TXU

TXU Electric
Comanche Peak
Steam Electric Station
P.O. Box 1002
Glen Rose, TX 76043
Tel: 254 897 8920
Fax: 254 897 6652
lterry1@txu.com

C. Lance Terry
Senior Vice President & Principal Nuclear Officer

Ref: 10CFR50.90
10CFR50.91

CPSES-200001271
Log # TXX-00110
File # 00236

May 19, 2000

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) 00-03
REVISION TO TECHNICAL SPECIFICATION (TS) 3.7.3
FEEDWATER ISOLATION VALVES

REF:
1) TXU letter logged TXX-00077, dated May 12, 2000, from C. L. Terry to the NRC.

2) TXU letter logged TXX-00106, dated May 16, 2000, from C. L. Terry to the NRC.

3) TXU letter logged TXX-00019, dated January 19, 2000, from C. L. Terry to the NRC.

Gentlemen:

This letter supplements licensing amendment request (LAR 00-03) transmitted by reference 1. The supplement supercedes LAR 00-03 in its entirety and requests that the revised LAR 00-03 be processed as an exigent change per 10 CFR 50.91. In addition, this letter withdraws the request for enforcement discretion transmitted by reference 2.

The Commission's regulations in 10 CFR 50.91 contain provisions for issuance of an amendment where the Commission finds that exigent circumstances exist, in that a licensee and the Commission must act quickly and that the time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment. 10 CFR 50.91(a)(6)(vi) requires that the licensee explain the exigency and

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why the licensee cannot avoid it and use the normal public notice and comment procedures in 10 CFR 50.91(a)(2). TXU Electric believes an exigency exists as described below.

Based on the root cause analysis performed in conjunction with the repair of the hydraulic pump for Feedwater Isolation Valve (FIV 1-03), (see reference 3) it is probable that each of the Unit 1 and Unit 2 FIV hydraulic pumps contain seals made from a material that is not in accordance with vendor drawings and that is not compatible with the hydraulic fluid. Currently, one of these pumps (FIV 1-04) is exhibiting early symptoms of seal degradation, however, the time to failure of the hydraulic pump cannot be accurately predicted. A sudden failure of a seal could cause the affected FIV to close which would likely result in a reactor trip. TXU Electric believes that it is prudent to repair the pump prior to the summer load demand when the potential adverse impact to the public from a plant shutdown is greater. The proposed change would extend the Completion Time for one or more FIVs inoperable to allow sufficient time to repair the FIV hydraulic system.

TXU Electric requests approval of LAR 00-03 by June 2, 2000, to be implemented within 10 days of the issuance of the license amendment. Failure to receive the license amendment in a timely manner potentially affects the capability of the units to continue to operate at full power. The approval date was selected to minimize the time before repairs to the hydraulic pump (FIV 1-04) could be accomplished.

This communication contains no new licensing basis commitments regarding CPSES Units 1 and 2.

In accordance with 10CFR50.91(b), TXU Electric is providing the State of Texas with a copy of this amendment supplement.

Should you have any questions, please contact Mr. Bob Dacko at (254) 897-0122.

Sincerely,

C. L. Terry

By: Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

BSD/bsd

Attachments
1. Affidavit
2. Description and Assessment
3. Markup of Technical Specifications pages
4. Markup of Technical Specifications Bases pages (for information)
5. Retyped Technical Specification Pages
6. Retyped Technical Specification Bases Pages (for information)
7. Reference Information (for information)

c - E. W. Merschoff, Region IV
J. I. Tapia, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Mr. Authur C. Tate
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78704

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)		
	)		
TXU Electric	)	Docket Nos.	50-445
	)		50-446
(Comanche Peak Steam Electric Station,	)	License Nos.	NPF-87
Units 1 & 2)	)		NPF-89

AFFIDAVIT

Roger D. Walker being duly sworn, hereby deposes and says that he is Regulatory Affairs Manager of TXU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this supplement to License Amendment Request 00-03; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

STATE OF TEXAS)
)
COUNTY OF Somervell)

Subscribed and sworn to before me, on this 18th day of May, 2000.

Dayle R. Peck Jespersen

Notary Public

Dayle R. Peck Jespersen
Notary Public, State of Texas
My Comm. Expires 08/13/22

ATTACHMENT 2 to TXX-00110

DESCRIPTION AND ASSESSMENT

Description and Assessment

1.0 INTRODUCTION

1.1 Proposed change LAR-00-003 is a request to revise Technical Specifications (TS) 3.7.3, "Feedwater Isolation Valves (FIVs) and Associated Bypass Valves," for Comanche Peak Steam Electric Station (CPSES) Units 1 and 2.

1.2 MARKUP OF EXISTING TECHNICAL SPECIFICATIONS AND BASES

Technical Specifications:	See Attachment 3
Technical Specifications Bases:	See Attachment 4

1.3 PROPOSED TECHNICAL SPECIFICATIONS AND BASES

Technical Specifications:	See Attachment 5
Technical Specifications Bases:	See Attachment 6

1.4 FINAL SAFETY ANALYSIS REPORT (FSAR) SECTION

The evaluations performed in support of this License Amendment Request do not result in any required changes to the FSAR per 10CFR50.71(e), the guidance provided by Regulatory Guide 1.181 "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," and NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports."

2.0 DESCRIPTION

The proposed change would revise TS 3.7.3, Condition A to extend the Completion Time for one or more inoperable FIVs from 4 hours to 24 hours, if within fours hours, the respective FCVs and the FCV bypass valves in the same flowpath(s) as the inoperable FIV(s) are verified to be capable of performing the feedwater isolation function. A footnote is added that indicates that the extension of the Completion Time to 24 hours is only applicable for repair of the FIV hydraulic system through fuel cycle 8 for Unit 1 and fuel cycle 5 for Unit 2.

For Information purposes only, this LAR includes proposed changes to the Technical Specification Bases required to maintain consistency with the revised Condition A described in the proposed Technical Specification above.

3.0 BACKGROUND

CPSES Technical Specification 3.7.3 for Feedwater Isolation Valves (FIVs) provides a 4 hour Completion Time for one or more FIVs inoperable. The NUREG-1431 Standard Technical Specification (STS) (Ref. 1) version of this specification extends that Completion Time to 72 hours. The STS Bases indicates that the 72 hour Completion Time for the FIVs takes into account the redundancy afforded by the feedwater control valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour FIV Completion Time is reasonable, based on operating experience. However, STS 3.7.3 also adds new requirements associated with the feedwater control valves (FCVs) and their bypass valves which were not part of our previous specifications and which are not consistent with the CPSES design and licensing basis. During the TS conversion, the STS version of TS 3.7.3 was evaluated to determine whether it should be adopted. The STS specification was not proposed by TXU Electric based on concerns that a deficiency in the isolation capability of the FCVs could potentially result in a unit shutdown when such a shutdown is not required and would not be necessary per the CPSES licensing basis. Such a shutdown would put the unit through a transient that would be less safe than continuing to operate. CPSES proposed an alternative specification which provided a 72 hour Completion Time contingent on the availability of the FCVs, however, that proposal was rejected by the NRC primarily because it was considered to be beyond the scope of the TS conversion.

Recently, enforcement discretion was sought and granted to extend the FIV Completion Time in order to repair a defective FIV hydraulic pump. Similar concerns associated with the hydraulics of the remaining FIVs now make it necessary to consider a change to this specification.

4.0 TECHNICAL ANALYSIS

The safety grade FIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Each FIV has a FIV Bypass Valve (FIBV) and a Feedwater Preheater Bypass Valve (FPBV) which are its associated bypass valves. The associated function of the Feedwater Control valves (FCVs) and their associated bypass valves (FCBVs) is to provide backup isolation of MFW flow to the secondary side of the steam generators following an HELB. The FCVs are not designated as active (i.e. are not full safety grade) but are designed as highly reliable backups to the FIVs. This licensing basis is reflected in FSAR Section 6.2.1. The NRC found this design basis to be generically acceptable for PWRs in NUREG-0138 (Ref 2). Closure of the FIVs and associated bypass valves or FCVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FLBs) occurring upstream of the FIVs or FCVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the FIVs will be mitigated by their closure. Closure of the FIVs and associated bypass valves, or FCVs and associated bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FLBs inside containment, and reducing the cooldown effects for SLBs.

The FIVs and associated bypass valves, and the main feedwater check valves, isolate the non-safety related portions from the safety related portions of the system. In the event of a feedwater pipe rupture in the non-safety portion of the system, the check valves will close to terminate the loss of fluid from the secondary side. In the event of a secondary side pipe rupture inside containment, the FIVs and associated bypass valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

The FIVs and associated bypass valves, and FCVs and associated bypass valves, close on receipt of a safety injection signal, T_{avg} — Low coincident with reactor trip (P-4) or steam generator water level — high high signal. They may also be actuated manually. Each FIV and associated bypass valves and each FCV and associated bypass valve is a two train valve (i.e., both Train A and Train B controls are independently provided to perform the close function). Single active failure of the FIV and associated bypass valves is not assumed; however, the FCVs and associated bypass valves are provided as a backup in the unlikely event a mechanical failure prevented the primary isolation valves from fully closing.

The GDC-4 design basis of the FIVs is established by the analyses for large SLB. It is also influenced by the accident analysis for the large FLB. Closure of the FIVs and associated bypass valves may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level — high high signal .

The current LCO ensures that the FIVs and their associated bypass valves will isolate MFW flow to the steam generators, following an FLB or main steam line break. The FCVs, while not credited to perform the Nuclear Safety Function for these events, are nevertheless expected to be available as non-safety grade backups to the FIVs. The availability of the FCVs and their bypass valves to perform the backup isolation function is assured by the existing Technical Requirements Manual (TRM) requirement TRM 13.7.40. The TRM surveillance testing is identical to that required by the STS version of the specification for these valves (i.e., on a refueling outage frequency, the TRM surveillance requires that each FCV and associated bypass valve initiate closure on the same actuation signals and with the same closure time as the FIVs). The TRM surveillances for the FCVs were satisfactorily performed during their last respective outages. Because, the TRM requirements provide the same level of assurance that FCVs and FCV bypass valves can perform their isolation function, a 24 hour Completion Time for one or more FIVs inoperable is warranted. In the event that the FCVs or FCV bypass valves cannot perform their isolation function, the current 4 hour Completion Time would be applicable.

The 24 hour Completion Time is reasonable, based on operating experience and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is also conservative with respect to the Completion Time allowed in Condition C of TS 3.6.3 , "Containment Isolation Valves", which allows 72 hours for one or more penetrations inoperable (applicable to penetration flow paths with only one containment isolation valve and a closed system).

In summary, the proposed increase in the Completion Time for one or more FIVs inoperable is justified based on the redundancy afforded by the FCVs to terminate MSLB and FLB events. The TRM surveillance requirements for the FCVs and associated bypass valves demonstrate their ability to initiate closure on the same actuation signals and with the same closure time requirements as the FIVs. Allowing 24 hours to correct a FIV hydraulic system problem based on credit for the FCV is consistent with the STS (which allows 72 hours) and could prevent an unnecessary plant shutdown transient or prevent a feedwater transient due to a less than adequate time allowed for a repair.

5.0 REGULATORY ANALYSIS

5.1 No significant Hazards Determination

TXU Electric has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10CFR50.92 as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change extends the Completion Time for one or more Feedwater Isolation Valves (FIVs) inoperable from 4 hours to 24 hours for repair of FIV hydraulic system, if the Feedwater Control Valve (FCV) and associated bypass valve in the same flowpath has been verified to be available to perform feedwater isolation. Extending the Completion Time is not an accident initiator and thus does not change the probability that an accident will occur. However, it could potentially affect the consequences of an accident if an accident occurred during the extended unavailability of the inoperable FIV. The increase in time that the FIV is unavailable is small and the probability of an event occurring during this time period which would require isolation of the MFW flow paths is low. Moreover, the redundancy provided by the FCVs, which have the same actuation signals and closure time requirements as the FIVs, provides adequate assurance that automatic feedwater isolation will occur if required.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Closure of the FIVs is required to mitigate the consequences of a Main Steam Line Break and Main Feedwater Line Break accidents. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The proposed change does not change any Technical Specification Limit or accident analysis assumption. Therefore it does not involve a reduction in a margin of safety.

Based on the above evaluations, TXU Electric concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10CFR50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 Regulatory Safety Analysis

Applicable Regulatory Requirements / Criteria

10CFR50, Appendix A, General Design Criteria (GDC) 4, "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

GDC 16, "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

GDC 50, "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

GDC 53, "The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows ."

GDC 54, "Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits ."

GDC 57, "Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve ."

U. S. NRC Regulatory Guide (RG) 1.22 is NRC guidance for ensuring the adequacy of protection system actuation functions through periodic testing.

The specification of concern helps assure compliance with GDC 4, GDC 16, GDC 50, GDC 53 and GDC 54, such that, in the event of a Main Feedwater Line break or Main Steam Line Break inside containment, the containment will be appropriately isolated and preventing additional mass and energy from being delivered to the steam generators.

Analysis

The proposed change extends the Completion Time for one or more Feedwater Isolation Valves (FIVs) from 4 hours to 24 hours for repair of FIV hydraulic system, if the Feedwater Control Valve (FCV) and associated bypass valve in the same flowpath have been verified to be available to perform feedwater isolation. This change does not change the compliance with any of the above General Design Criteria and is consistent with the basis under which the NRC approved STS allowed the 72 hour Completion Time (i.e., the availability of the FCVs to perform the isolation function). The change does not affect the commitment to Regulatory Guide 1.22 as documented in FSAR Section 1A(N).

Conclusion

The technical analysis performed by TXU Electric demonstrates the availability of the FCVs and their bypass valves to perform the backup isolation function. In the event that the FCVs or FCV bypass valves cannot perform their isolation function, the current 4 hour Completion Time would be applicable. The change assures that there is sufficient feedwater line isolation redundancy to support the accident analyses of Chapter 15 and thus continues to be compliant with the above regulatory requirements.

6.0 ENVIRONMENTAL EVALUATION

TXU Electric has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. TXU Electric has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required.

7.0. REFERENCES

1. NUREG1-1431, Revision 1, Standardized Technical Specifications for Westinghouse Plants.

2. USNRC Office of Nuclear Reactor Regulation, NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRC Staff", November 1976.

ATTACHMENT 3 to TXX-00110

MARKUP OF TECHNICAL SPECIFICATION PAGE

Pages 3.7-8 and 3.7-9

3.7 PLANT SYSTEMS

3.7.3 Feedwater Isolation Valves (FIVs) and Associated Bypass Valves

LCO 3.7.3 Four FIVs and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when FIV or associated bypass valve is closed and de-activated or isolated by a closed manual valve.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more FIVs inoperable.	A.1.1 Close or isolate FIV.	4 hours
	OR*	
	A.1.2.1 Verify that the Feedwater Control Valve and associated bypass valve in the same flowpath are available to perform feedwater isolation.	4 hours
	AND	
	A.1.2.2 Close or isolate FIV	24 hours
	AND	
	A.2 Verify FIV is closed or isolated.	Once per 7 days

(continued)

* Actions A.1.2.1 and A.1.2.2 are only allowed for repair of the FIV hydraulic system through the end of fuel cycle 8 for Unit 1 and fuel cycle 5 for Unit 2.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more FIV bypass valves inoperable.	B.1 Close or isolate bypass valve. AND B.2 Verify bypass valve is closed or isolated.	4 hours Once per 7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time for each FIV and associated bypass valve is ≤ 5 seconds	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify each FIV and associated bypass valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

ATTACHMENT 4 to TXX-00110

MARKUP OF TECHNICAL SPECIFICATION BASES PAGES
(For Information Only)

Pages B 3.7-18

BASES (continued)

APPLICABILITY The FIVs and the associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the FIVs and the associated bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function. In MODES 4, 5, and 6, steam generator energy is low. Therefore, the FIVs and the associated bypass valves are normally closed since MFW is not required.

ACTIONS The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one FIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 4 hours. When these valves are closed or isolated, they are performing their required safety function. Alternately, for the repair of the FIV hydraulic system only, if within 4 hours the FCV and associated FCV bypass valve in the same flowpath are verified to be capable of performing the isolation function, the closure/isolation of inoperable affected valves can be extended to 24 hours. If the FCV or associated FCV bypass valve in the same flowpath are not capable of performing the isolation function then the inoperable FIVs must be closed or isolated within 4 hours. The FCV and associated FCV bypass valve are considered to be capable of performing the isolation function when TRM SRs 13.7.40.1 and 13.7.40.2 have been performed within the required testing interval.

The 24 hour Completion Time takes into account the redundancy afforded by the FCV and associated bypass valve which are capable of performing the isolation function. The 24 hour and the 4 hour Completion Times takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. ~~The 4 hour~~ The Completion Times are ~~is~~ reasonable, based on operating experience.

Inoperable FIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. LCO 3.0.5 allows the FIVs to be opened as needed for post maintenance testing to demonstrate operability.

(continued)

ATTACHMENT 5 to TXX-00110

RETYPED TECHNICAL SPECIFICATION PAGES

Pages 3.7-8 and 3.7-9

3.7 PLANT SYSTEMS

3.7.3 Feedwater Isolation Valves (FIVs) and Associated Bypass Valves

LCO 3.7.3 Four FIVs and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when FIV or associated bypass valve is closed and de-activated or isolated by a closed manual valve.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more FIVs inoperable.	A.1.1 Close or isolate FIV.	4 hours
	OR	
	A.1.2.1 Verify that the Feedwater Control Valve and associated bypass valve in the same flowpath are available to perform feedwater isolation.	4 hours
	AND	
	A.1.2.2 Close or isolate FIV	24 hours
	AND	
	A.2 Verify FIV is closed or isolated.	Once per 7 days

(continued)

* Actions A.1.2.1 and A.1.2.2 are only allowed for repair of the FIV hydraulic system through the end of fuel cycle 8 for Unit 1 and fuel cycle 5 for Unit 2.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more FIV bypass valves inoperable.	B.1 Close or isolate bypass valve. AND B.2 Verify bypass valve is closed or isolated.	4 hours Once per 7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the isolation time for each FIV and associated bypass valve is ≤ 5 seconds	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify each FIV and associated bypass valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

ATTACHMENT 6 to TXX-00110

RETYPED TECHNICAL SPECIFICATION BASES PAGES
(For Information Only)

Pages B 3.7-18

BASES (continued)

APPLICABILITY The FIVs and the associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the FIVs and the associated bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function. In MODES 4, 5, and 6, steam generator energy is low. Therefore, the FIVs and the associated bypass valves are normally closed since MFW is not required.

ACTIONS The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one FIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 4 hours. When these valves are closed or isolated, they are performing their required safety function. Alternately, for the repair of FIV hydraulic system only, if within 4 hours the FCV and associated FCV bypass valve in the same flowpath are verified to be capable of performing the isolation function, the closure/isolation of inoperable affected valves can be extended to 24 hours. If the FCV or associated FCV bypass valve in the same flowpath are not capable of performing the isolation function then the inoperable FIVs must be closed or isolated within 4 hours. The FCV and associated FCV bypass valve are considered to be capable of performing the isolation function when TRM SRs 13.7.40.1 and 13.7.40.2 have been performed within the required testing interval.

The 24 hour Completion Time takes into account the redundancy afforded by the FCV and associated bypass valve which are capable of performing the isolation function. The 24 hour and the 4 hour Completion Times takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The Completion Times are reasonable, based on operating experience.

Inoperable FIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. LCO 3.0.5 allows the FIVs to be opened as needed for post maintenance testing to demonstrate operability.

(continued)

ATTACHMENT 7 to TXX-00110

REFERENCE INFORMATION (For Information Only)

TR 13.7.40 Feedwater Control Valves (FCVs) and Associated Bypass Valves

Pages 13.7-30 and 13.7-31

Excerpt from NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRC Staff", November 1976. (Issue No. 1)

Pages 1-1 thru 1-12

13.7 PLANT SYSTEMS

TR 13.7.40 Feedwater Control Valves (FCVs) and Associated Bypass Valves

TR LCO 13.7.40 Four FCVs and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when FCV or associated bypass valve is closed and de-activated or isolated by a closed manual valve.

ACTIONS

----------NOTE----------
Separate Condition entry allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more FCVs inoperable.	A.1 Restore affected FCV(s) to OPERABLE. OR A.2 Perform an assessment per the corrective action program to allow continued operation beyond 7 days.	7 days 7 days
B. One or more FCV bypass valve(s) inoperable.	B.1 Restore affected FCV(s) to OPERABLE. OR B.2 Perform an assessment per the corrective action program to allow continued operation beyond 7 days.	7 days 7 days

SURVEILLANCE REQUIREMENTS

----------NOTE----------

These surveillance requirements may be satisfied by an engineering evaluation following packing adjustment.

SURVEILLANCE	FREQUENCY
TRS 13.7.40.1 Verify the isolation time of each FCV and associated bypass valve is ≤ 5 seconds.	18 months
TRS 13.7.40.2 Verify each FCV and associated bypass valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

Issue No. 1

Treatment of Non-Safety Grade Equipment in Evaluations of Postulated Steam Line Break Accidents

This issue was identified in the NRC Inspector and Auditor's report of July 1976 and in a meeting of the Electrical, Instrumentation and Control Systems Branch held on September 10, 1976. In the attachment to the November 3, 1976 memorandum from the Director, NRR to the NRR staff it was listed as Issue #1 and defined as follows:

> "In evaluating the consequences of postulated breaks of steam lines the current staff position (SRP 10.3) states that the design should preclude the blowdown of more than one steam generator, assuming a concurrent single component failure, and assuming that the turbine stop and control valves remain functional. Provided that these valves and their control systems are designed for closure under the postulated conditions, and because they are high quality components, the staff does not require that they be designed to the requirements for safety-related equipment."

A meeting of all members of the Electrical, Instrumentation and Control Systems Branch was held on November 12, 1976 to discuss, clarify and redefine this issue as necessary in order to aid in developing a staff response. As a result, the issue was redefined by one or more

concerned members of the Branch as follows:

> "Treatment of Non-Safety Grade Equipment in Evaluation of Postulated Steam Line Break Accident. The steam line break accident (inside or outside containment) is a design basis accident and either the consequences are demonstrated to be acceptable or the equipment provided for mitigating the consequences shall be designed to safety criteria. The statement that the turbine stop valves can be used to protect against steam line break is inadequate. GDC-2 requires components and systems important to safety to be designed to withstand seismic events. Part 100 Appendix A requires components necessary to assure the capability to prevent or mitigate accidents to remain functional in the event of the SSE (See Appendix A, III (c))."

SUMMARY OF ISSUE

In evaluating a postulated steam line break accident, the staff assumes that certain "non-safety grade" valves operate when needed to limit both the resultant blowdown to a single steam generator and the consequences of the postulated accident. Of particular concern is that these "non-safety grade" valves are not designed to function during or following the design basis earthquake (the Safe Shutdown Earthquake as defined in Appendix A to 10 CFR Part 100), but may be required to mitigate the consequences of an assumed steamline break accident.

SUMMARY RESPONSE

The staff requires that only seismically qualified, safety grade equipment be assumed to function in mitigating the consequences of a safe shutdown earthquake (SSE) in accordance with General Design Criterion 2 and 10 CFR 100, Appendix A.

For loss-of-coolant accidents (LOCA) involving a spontaneous rupture of the primary system boundary, where significant damage to the fuel and a major release of fission products are potential consequences, the most stringent quality and design requirements, including seismic qualification, are imposed on those systems needed to prevent and cope with a LOCA. However, for accidents involving spontaneous failures of secondary system piping not part of the primary system boundary, where the potential consequences are significantly lower, less stringent requirements are imposed on the quality and design of the systems needed to cope with such secondary system ruptures. This approach results, in the staff's judgment, in a proper weighing of consequences and safety requirements in order to assure a balanced level of safety over the entire spectrum of postulated design basis accidents.

Detailed Discussion

The NRC has historically imposed the most stringent quality and design requirements on those systems needed to cope with a loss-of-coolant accident (LOCA) where significant damage to the fuel and a major release of fission products from the primary system boundary are potential consequences. One of the stringent requirements associated with a LOCA is that the pressure boundary of the primary system must be designed to withstand all operational and accident loads, including SSE loads, and all safety equipment required to function to mitigate the consequences of a LOCA must also be designed to withstand the SSE.

With regard to secondary system piping, the NRC has not required that steam piping downstream of the main steam line isolation valves (MSIV's) be designed to withstand earthquake loads. The consequences of a rupture of this piping as a result of an earthquake are limited by the operation of equipment which is designed to withstand an SSE assuming a single active component failure; e.g., the MSIV's, the piping upstream of the MSIV's, Reactor Trip System, the Safety Injection System, and the Auxiliary Feedwater System.

Where the design basis event under consideration is the SSE itself, only Category I equipment and systems are credited in the safety evaluation,

and the single active failure criterion is also applied. In this manner it is assured that all essential features necessary to shutdown the plant are functional, if an earthquake should occur, considering an additional random failure of any Category I component.

Consistent with the lesser safety importance of the secondary system boundary, the staff does not require that an earthquake be assumed to occur coincident with a postulated spontaneous break of the steamline piping; i.e., loss of equipment not designed to withstand a SSE is not assumed coincident with an assumed spontaneous steamline break accident.

If such an instantaneous major steamline break were assumed to occur, a blowdown of some fraction of the secondary system water inventory would result. In evaluating this accident, the staff considers the effects on the core, on primary system components, on safety-related components in the vicinity of the steamlines, and on the containment structure. In addition, the staff evaluates the radiological consequences associated with such a postulated break.

In the event of such a major steamline break, the initial rapid depressurization of one or more steam generators would result in cooling and depressurizing the primary system. An engineered safety features actuation signal (ESFAS) generated by this event would initiate the isolation of all the steam generators in addition to other safety actions, including initiation of safety injection. Depending on the break location and the single active failure assumed in evaluating this event, there could be continued cool-down of the primary system because of the blowdown of a non-isolable steam generator.

The staff's evaluation of the consequences of a major steamline break, either inside or outside of the containment, is based on the continuous rapid depressurization of one steam generator and the isolation of the main feedwater system when necessary. As part of this evaluation, it is assumed that a single active failure occurs in the systems required to mitigate the consequences of such events. The availability of offsite power may or may not be assumed, depending on which assumption is more severe; and it is further assumed that the highest worth control rod fails to scram.

Analyses performed of such an accident predict that the reactor may return to criticality because of the rapid cooling of the primary coolant. Calculated radiological consequences for this assumed accident are dominated by iodine released via primary coolant

leakage into the depressurized steam generator and subsequently through the assumed break to the atmosphere outside of containment. The iodine release is a strong function of the additional fuel failures that might result from the accident. No fuel failures are usually predicted. Appropriate technical specification limits are established on primary and secondary system activity and primary-to-secondary systems leak rates to limit potential consequences to a small fraction of 10 CFR Part 100 for the case of a steamline break with a single active component failure.

The steamline and main feedwater lines, connected to the steam generators, each have a number of valves located along their length. The valve closest to the steam generator on each steam line is a safety grade component and is referred to as the main steam isolation valve (MSIV). For the purposes of this discussion, a safety grade component is defined as one which is designed to seismic category I (Regulatory Guide 1.29), quality group C or better (Regulatory Guide 1.26), and is operated by electrical instruments and controls that meet IEEE-279. The remaining valves in the steam and main feedwater lines are designated as non-safety grade components because they may not meet all the above criteria.

One postulated steamline break accident that is pertinent to this issue is the rupture of a main steamline inside containment resulting in the blowdown of the affected steam generator. The radiological consequences may be increased, as implied by this issue, if an additional failure occurs and if no credit is taken for "non-safety grade" valves functioning following this assumed event, which would then result in a rapid depressurization of a second steam generator. Specifically, the following accident scenario is one that has been suggested by this issue as more appropriate than those currently evaluated by the staff.

(1) A rupture occurs upstream of the MSIV in one of the main steam supply lines.

(2) A safety grade MSIV associated with one of the remaining steam generators fails to close on demand.

(3) In addition, the non-safety grade valves, such as turbine stop valves and turbine control valves upstream of the turbine, or the turbine bypass valves fail to close on demand, providing a path for blowdown of a second steam generator.

The probability of blowing down more than one steam generator as a result of the accident scenario described above is quite low. A probabilistic assessment is useful in placing a proper perspective on the role of non-safety grade equipment in evaluating a postulated steamline break inside containment. Data indicate that the likelihood of complete rupture of a seismic Category 1 main steamline is about 10^{-3} per reactor year.*

Continued reliability of these components over the life of the plant is assured by frequency (generally weekly) in-service tests. The staff has made a survey of the reliability of the tubine stop, control, and intercept valves in operating LWR's. The findings include the following:

1) there have been no control system failures;
2) there have been a few incidents in which one control or stop valve did not fully close (all these occurred during in-service testing);
3) based on the fact that closure of either the turbine stop or control valve (which are in series) will achieve the required isolation, the reliability of these valves is of the same order of magnitude as that accepted for nuclear safety-grade components.

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* All of the probability values presented in this discussion should be treated as approximate; refined analyses are being made by the staff.

An approximate probabilistic assessment for the above (steamline break of seismic Category I piping inside containment) scenario yields a probability of less than 10^{-7} per reactor year. This low probability does not even include the failure rate for a control rod failing to scram. Thus, even if some of these probability estimates were off by one or several orders of magnitude, the overall probability of this postulated event would remain small.

This particular scenario is not analyzed by the staff, because the staff permits reliance on the downstream steamline valves to prevent the blowdown of a second steam generator in the unlikely event that the first two steps of the scenario should actually occur. Reliance on these non-safety grade valves in the postulated accident evaluation is permitted based on the reliability of these valves.

For all these postulated scenarios, a steam line break of the type envisioned would have a negligible contribution to the overall risk, relative to other possible accident scenarios having a greater or equal likelihood of occurrence. The staff therefore concludes that the scenario suggested in the issue need not be considered as a design basis accident.

In the event of a steamline break inside containment, it is necessary to isolate the main feedwater to the steam generator associated with the failed line to preclude overpressurizing the containment and to limit the reactivity transient. If the single active failure postulated for this accident is the failure of the appropriate safety grade main feedwater isolation valve to function, then credit is taken for closing the nonsafety grade main feedwater control valve or tripping the feedwater pump in that line. The rationale for reliance on these "non-safety grade" feedwater components is similar to that presented above for the steamline valves.

Thus, the staff believes that it is acceptable to rely on these non-safety grade components in the steam and feedwater systems because their design and performance are compatible with the accident conditions for which they are called upon to function. It is the staff position that utilization of these components as a <u>backup</u> to a single failure in safety grade components adequately protects the health and safety of the public.

General Design Criterion I of 10 CFR 50, Appendix A, states that:

> "Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." (Emphasis added).

This design criterion expressly permits flexibility in the acceptance level for safety related equipment. The staff has imposed the most stringent requirements on those systems needed to cope with a loss of coolant accident, where significant damage to the fuel and primary system is assumed to occur. The potential consequences from a steamline break accident are judged not to be as severe; therefore, less stringent quality standards for these "non-safety grade" valves are appropriate.

The use of non-safety grade valves to mitigate the consequences of an assumed steamline break accident has been the subject of staff discussion since 1975, and was one of the issues raised earlier this year. As indicated in the document, "Report to the Director, Office of Nuclear Reactor Regulation, Concerning R. Pollard's Allegations," dated February 28, 1976, the staff is re-evaluating our present position on this matter. Because of the low probability of occurrence of the series of events that would lead to a significant increase in the consequences of a steamline break, this re-evaluation is not considered a high priority item in our generic technical activities.