, 1989, which included the modification of the control room HVAC system to incorporate the single failure design criteria specified in the DBD EE-054, "Control Circuits Parameters/Loading Requirements" and IEEE-323. The NRC inspector also reviewed the applicant's design change specified in letter TXX-89356 dated July 14, 1989, which identified the inclusion of safety-related controls to automatically trip the primary plant ventilation system supply fans.

Based on these document reviews and selected system inspections of Unit 1 components, the NRC inspector concluded that the applicant's corrective actions appeared to be adequate. This construction deficiency is closed for Unit 1 only.

e. (Closed - Unit 1 only) Construction Deficiency (SDAR CP-88-08): "Battery Room Heaters." As previously reported in NRC Inspection Report 50-445/89-64; 50-446/89-64, this deficiency involved the replacement of battery room heaters with Class 1E seismically qualified units powered from redundant Class 1E power supplies. During this reporting period, the applicant subsequently provided the NRC inspector with a list of work packages which indicated the status for the completion of this work. A review of this work schedule indicated that approximately half of the listed packages were identified as being complete. The NRC inspector examined the installation of two of the battery room heaters and determined that the corrective construction activities for these battery rooms appeared to be complete. Based on a review of the completed installations and the work packages in place for Unit 1, the NRC inspector determined that the applicant's corrective actions and committed completion schedules prior to fuel load for Unit 1 is acceptable. Therefore, this construction deficiency is closed of Unit 1 only.

f. (Closed - Unit 1 only) Construction Deficiency (SDAR CP-89-16): "Turbine Driven Auxiliary Feedwater Pump Overspeed Trip." This deficiency involved a potentially reportable concern relative to the setpoint tolerance on the turbine driven auxiliary feedwater (TDAFW) pump mechanical overspeed trip device which could have resulted in the overpressurization of the AFW system including the pump casing. As documented in the applicant's final report contained in letter TXX-89494, the TDAFW pump was designed to trip if the turbine overspeed reached 125% of the turbine rated speed. During uncoupled overspeed testing of the auxiliary feedwater (AFW) pump turbine, the turbine tripped at a speed of 5147 revolutions per minute (RPM) which was three RPM over the maximum allowable trip speed (including setpoint tolerance) of 5144 RPM. Subsequent review indicated that the maximum allowable trip speed (including setpoint tolerance) was 44 RPM higher than the maximum speed utilized for the maximum system pressure calculation.

The inspector reviewed the supporting engineering calculation No. ME(B)022, Revision 4, and determined that the maximum reported RPM was marginally below the value used in the established system design pressures for the AFW during postulated accident conditions. Moreover: (1) no equipment damage was incurred during testing in that the turbine overspeed testing was performed with the turbine uncoupled and pressures experienced by the turbine during overspeed testing did not exceed the turbine design pressure, and (2) the turbine overspeed setting has been reduced from 25% to 16.6% over rated speed and will be tested in the future with the turbine uncoupled. Based on these reviews, the NRC inspector determined that the applicant's assessment of nonreportability was acceptable and that the supporting analysis was adequate. This deficiency is closed for Unit 1 only.

g. (Closed) Construction Deficiency (SDAR CP-89-018): "Soldering in Elgar Inverters." The applicant reported finding cracked, broken, or defective solder joints on: terminal block drive board connections, a transformer to drive board connection, and on the drive board connector pins in various Elgar inverters. As determined by the applicant, the defective joints were attributable to troubleshooting and maintenance activities related to Inspection and Enforcement (IE) Notice 88-57. This Notice involved information relative to the proper torquing of silicon controlled rectifiers (SCRs) to circuit boards/heat sink connections. The NRC inspector reviewed the applicant's corrective actions which include reinspection of all Elgar inverters by a factory representative, training of maintenance personnel, and issuance of appropriate NCRs and/or work orders. The Elgar trip report dated September 15, 1989 detailed the vendor's findings for both Units 1 and 2 inverters and the training which was provided to the applicant's personnel. Additionally, the NRC inspector reviewed several associated work orders and determined that the vendor recommendations had been implemented by the applicant. Based on the above reviews and inspections, the NRC inspector determined that the corrective actions were adequate and that the administrative programs in place should assure appropriate follow-up of the work items on Unit 2. This construction deficiency is closed.

h. (Closed - Unit 1 only) Construction Deficiency (SDAR CP-89-22): "Atmospheric Cleanup Heater Control Panels." This issue involved two of the primary plant ventilation system engineered safety features exhaust filtration unit control panels which were determined not to be seismically qualified. In particular, control panels CPX-VAFUPK-01P and -02P were not supplied by the manufacturer with the appropriate documentation to certify that these panels were seismically qualified. Additionally, as stated in the applicant's letter TXX-89673 dated September 13, 1989, CPSES calculations demonstrated that the specified requirement for an overall natural frequency of equal to or greater than 33 Hertz had not been met by these panels.

Subsequent to the applicant's determination that the subject panels were not qualified for their safety-related application, CAR 88-31 was initiated to identify and resolve concerns related to the failure of the manufacturer to properly comply with the purchase specification requirements. The applicant also initiated Field Requisition 6R374900 to purchase seismically qualified replacement heater control panels.

The NRC inspector examined the applicant's completed corrective actions associated with this issue including NCR 89-8130, Revision 0, P.O. 665-72045, and DCA 75000, Revision 4. Based on these reviews, the NRC inspector determined that the applicant's actions in replacing the subject heater control panels with seismically qualified components was acceptable and that this issue was adequately resolved for Unit 1. However, pending the implementation of corrective action, this item will remain open for Unit 2.

5. Allegation Follow-up (50100, 55100, 99014)

a. (Open) Allegation (OSP-88-A-0053): As previously documented in NRC Inspection Report 50-445/89-04; 50-446/89-04 this, allegation concerned installation practices utilized on Conax electrical penetrations. These penetrations contained Kapton insulated wiring in various conductor sizes. Specifically, the issues were that installation practices violated the specified minimum bend radius requirements during the arrangement of conductors in the cable trays and that inappropriate care was exercised in the installation process to protect the conductors from damage. Also, concern was expressed relative to the applicant's practice of bundling the conductors together and tie wrapping them to the lateral supports in the bottom of the cable trays in that plant induced vibration could then result in chaffing of the Kapton insulation. This chaffing could result in a direct short, thus affecting both control and instrumentation functions. Although these concerns were identified in Unit 2, they have generic implications for Unit 1 penetrations which also utilize Conax penetrations with Kapton insulated conductors.

The evaluations conducted by the NRC inspectors and documented in the referenced inspection report indicated that the applicant had adequately addressed the potential design concerns relative to the functional adequacy of installed Kapton insulated Class 1E equipment. Additionally, the applicant had identified all applications of Kapton insulation at CPSES and did not plan any further action in regard to redesign or replacement of Kapton. The allegation relative to the installation deficiencies remained open pending completion of detailed inspections to be performed by TU Electric in accordance with Electrical Specification ES-100 prior to the installation of cable tray covers in the penetration areas.

During the latter portion of this reporting period, the applicant initiated their cable inspection program which included both safety-related and nonsafety-related Kapton insulated penetration termination configurations. The NRC inspectors witnessed approximately 15 of these inspections and determined that the electrical craft personnel involved in the cleaning and preservation activities were sensitive to the special handling requirements associated with Kapton and that defects identified by craft personnel were brought to the attention of the inspecting organization. Additionally, it was observed that the inspecting personnel (QC for Class 1E applications and construction engineering (CE) for non-Class 1E applications) were familiar with the inspection requirements of electrical Specification ES-100 and that identified deficiencies were properly documented.

The NRC inspector also attended a scheduled training session conducted for the second shift craft personnel and construction/field engineers regarding inspection of Kapton wiring and the subsequent installation of cable tray covers. The training appeared to be very thorough and it emphasized: the adherence to Specification ES-100, the referral of all questionable Kapton configurations to engineering, the applicable inspection requirements, and the necessity for careful handling of the Kapton insulated conductors.

On September 28, 1989, while QC personnel were performing an inspection of a junction box associated with a containment penetration, several of the Kapton insulated conductors at the penetration were inadvertently grounded to a cross brace in the junction box. The incident resulted in the insulation breakdown of two of the conductors and the flash-over damage to approximately three adjacent conductors. Based on the available information, it could not be determined if the electrical grounding was caused by a previous defect in the Kapton covering or as a result of possible wear due to the rubbing of the wire on the structure brace. The conductors involved provided power to one of the redundant Unit 1 Train A, RHR pump suction isolation valves. The applicant is currently in the process of evaluating the implications of this event including the generic ramifications.

At the conclusion of the inspection period, the applicant's implementation of their penetration inspection and cable tray installation program was still in progress and no conclusions have been developed regarding its acceptability. Therefore, this allegation will remain open pending the applicant's completion of inspections of Kapton insulated penetration configurations for Unit 1.

b. (Open) Allegation (OSP-89-A-0061): This allegation involved a former worker's concerns relative to safety issues which were identified to members of the NRC resident staff at Comanche Peak on July 13, 1989. The initial concerns relating to plant electrical components and systems are addressed in NRC Inspection Report 50-445/89-64; 50-446/89-64. This report will address the remaining concerns which involve welding issues.

The first welding concern identified by the alleger was that during the welding process when the HVAC welding checklist continuation sheet required shielded metal arc welding (SMAW) using welding process No. CHV-501, the craft used the gas metal arc welding (GMAW) process in order to speed up the welding process. The alleger went on to state that up until January/February of 1989 the procedures had allowed the welder to GMAW all the joints on the duct flanges, but that subsequent changes in procedures (CSP-FD-HV-501, 502, and 504) mandated the use of E7018 (stick) SMAW. The alleger stated that this had been previously identified to SAFETEAM and Corporate Security. The alleger also expressed concerns that the SAFETEAM and Corporate Security would try to cover this up.

The NRC inspector reviewed the SAFETEAM and Corporate Security reports with the following results: SAFETEAM reviewed the concerns and found cause for an investigation by Corporate Security thereby relinquishing the concern to Corporate Security on or about June 15, 1989. Corporate Security interviewed the alleger for any further information he might have. They then interviewed four additional welders and one welding foreman on July 10, 1989, with the following results: the welders had knowledge of other personnel performing GMAW welds in lieu of the SMAW stick welds that were specified in the work package weld records. The welders claimed that a significant percentage of the welders currently do this. The welding foreman, however, claimed to have never witnessed this action and further indicated that it was strictly prohibited. Corporate Security's investigation concluded that evidence existed that suggested numerous procedural violations had occurred in the HVAC welding process and that a significant number of welds that were procedurally required to be SMAW stick welds were in fact gas metal arc (GMAW) welds in their response to SAFETEAM dated August 14, 1989.

The NRC inspector reviewed the applicant's welding Procedures FD-HV-501, -502, -504, and CSP-CHV-107 as well as DCA 75357, Revision 5, issued January 23, 1989, which resulted in the revision of Specification 2323-MS-85 to change welding of sheet metal to structural steel from AWS D9.1 to AWS D1.1 requirements. This DCA essentially changed the definition of ductwork such that subsequent to the DCA all angle iron reinforcement was classified as structural steel which required SMAW (AWS D1.1). The inspector also performed detailed walkdowns of HVAC ductwork in the Unit 1 safeguards building as well as the fuel building and conducted interviews with three welders currently involved with HVAC ductwork fabrication. Additionally, the NRC inspector reviewed selected samples of the welding surveillance check lists performed in accordance with process Procedure CSP-CHV-107 covering the period between March 23, 1989, and June 6, 1989. This review indicated that during this time frame their were approximately seven examples of welders performing GMAW welds on square grove butt welds, which is not allowed by Specification 2323-MS-85. These examples were identified by welding technicians and were documented on the welding surveillance checklists provided by the applicant.

Based on the review of the above stated DCA and the associated specification, welding procedures, and welder surveillance checklists, combined with the inspections of installed HVAC stiffeners and supports, the documentation reviews of Corporate Security files, and the examination of construction travelers and weld withdrawal slips, the NRC inspector concluded that this portion of the allegation was substantiated and that this condition does exist.

Given that the square grove butt welds on the HVAC companion angle flanges are characterized as seal welds which are not taken credit for in the applicant's structural/seismic analysis, the impact on the design and adequacy of the HVAC companion angle flanges is negligible. However, in that the applicant's program failed to control the application of the specified weld process at the subject weld joints and that there is a potential that this practice may have resulted in the misapplication of GMAW welds on other structural weld joints which specified SMAW, this example of failure to follow procedures by the applicant's welding personnel is identified as a violation (445/8973-V-01).

The alleger's second welding concern involved the welders use of rod withdrawal slips and weld records for recording the identifications of the welder making the weld. The alleger expressed his concern that some of the welders making the welds were not certified for the welding process being used, for example, SMAW versus GMAW, and that when this was the case a welder that was certified to the process being used would then sign for and claim as his the weld in question.

During the Corporate Security interview identified in the preceding concern, two of the welders had made specific mention of having seen this practice and one of them having had this happen to him. The welder personally examined the weld to determine if it looked good to him. Corporate Security felt that due to the similarity of events, the alleger may be referring to a previously identified welding foreman who was alleged to have committed similar acts. However, the NRC inspector, during a document review found several weld records that did not coincide with the rod withdrawal slips. These examples included Traveler No. B-1-3603-652-040 which contained inspection report (IR) No. B-1-652-040-02, "Welding Checklist Continuation Sheet" which identified Field Weld F17 as having been performed by welder FD-402 using welding process CSP-FD-HV-501 (SMAW) E7018 rod on April 12, 1989. The NRC inspectors documentation and rod withdrawal review determined that welder FD-402 had not withdrawn any E7018 rod that day. Additionally, for Traveler B-1-3603-654-068, IR No. B-1-654-068-02, the welding checklist continuation sheet identifies that welder FD-98 made welds F-49 and F-52 using welding process CSP-FD-HV-501 and E7018 rod withdrawn on April 3, 1989. However, during a document and rod slip withdrawal review, the NRC inspector determined that welder FD-98 had not withdrawn any E7018 rod on April 3, 1989.

Based on a detailed review of the welding checklist continuation sheets and the rod withdrawal slips for specific welds performed on corresponding days, the NRC inspector determined that this aspect of the allegation which dealt with weld record discrepancies was supported by documentation inconsistencies.

This failure on the part of the applicant to maintain accurate documentation related to weld records which provide evidence of activities affecting quality is a violation (445/8973-V-02), failure to maintain proper records.

During the process of reviewing this allegation, the NRC inspector determined that the Corporate Security investigation into these matters appeared to be thorough and timely. As evidenced by an examination of the SAFETEAM files, Corporate Security was provided with Concerns 12496 and 12497 relative to HVAC welding concerns on or about June 28, 1989. Shortly after this date, the NRC inspector met with Corporate Security personnel involved in the investigation and determined that they were actively involved in the investigation. The NRC inspector also determined that on July 18, 1989, Corporate Security requested an engineering evaluation and response regarding HVAC welding procedural violations. A response to this request for an engineering evaluation was provided by the Consolidated Engineering and Construction Organization (CECO) by letter CECO-2284 dated August 9, 1989. This letter stated in part that Engineering had previously accepted GMAW welds at butt joints that had been specified to be SMAW welded and that they could "still accept work subject to allegations on this issue." This letter went on to state that CAR 88-39 had addressed welds which had been made "out of procedure" and that the CAR had determined that there was no impact on the structural integrity of the reinforced duct. The NRC inspector determined that although these engineering evaluations were technically correct, the associated significance of craft personnel failing to follow procedures as well as the implications that weld records may have been adversely affected were not adequately addressed by TU Electric management. This determination was based on a review of the applicant's documentation contained in the SAFETEAM files, Corporate Security records, and discussions with the applicant's licensing and QA organizations. Prior to these items being identified to TU Electric's management during the NRC exit conducted on September 5, 1989, there was no discernible indication that this issue was being resolved expeditiously or that the adverse implications were being adequately addressed. The applicant's failure to take prompt corrective action in pursuing these issues is a violation (445/8973-V-03).

This allegation will remain open pending the completion of inspection activities in the electrical area.

6. <u>Electrical Components and Systems (51051, 51053, 51055, 51061, and 51063)</u>

During this reporting period, the NRC inspectors performed direct inspections of work performance to determine if the technical requirements contained in the applicant's Final Safety Analysis Report (FSAR) for safety-related electrical systems and components had been adequately translated into applicable drawings, procedures, and instructions. Additionally, the NRC inspectors evaluated the applicant's work control program to determine if the specified documents and procedures were of sufficient detail to provide adequate work performance and control.

In particular, the NRC inspector observed portions of the cable pulling for package CP1-ECPRLV-01. The cable pull was part of DCA 72619, Revision 4, which involved Class 1E associated cables A0150553 and AG150554 for the remote shutdown panel. The NRC inspector observed that QC personnel were present during the pull and that the craft personnel involved handling the cable during the pull correctly implemented the specified requirements. The

NRC inspector also reviewed the documentation package present at the pull site and determined that there were no discrepancies.

The NRC inspector also observed the performance of activities associated with Work Order C89-12527 involving motor operated valve (MOV) 1-HV-4288. The spring pack for this valve was dimensionally checked by test engineering personnel who recorded the required "as found" data and then reassembled the spring pack and performed "as-left" preloading tests. The NRC inspector determined that the accompanying documentation appeared complete, that QC was present for the required verifications, and that the test personnel were knowledgeable regarding the operation and testing of the hardware. The inspector also observed a portion of the MOVATS static testing of MOV 1HV-2494B related to NRC Inspection and Enforcement Bulletin 85-03 actions. Further inspections in these areas will be conducted and documented in subsequent inspection reports.

No violations or deviations were identified within this area.

7. Safety-Related Components, Mechanical (50072, 50073)

NDE, Reinstallation, and Reverse Flow Testing of Borg-Warner Check Valves

The applicant began a program to disassemble the approximately 80 Unit 1 Borg-Warner check valves and perform nondestructive evaluation (NDE) inspection on the swing arms following the failure of the swing arm for valve 1SW-0048 in the service water system (see NRC Inspection Reports 50-445/89-30, 50-446/89-30; and 50-445/89-64, 50-446/89-64). The inspection of the swing arms is now complete with 77 valves inspected and 14 rejected swing arms. The causes of the rejections are summarized below:

- Eight swing arms failed the examination by replication and exhibited hot cracks (i.e., poor casting quality).
- Four swing arms failed due to the minimum wall thickness criteria (i.e., insufficient material to perform replications).
- Two swing arms failed due to the presence of linear indications discovered during liquid penetrant and/or visual examinations.

In addition, 3 valves were not inspected; thus, the swing arms for these valves were rejected. The swing arms for two accepted valves were used for off-site materials analysis. In summary, the swing arms for 19 valves were replaced with either Unit 2 swing arms (8 valves) or new swing arms purchased from Borg-Warner (11 valves). The NRC inspectors are continuing to follow the swing arm replacement program and the results will be documented in a subsequent report.

The NRC inspector reviewed the applicant's program to reverse flow test the Borg-Warner check valves following reassembly. Procedure EGT-328A, Revision 1 is used for testing of the pressure seal check valves in the AFW system. This procedure was previously reviewed by the NRC inspector (see NRC Inspection Reports 50-445/89-64; 50-446/89-64) and found to be acceptable. Procedure EGT-716A is used to test the six Borg-Warner bolted bonnet valves which are designated as containment isolation valves. Procedure EGT-165, Revision 0, "Check Valve Reverse Flow Functional Test" is used for the balance of the Borg-Warner pressure seal and bolted bonnet valves. The NRC inspector reviewed EGT-165 and identified the following concerns to the applicant:

- In the statement of purpose, the procedure referenced nonsafety valves only. The procedure was, however, intended to be used for safety-related valves.
- The acceptance criteria in paragraph 2.1.1 was vague.
- The pressure of the test source (demineralized water) was not required to be recorded following the test.

Following discussions with the NRC inspector, the applicant amended the procedure with Procedure Change form EGT-165-RO-1 which adequately addressed these concerns.

Additionally, during this report period, the NRC inspector observed the disassembly of valve 1AF-057 and witnessed the reverse flow leak testing of the following valves:

1CC-0713, 8" bolted bonnet, Procedure EGT-716A
1AF-0167, 8" bolted bonnet, Procedure EGT-165

The NRC inspector determined that the test personnel involved appeared knowledgeable and that they efficiently performed the subject tests and valve disassembly. Both tests had satisfactory results and no discrepancies were identified during the test performance or documentation completion.

The reverse flow testing of the 80 Unit 1 Borg-Warner check valves is approximately 65 percent complete. Valve 1AF-0057 failed the reverse flow test apparently due to a combination of body/bonnet rotational misalignment and incorrect bonnet height. Valve 1CA-0016 (a containment isolation valve subject to very strict leak rate requirements) failed apparently due to an axial play problem between the arm and the disk. NRC review of the root cause and generic implications of the failure of Borg-Warner check valves to pass the reverse flow leak test and the adequacy of the inspection process prior to reassembly is identified as an open item pending the applicant's implementation of corrective actions (445/8973-O-04).

Additionally, the following open items were identified by the NRC AIT subsequent to their inspection concerning the multiple failures of Borg-Warner swing check valves experienced at Comanche Peak during the recent performance of HFT. (See NRC Inspection Report 50-445/89-30; 50-446/89-30). They are listed in this report to insure applicant action and followup and will be evaluated during future inspections.

a. In 1985, Failure Analysis Report FA 85-001, Revision 0, correctly identified the root cause of 1MS142 check valve failure as the bonnet and retainer incorrectly placed too low in the body. The applicant revised the root cause after Borg-Warner apparently convinced them that the valve failure was not due to incorrect installation. This item is open pending receipt of additional information from the applicant regarding documentation of the 1985 discussions with Borg-Warner which led to the decision that the valves were correctly reinstalled (445/8973-O-05).

b. The present design of the AFW system apparently does not allow for a thorough flushing of sections of the system using the existing drain valves. As discussed in the NRC AIT Inspection Report (50-0445/89-30; 50-446/89-30), numerous drain downs of the AFW piping have been accomplished over the years in order to perform welding repairs. Check valve internals were removed to provide the appropriate drain paths. Records of this activity do not appear to be available. It would appear that this activity may be related to the check valve failures. This item is open pending NRC review of: (1) the adequacy of the existing AFW drain valves for thorough system flushing, (2) applicant action to install additional drain valves in the AFW system, and (3) the applicant's plans to use the Borg-Warner check valves in the AFW system as system drains in the future (445/8973-O-06).

c. Based on reviews of maintenance histories and discussions with personnel, the NRC is concerned that no provisions were made for continued maintenance and system preservation during the period from completion of preoperational testing in 1984 until the recently recompleted HFT. This item is open pending receipt of information concerning maintenance and system preservation during this period (445/8973-O-07).

d. The applicant informed the AIT of their intent to administratively isolate the feedwater isolation bypass valves during startup and shutdown conditions except when the valves are actually needed. This would be done by closing the manual block valves in the feedwater isolation bypass line. The applicant is also considering eliminating the currently installed interlock between the feedwater isolation bypass valves and the feedwater preheater bypass valves. This interlock currently forces one of these two valves to be open and the other closed at all times other than during a feedwater isolation signal (when both close). This item is open pending completion of the applicant's action and subsequent NRC review (445/8973-O-08).

e. As a result of the AFW backflow events, approximately 70 of the 563 supports, restraints, and anchors used in the AFW piping system experienced loads in excess of the design loads. In addition , several areas in the piping experienced thermal stresses higher than ASME code allowables. Two areas of concern are the elbow adjacent to the failed support and some instrument connections. NRC review of the completed engineering analysis of the effects of the AFW backflow events on the AFW piping system is an open item (445/8973-O-09).

f. There was a perception among those interviewed by the AIT that the use of remote valve operators is prevalent. The design and placement of some of these operators appears to have been executed without proper regard to human factors issues. For example, the recirculation test line isolation valve on one motor driven AFW pump has a chain operator, while the equivalent valve on the other pump is manipulated with reach rods. This item is open pending applicant review of the use of these remote valve operators (445/8973-O-10).

g. Immediately prior to the April 23, 1989, AFW backleakage event, the control room operators sent only one auxiliary operator, near the end of the shift, to operate valves 1AF041 and 1AF042. This reflects a lack of understanding in the control room regarding task manpower assignments. The control room operators should have been aware of the time required for one individual to operate these valves. This item is open pending applicant action to ensure the control room operators are aware of the manpower requirements for required tasks (445/8973-O-11).

h. The NRC considers the difficulty of operation of valves 1AF041 and 1AF054 to be a contributing cause to the April 23 and May 5 events, but of minor safety significance. The NRC will review the applicant's intended actions to make these valves easier to operate. This is an open item (445/8973-O-12).

i. Check valve axial play is the total amount of movement within the disk arm socket in the axial direction. In order to assure that axial play would not adversely affect operability of the check valves, Borg-Warner was to establish acceptance criteria for the maximum and minimum axial play. The acceptance criteria and the applicant's review and approval of this acceptance criteria, based on the calculation procedure established for determining the bonnet height adjustment, is an open item (445/8973-O-13).

j. The NRC is concerned that the AFW backleakage events reflect negatively on the quality of training received by the plant operators. The necessity of in-sequence valve operation was apparently not sufficiently emphasized. Another training-related concern was the failure of plant operators to document the discovery of three failed AFW check valves on a Plant Identification Report (PIR) or an NCR. The applicant has committed to raising the awareness of plant operators to operational issues by conducting training. NRC review of this training of operators is an open item (445/8973-O-14).

k. Kalsi Engineering, Inc., is assisting TU Electric in developing and implementing a program based on recommendations contained in SOER-86-03, "Check Valve Failure or Degradation." NRC review of this program, including TU Electric's commitment to either modify the three AFW minimum flow recirculation check valves (1AF-045, -057, and -069) or to increase the distance between the orifices and the subject check valves prior to fuel load, is an open item (445/8973-O-15).

8. Plant Tours (51063)

The NRC inspectors conducted routine plant tours during this inspection period which included evaluation of work in progress as well as completed work to determine if activities involving safety-related electrical systems and components including electrical cable were being controlled and accomplished in accordance with regulatory requirements, industry standards, and the applicant's procedures.

No violations or deviations were identified.

9. Open Items

Open items are matters which have been discussed with the applicant, which will be reviewed further by the inspector, and which involve some action on the part of the NRC or applicant or both. An open item disclosed during the inspection is discussed in paragraph 7. Eleven additional open items which resulted from the NRC AIT evaluation of multiple check valve failure experienced during HFT are also identified in paragraph 7 of this report.

10. Exit Meeting (30703)

An exit meeting was conducted October 3, 1989, with the applicant's representatives identified in paragraph 1 of this report. No written material was provided to the applicant by the inspectors during this reporting period. The applicant did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. During this meeting, the NRC inspectors summarized the scope and findings of the inspection.

Attachment P

CFLK3

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

May 16, 1990

Docket No. 50-445

LICENSEE: Texas Utilities Electric Company (TU Electric)

FACILITY: Comanche Peak Steam Electric Station (CPSES), Unit 1

SUBJECT: SUMMARY OF MEETING ON MAY 9, 1990 TO DISCUSS PROBLEMS WITH VALVES IN THE AUXILIARY FEEDWATER AND MAIN FEEDWATER SYSTEMS

On May 9, 1990, the staff met with representatives of TU Electric at the Comanche Peak site to discuss recent operational problems with leaking 4" Borg-Warner swing arm check valves in the auxiliary feedwater (AFW) system discharge piping and with main feedwater system isolation valves that fail to open due to thermal binding of the valve internals. A list of attendees at the meeting is provided as Enclosure 1. The slides used in TU Electric's presentation are provided as Enclosure 2.

With respect to the AFW valves that leak by resulting in elevated temperatures in the AFW system discharge piping, the licensee stated that, in the short term, it intends to perform upstream venting of the check valves in order to facilitate more positive seating of the valves. (These check valves tend to unseat slightly under low differential pressure conditions, allowing the leakage. Venting of the upstream piping provides greater differential pressure and hence, greater closing forces.) The licensee indicated that the evaluation of the leaking check valve problem would be complete by May 25, 1990. The evaluation report will contain recommendations for a long-term solution for the leaking check valves. Possible long-term solutions include modification of the existing check valves, replacement of five existing check valves with another design, or modification of the existing auxiliary feedwater system configuration for standby service. Licensee management indicated that it would inform NRC of its long-term actions prior to completing its 50% power plateau self-assessment.

The licensee also discussed the use of hydraulic lifting devices to facilitate the opening of the main feedwater system isolation valves. The NRC staff raised the concern that the use of such devices may result in damage to the valves because of possible excessive lifting forces. TU Electric stated that it performed an analysis that demonstrated that the use of hydraulic lifting devices was acceptable for long-term use. The licensee indicated that it intended to use the hydraulic lifting devices during the short term while it was evaluating long-term options.

Because there has been some leakage through the feedwater preheater bypass valves (FWPBVs) which has contributed to operational problems associated with the leaking AFW check valves, the staff questioned their operability. The licensee

TID

~~9005210150~~

stated that the leakage past the FWPBVs did not affect their operability because there are no leakage criteria in the plant technical specifications associated with the valves, and they can be isolated with a manual bypass valve.

James H. Wilson

James H. Wilson, Assistant Director
for Projects
Comanche Peak Project Division
Office of Nuclear Reactor Regulation

Enclosure:
1. List of Attendees
2. TU Electric's Presentation Slides

cc: See next page

cc w/enclosures:
Assistant Director
for Inspection Programs
Comanche Peak Project Division
U. S. Nuclear Regulatory Commission
P. O. Box 1029
Granbury, Texas 76048

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Ms. Billie Pirner Garde, Esq.
Robinson, Robinson, et al.
103 East College Avenue
Appleton, Wisconsin 54911

Mrs. Juanita Ellis, President
Citizens Association for Sound Energy
1426 South Polk
Dallas, Texas 75224

E. F. Ottney
P. O. Box 1777
Glen Rose, Texas 76043

Mr. Roger D. Walker
Manager, Nuclear Licensing
Texas Utilities Electric Company
400 North Olive Street, L. B. 81
Dallas, Texas 75201

Texas Utilities Electric Company
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, Maryland 20814

William A. Burchette, Esq.
Counsel for Tex-La Electric
Cooperative of Texas
Heron, Burchette, Ruckert & Rothwell
1025 Thomas Jefferson Street, NW
Washington, D.C. 20007

GDS ASSOCIATES, INC.
Suite 720
1850 Parkway Place
Marietta, Georgia 30067-8237

Jack R. Newman, Esq.
Newman & Holtzinger
1615 L Street, NW
Suite 1000
Washington, D.C. 20036

Chief, Texas Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756

Honorable George Crump
County Judge
Glen Rose, Texas 76043

Enclosure 1

NRC/TU ELECTRIC MEETING AT COMANCHE PEAK SITE
CONCERNING OPERATIONAL PROBLEMS WITH THE MAIN FEEDWATER
AND AUXILIARY FEEDWATER SYSTEMS

May 9, 1990

NRC	TU Electric	CASE
P. Gwynn	W. Cahill	E. Ottney
J. Jaudon	A. Scott	O. Thero
J. Wilson	J. Beck	
J. Wiebe	J. Kelley	
D. Chamberlain	C. Hogg	
A. Howell	R. Walker	
W. Johnson	M. Blevins	
R. Latta	K. Tipton	
S. Bitter	I. Whitt	
D. Graves	D. Reimer	
M. Malloy	S. Ellis	
	J. Donahue	
	S. Palmer	
	J. Boatwright	
	B. Rice	
	T. Jenkins	
	K. Bishop	
	T. Heatherly	
	M. Axelrad	

Enclosure 2

NRC MEETING

ELEVATED AFW PIPING TEMPERATURES DURING FEEDWATER STARTUP

CPSES

MAY 9, 1990

AGENDA

o	Introduction and Overview	Mike Blevins
o	Evaluation Team Report	Ken Tipton
o	Feedwater Isolation Valve Opening	Ivan Whitt
o	Summary & Questions	Jim Kelley

OVERVIEW

- Elevated Auxiliary Feedwater piping temperatures observed during plant startup.

- Responded promptly with technical and management assistance.

- Evaluated situations and took corrective action.

- Different problem than 1989 HFT
  - Check valves were not hung open.
  - No operator errors involved with the event.
  - Reverse flow was not from steam generators.

- Conclusions to date

- Current Plant Status

SYSTEM OPERATION SYNOPSIS

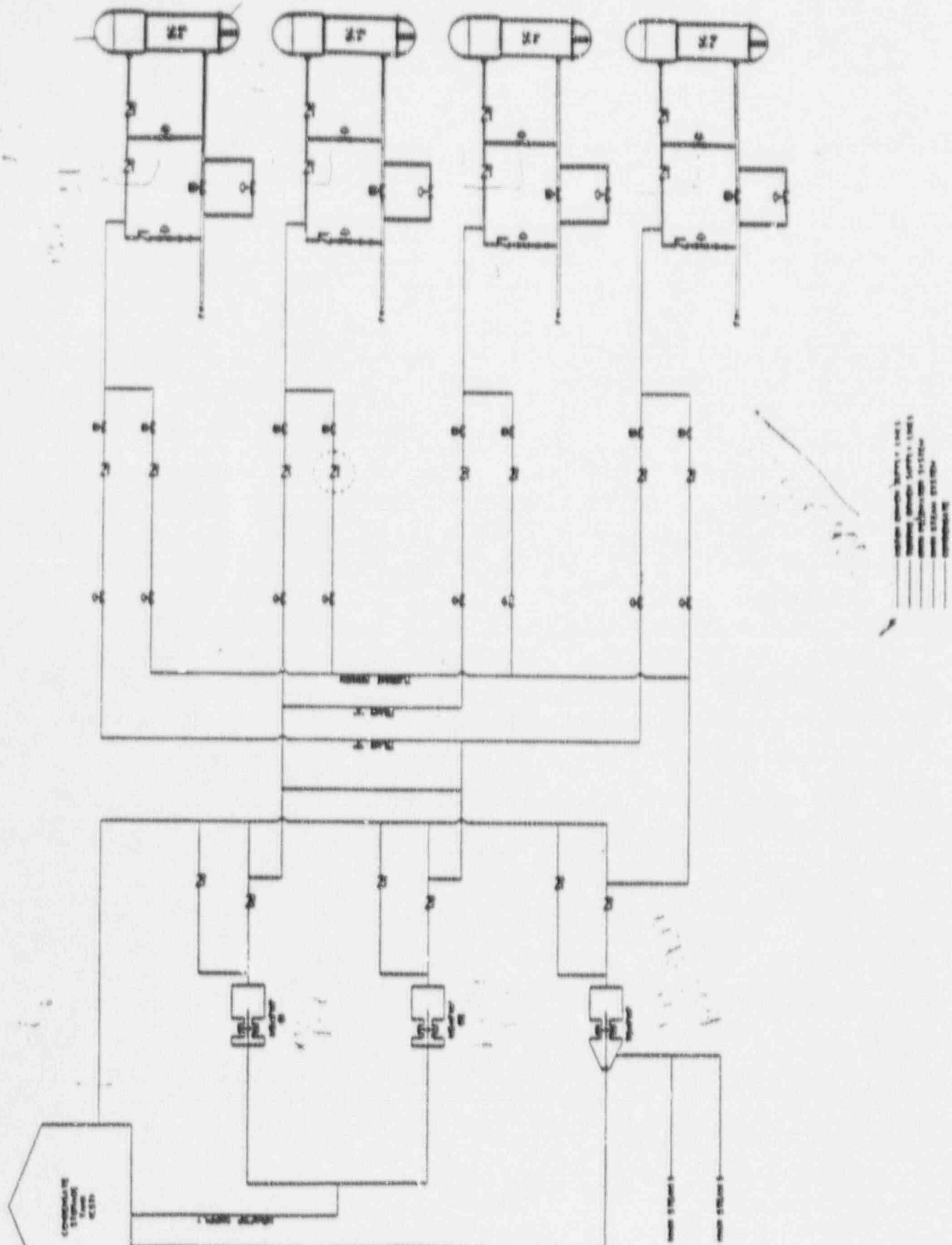
- AUXILIARY FEEDWATER FLOW PATH - Rx POWER LESS THAN 3%

- FEEDWATER STARTUP FLOW PATH

- FEEDWATER BYPASS FLOW PATH - PRIOR TO 250° F INTERLOCK

- TRANSITION FLOW PATH

- MAIN FEEDWATER FLOW PATH - Rx POWER GREATER THAN 30%



Attachment
Q

CP-89-01
IR 89 30, IR 89

TU

July 27, 1990

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-445
BORG WARNER/INTERNATIONAL PUMP, INC. (BW/IP)
CHECK VALVE SWING ARMS

Ref: TU Electric letter logged TXX-90139 from William J. Cahill to NRC dated April 9, 1990

Gentlemen:

In the referenced correspondence, TU Electric committed to provide a schedule for the replacement of installed BW/IP check valve swing arms within 90 days of the CPSES full power license. As discussed with members of the NRC staff, an extension for schedule submittal was granted until July 27, 1990. The following information regarding BW/IP check valve swing arm replacement is submitted.

?

There are no ASME Code Class 1 BW/IP check valves installed at CPSES. To date the swing arms of 24 BW/IP check valves have been replaced with investment cast swing arms. These include 3 ASME Code Class 2 valves in the Containment Spray System, 19 ASME Code Class 3 valves in various safety related systems and 2 BW/IP check valves in Non-ASME systems.

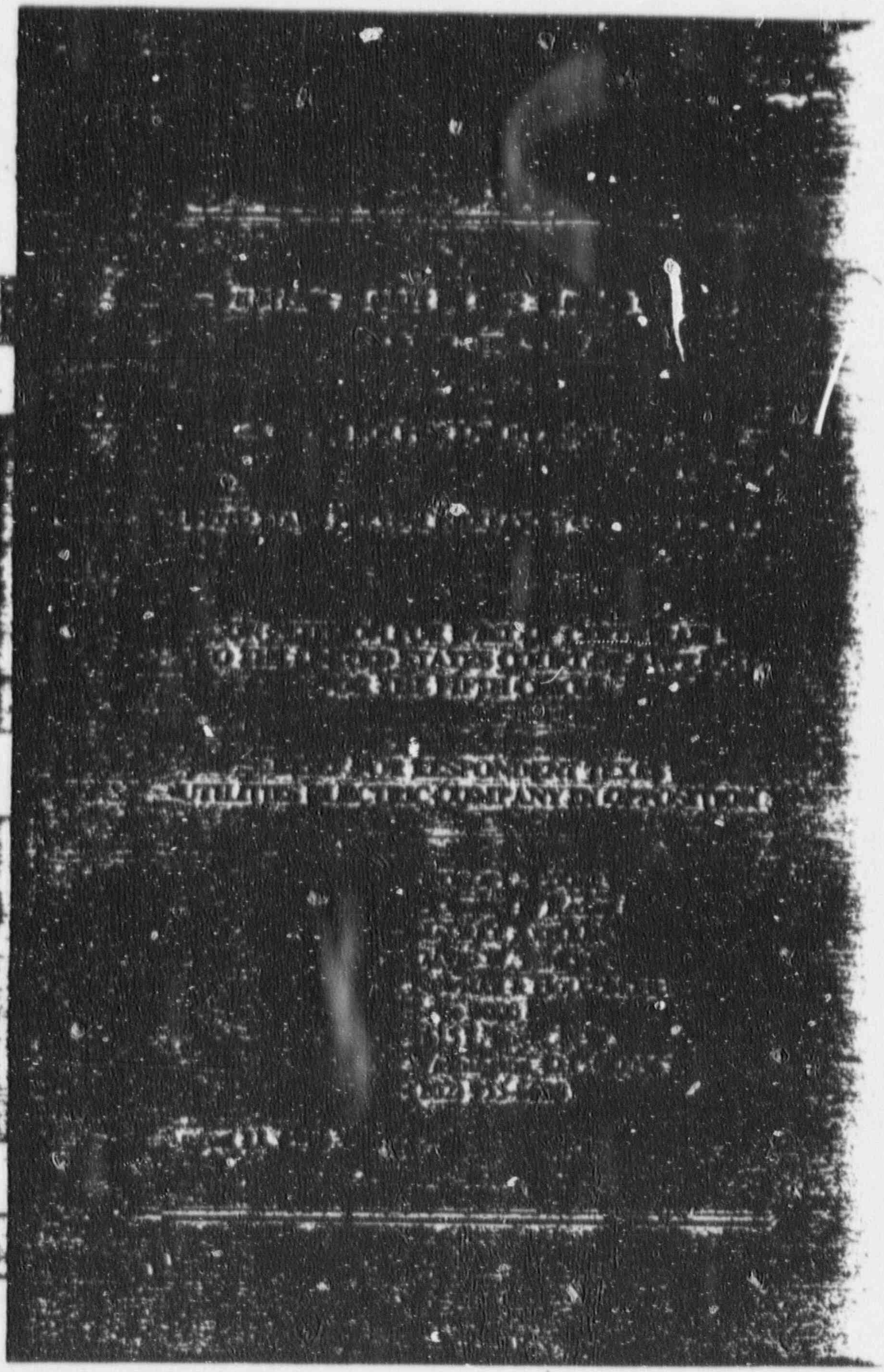
The replacement of swing arms in the remaining installed BW/IP check valves which do not contain investment type cast swing arms has been prioritized based on safety classification (ASME code class) and valve function. Generally, swing arms in ASME Code Class 2 check valves will be replaced during the first refueling outage consistent with the need to maintain systems in an operable status to support plant conditions during the [illegible] availability. Those remaining swing arms not replaced during the first refueling outage will be replaced during the second and [illegible] outages with priority again being given to any remaining [illegible] valves.

William J. Cahill, Jr.

DEN/daj

Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

Attachment
R



CASE as a party constitute a "recent event" providing a basis for CFUR's untimely intervention.

Finally, CFUR relies on *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-903, 28 NRC 499 (1988), and *Union of Concerned Scientists v. NRC*, 735 F.2d 1437 (D.C. Cir. 1984), *cert. denied sub nom. Arkansas Power & Light Co. v. Union of Concerned Scientists*, 469 U.S. 1132 (1985), to support its claim that certain (now corrected)[8] problems experienced by TU Electric with check valves constitute a "fundamental flaw" requiring the reopening of the record. Petition for Writ of Certiorari at 11-15. Apart from the fact that this legal issue was never raised before either the Commission or the Fifth Circuit, CFUR's argument is irrelevant to a determination of whether the denial of its intervention petition was an abuse of discretion for two reasons.

First, the case law regarding "fundamental flaws" has little or no application to problems experienced with specific pieces of equipment in a nuclear power plant. Rather, a fundamental flaw generally refers to serious programmatic or generic flaws in a program such as an emergency preparedness plan which is material to a licensing decision.[9] Indeed, the law could hardly be otherwise. Given the complexity of nuclear power plants, if any single problem with a piece of equipment could result in new hearings at the request of would-be intervenors, no nuclear power plant could escape virtually never-ending hearings.

---

[8] The problems experienced with the check valves were corrected by TU Electric and inspected by the NRC. The plant is now in commercial operation.

[9] *See Long Island Lighting Co.*, 28 NRC at 505 (a fundamental flaw "reflects a failure of an essential element of the [emergency preparedness] plan, *and*, second, it can be remedied only through a significant revision of the plan."). A fundamental flaw is never found on the basis of "minor or ad hoc problems" such as discrete and isolated equipment problems. *Union of Concerned Scientists*, 735 F.2d at 1448.

Second, CFUR does not explain how its new argument regarding a "fundamental flaw" in any way affects the Commission's decision that CFUR failed to establish good cause for its untimely intervention. At best, CFUR simply argues that if a "fundamental flaw" exists then the Commission, at the behest of *an existing party*, could have reopened the record. Whatever the merits of that proposition, it most assuredly adds nothing to the question of whether the Commission abused its discretion in finding that CFUR failed to demonstrate good cause for its untimely request for intervention.

II. The Petition For Writ Of Certiorari Does Not Raise An Important Unsettled Question Of Federal Law.

CFUR's Petition for Writ of Certiorari makes no attempt to affirmatively demonstrate that it satisfies the standards for grant of certiorari embodied in this Court's Rule 10. There is no attempt to show that the Fifth Circuit's decision conflicts with any decision of this Court, another United States court of appeals, or that of a state court of last resort. Sup. Ct. R. 10.1(a), (c). Nor is there any claim that the Fifth Circuit "departed from the accepted and usual course of judicial proceedings . . . ." Sup. Ct. R. 10.1(a). At best, CFUR is perhaps implicitly arguing that the Fifth Circuit has decided an important question of federal law which has not been, but should be, settled by this Court. *See* Sup. Ct. R. 10.1(c).

CFUR's Petition revolves around well-settled and straightforward questions of federal law. CFUR is not contesting the validity of the NRC's longstanding regulation or case law governing untimely petitions. Rather, CFUR is contesting an exercise of the Commission's discretion to deny CFUR's petition to intervene and request for a hearing that was filed nine years out-of-time, six years after CFUR had already withdrawn from the Comanche Peak operating license hearings, and one month after those hearings had been duly dismissed. In reality, the issue presented by CFUR's Petition for Writ of Certiorari is whether the Commission

Attachment
S

for itself as far as assessing management attitude. The performance was lousy.

I mean, there were check valve failures that historically failed. These check valves had historical failures.

There also was a problem industry-wide that they should have known about, obviously. The NRC had sent out I&E bulletins -- not I&E but the information bulletins about Borg Warner problems throughout the industry.

That's according to my understanding in the July 10th report.

So if in fact this is what you base this on, the performance, then the performance was pretty bad.

What kept jumping out at us as we read this report was that management philosophy was not sufficient to operate safely a nuclear powerplant.

I think those are very strong words coming from the regulatory agency that must decide on licensing a plant.

I guess our concern is, is there ever a point in time where you look at a utility and say, "We will give you no more time to get it right"?

MR. GRIMES: That's an interesting question.

[Laughter.]

MR. GRIMES: There again, I think you've raised a policy issue. That certainly is at least philosophically interesting and it's a matter that we can bring up with our Enforcement Staff and with the Commission itself, perhaps, in terms of: "Is there ever any occasion where a matter is so serious that it warrants stopping a process?"

To my knowledge, there are usually only two paths. One is enforcement and the other one is issuance of an order to show cause why a license might not be revoked.

I've been involved in a number of those cases. I've been involved in the issuance of such orders to have utilities show cause why their license shouldn't be revoked, because they've shown a pattern of serious problems.

Those have normally followed the issuance of enforcement actions that are severity level one or two.

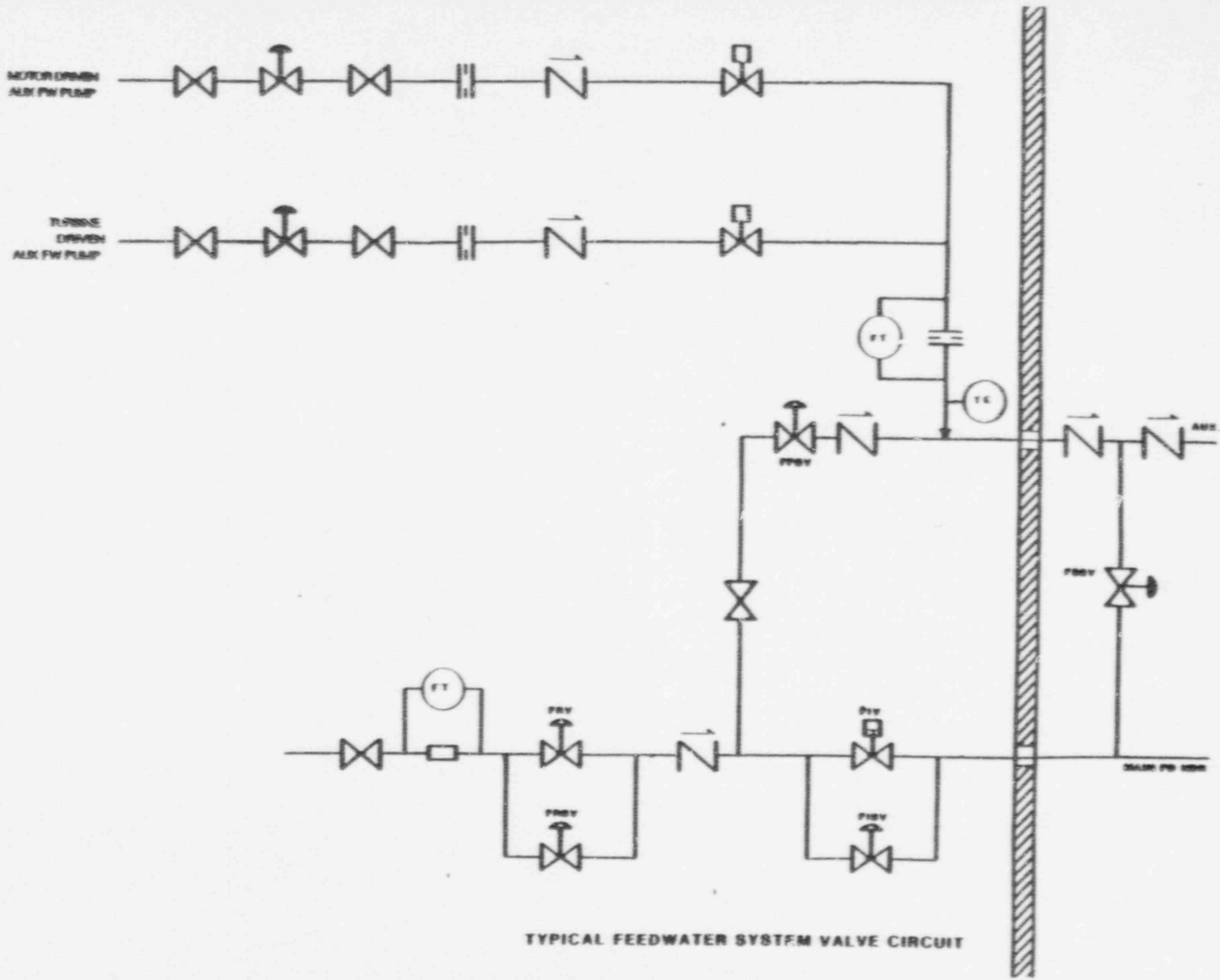
That is, they are matters where they made mistakes that are so bad that they have actually put public health and safety at risk.

They normally only get that opportunity after the license is issued.

MS. BRINK: That's a little chilling.

MR. GRIMES: During construction there are very few things that you can screw up so bad that you've actually put the public health and safety at jeopardy.

We intend to be looking in terms of the Readiness Team at the management attitudes and the operators' attitudes about how they would operate the plant.



TYPICAL FEEDWATER SYSTEM VALVE CIRCUIT

INITIAL PLANT CONDITIONS

- Synchronized - 100 MWe
- Extraction Steam in service
- Increasing Feedwater Temperature

TIME LINE

4/24/90 1642 Operator observes AF line temperatures increasing with Feedwater temperature as indicated on main control board temperature indicators. Check valve back-leakage suspected.

1645 Actions per abnormal condition procedure (ABN) initiated

2000 Technical team formed to evaluate operability, initiate required preliminary corrective actions.

2200 Piping temperatures in pump rooms 165 F.

4/25/90 0100 ONE Form initiated to document the AFW check valve back-leakage and determine operability.

0251 Check valves reseated per ABN procedure.

- No valves hung open
- Leak rates quantified
- Valves will perform required safety function
- No temperature effects on piping
- Operability not affected

0755 Temperatures increasing again. Operations suspects check valve back-leakage and initiates actions per ABN

0800 Evaluation team established to investigate/evaluate AF elevated line temperatures

1600 Management meeting to discuss status of evaluation team

4/26/90 0400 Performance and Test personnel verify check valve back-leakage rates.

0800 Management meeting to discuss short-term action options.

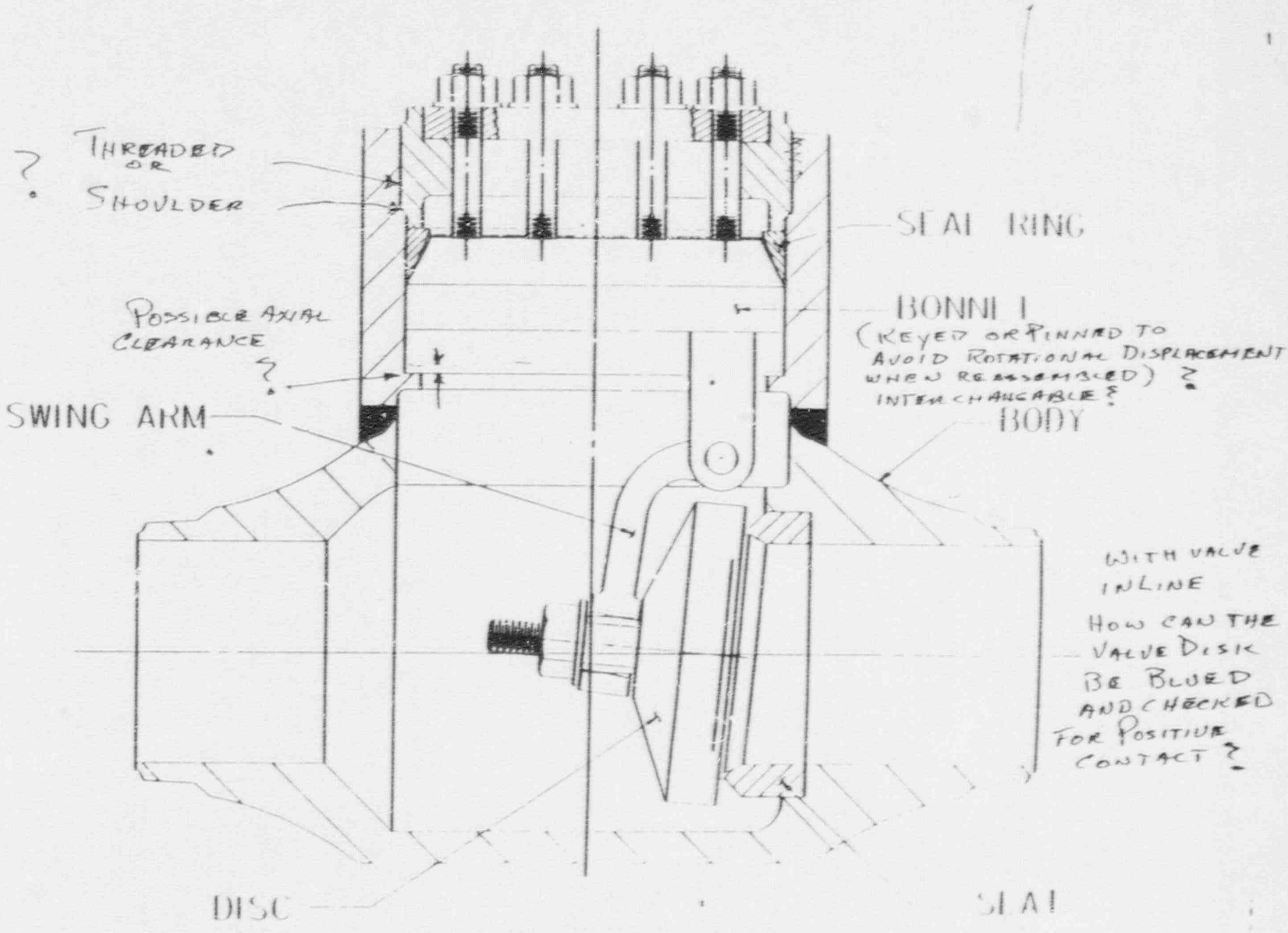
1600 Management meeting to discuss short-term corrective actions

- Controlled periodic venting of upstream AF piping.

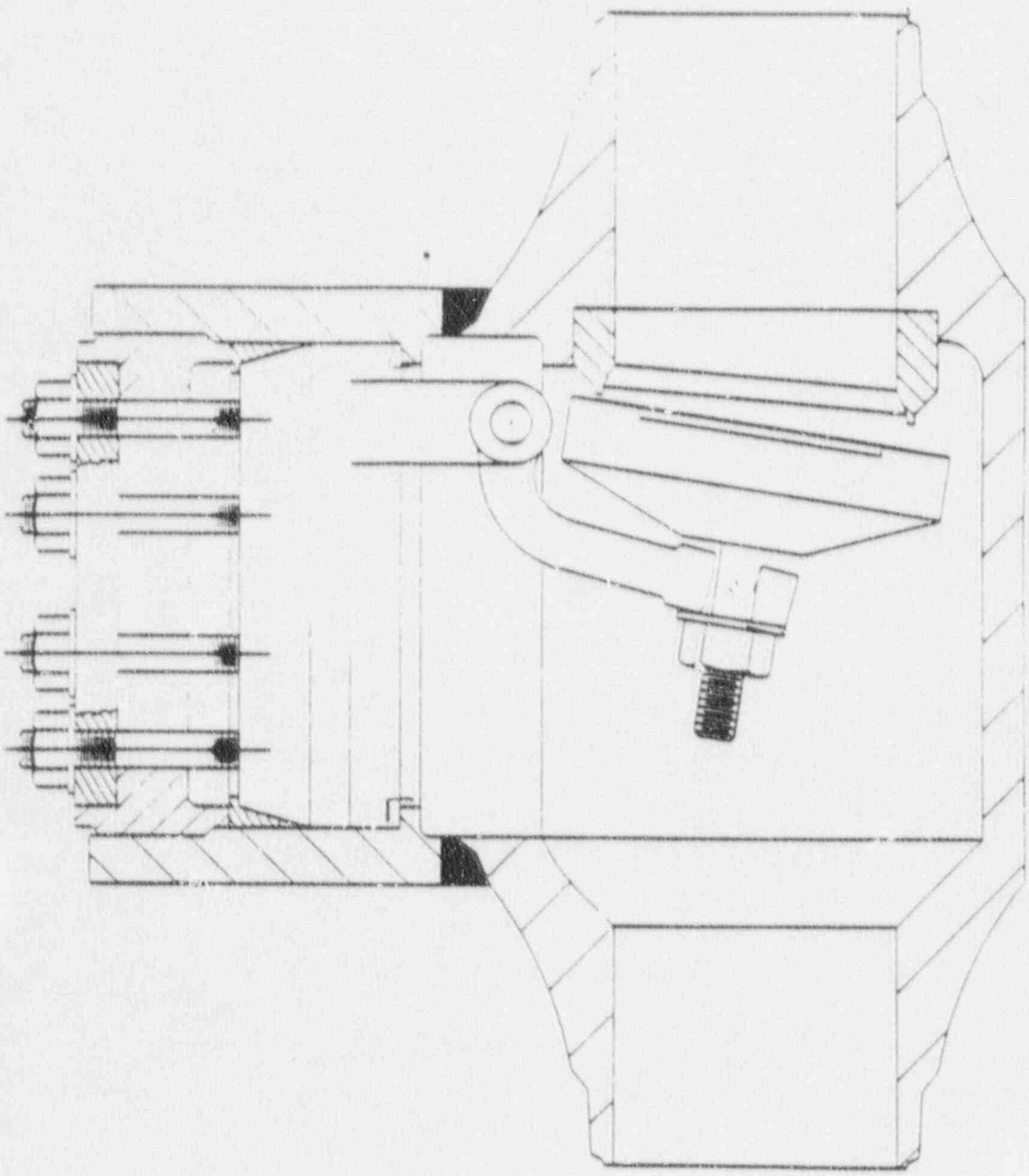
1700 Technical Evaluation to provide guidance on venting.

TIME LINE

4/27/90	Procedure change to Operations procedure to provide for controlled periodic manual venting. FWIV's opened. FWPBV's closed.
4/28/90	2 of 4 Feedwater Preheater Bypass valves leaking by Procedure change to Operations procedure to provide for isolating Feedwater Preheat Bypass valves above 30% reactor power through use of upstream manual isolation valve. Valves not iso lated at this time. Manually tripped Main Turbine to repair leaking pressure transmitter. FWIV's closed, FWBV's opened.
4/29/90	Main Generator back on grid. Opened FWIV's, FWPBV's closed/not isolated.
4/30/90	AF line temperatures increasing. Actions initiated per ABN procedure. Temperatures still increasing. Actions re-initiated per ABN. AF line temp reaches 235 degrees. Check valve reseated per ABN. Line temperatures decreasing. Upstream line pressure drc[illegible] [illegible]ow 50 psi limit to 25 psi (CST pressure).
5/1/90	Feedwater Preheater Bypass valves manually isolated. Auxiliary Feedwater system in normal standby operation, piping at ambient temperature.



VALVE CLOSED



FEEDWATER PREHEATER BYPASS VALVES
CORRECTIVE ACTION

- o THE MANUAL ISOLATION VALVES WERE CLOSED AFTER THE WATER HAMMER INTERLOCK WAS CLEARED AND FLOW THROUGH MAIN FEEDWATER ISOLATION VALVE ESTABLISHED.

- o ISOLATING FPBV MANUALLY WILL MAINTAIN AFW PIPING TEMPERATURES BELOW 210°F WITHOUT OPERATOR ACTION.

- o AT THE NEXT MAINTENANCE OUTAGE THE VALVES IDENTIFIED AS LEAKING WILL BE REPAIRED.

EVALUATION TEAM

CORRECTIVE ACTIONS UNDER CONSIDERATION

SHORT TERM

- o CONTROLLED PERIODIC VENTING

LONG TERM

- o MODIFY EXISTING CHECK VALVES TO PROVIDE MORE POSITIVE SEATING
- o REWORK FEEDWATER PREHEATER BYPASS VALVES DURING A MAINTENANCE OUTAGE
- o MODIFY AUXILIARY FEEDWATER SYSTEM CONFIGURATION FOR STANDBY SERVICE
- o REPLACE EXISTING CHECK VALVES WITH ANOTHER DESIGN
- o PROVIDE TEMPERING FLOW FROM AFW THROUGH FEEDWATER TRANSITION

FEEDWATER ISOLATION VALVES
DESIGN BASIS SUMMARY

- o REQUIRED TO ISOLATE CONTAINMENT.

- o REQUIRED TO ISOLATE FEEDWATER TO MINIMIZE MASS AND ENERGY RELEASE INSIDE CONTAINMENT DURING A LINE BREAK EVENT AND MINIMIZE RCS COOLDOWN.

- o CLOSE ON LOW FEEDWATER TEMPERATURE AS PART OF STEAM GENERATOR WATER HAMMER PREVENTION.

FEEDWATER ISOLATION VALVES

PROBLEM SUMMARY

- Operations tried to open the Feedwater Isolation Valve with the handswitches in the Control Room.

- All four Feedwater Isolation Valves would not open.

- Problem determined to be mechanical binding due to thermal growth of valve internals.

FEEDWATER ISOLATION VALVES

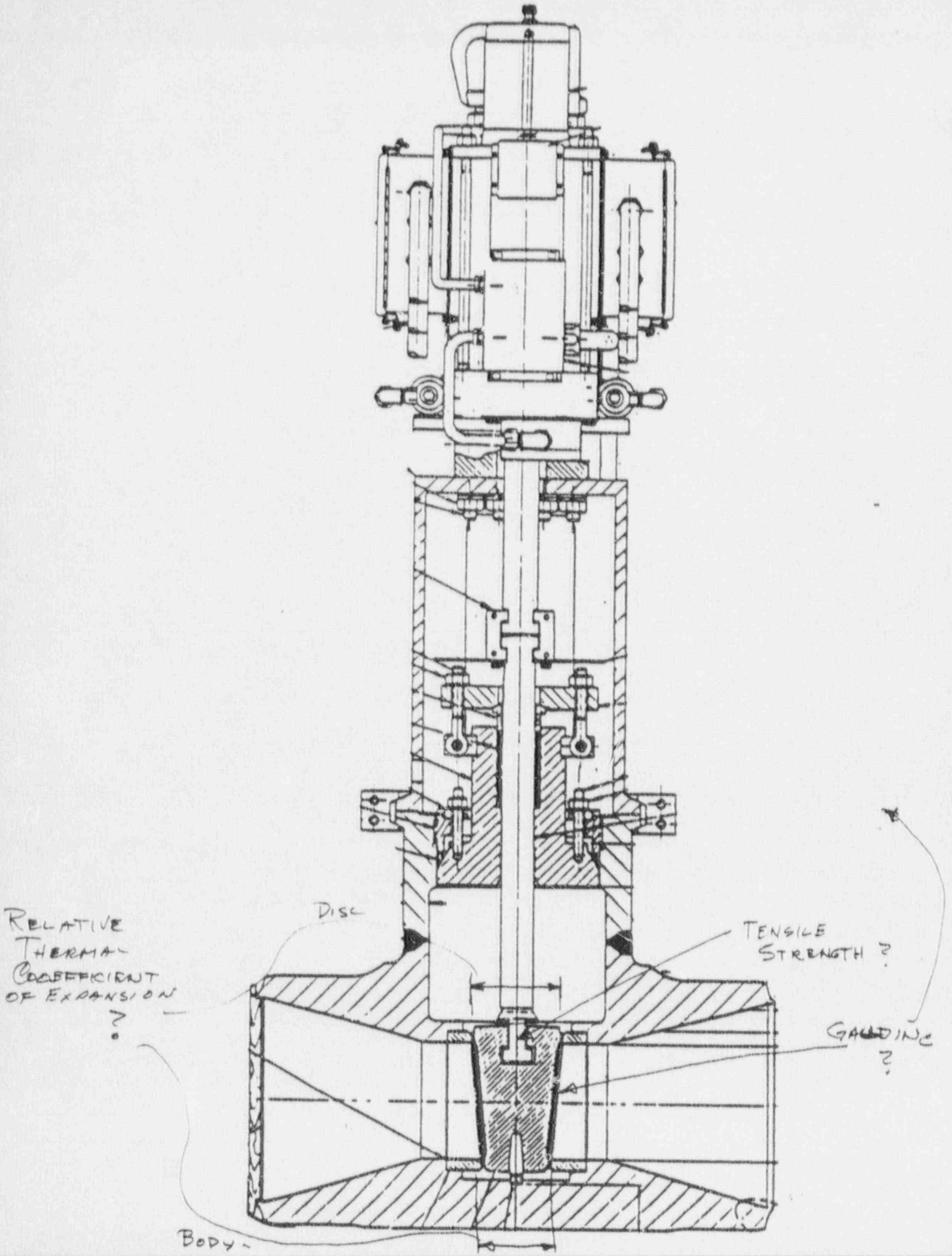
CORRECTIVE ACTION

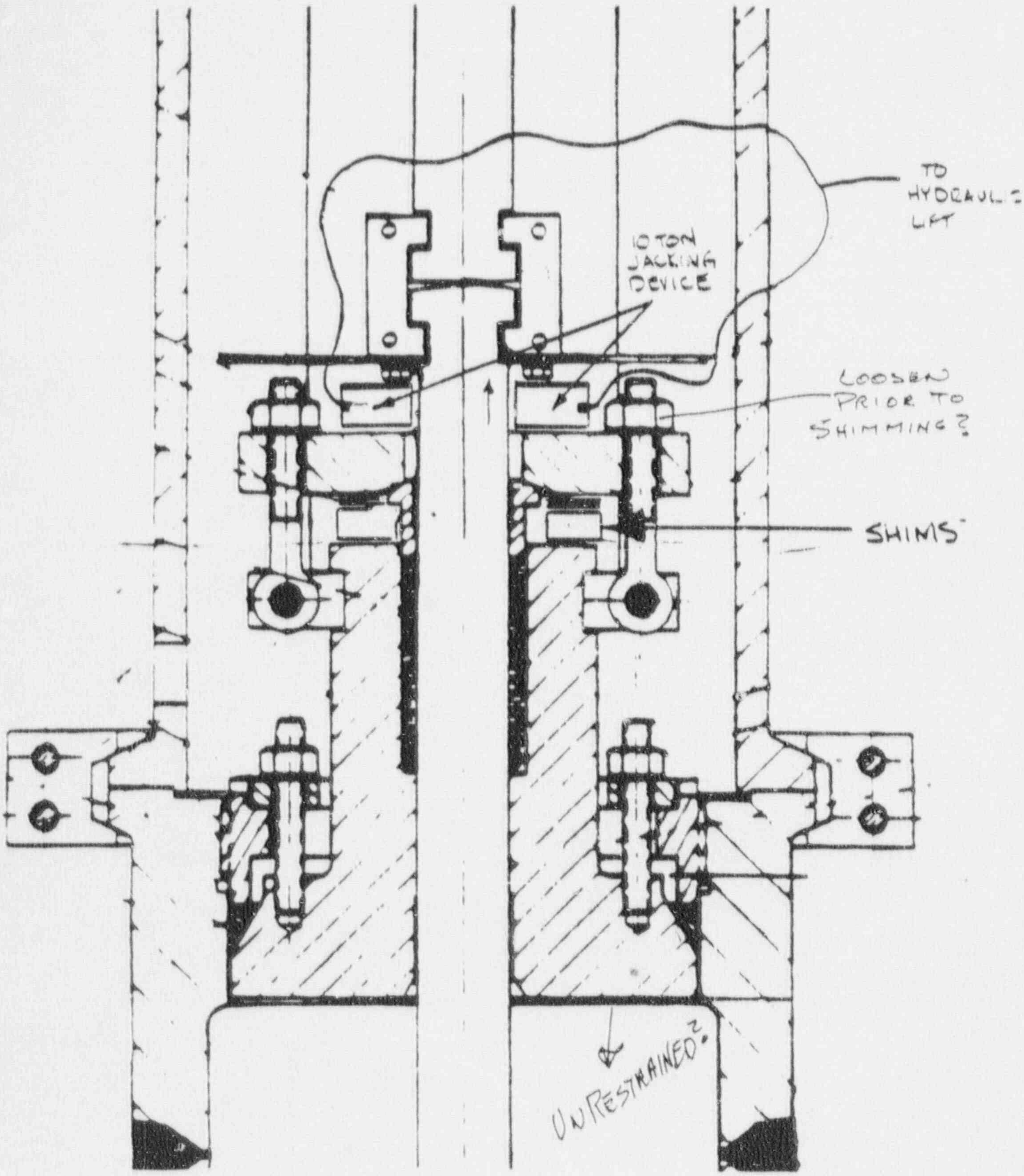
- SHORT TERM:

  A HYDRAULIC LIFTING DEVICE WILL BE USED TO ASSIST OPENING OF THE FEEDWATER ISOLATION VALVES.

- LONG TERM:

  STILL EVALUATING LONG TERM CORRECTIVE ACTIONS.


RELATIVE
THERMAL
COEFFICIENT
OF EXPANSION
?
DISC
TENSILE
STRENGTH ?
GALLING
?
BODY



TO HYDRAULIC LIFT
10 TON JACKING DEVICE
LOOSEN PRIOR TO SHIMMING?
SHIMS
UNRESTRAINED?