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August 14, 2003

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
SUPPLEMENTAL INFORMATION RELATED TO
LICENSE AMENDMENT REQUEST (LAR) 02-10,
REVISION TO TECHNICAL SPECIFICATION (TS) 3.6.3,
CONTAINMENT ISOLATION VALVES
(TAC NOS. MB8185 and MB8186)

REF: 1) TXU Energy Letter, logged TXX-03040, from C. L. Terry to U. S. Nuclear Regulatory Commission dated March 18, 2003

Gentlemen:

In Reference 1, TXU Generating Company LP (TXU Energy) submitted proposed changes to the Technical Specificications (TS) associated with containment isolation valves (LAR 02-10). The proposed amendment will delete two of the Surveillance Requirements (SR) in TS 3.6.3 entitled "Containment Isolation Valves." Specifically, safety injection valves 8809A, 8809B, and 8840 and containment spray valves HV-4776, HV-4777, CT-142, and CT-145 will no longer be leak tested.

Based on conversations with the NRC staff, TXU Energy provides the following information regarding LAR 02-10. This supplemental information will be discussed at the public meeting scheduled for August 19, 2003.

A review of the previous four outages at CPSES was performed to determine the total personrem of exposure and critical path time expended during testing of the containment isolation valves listed above. Estimated total exposure averaged 0.675 personrem/outage. Testing of 8809A and 8809B are critical path activities due to the system configuration required for testing. Average critical path time averaged 12 hours/outage. The projected personrem and critical path time for 2RF07 is currently 1.2 personrem and 24 hours, respectively. This is based on the current scope of work for 2-8809A and 2-8809B.

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The as-found Local Leak Rate Tests (LLRTs) for 8809A and 8809B require Mode 5 conditions with the RHR train to be tested inoperable, isolated, and at ambient temperature. During testing, RCS temperature is 170-180 degrees F and RCS pressure is greater than 100 psig. Due to requirements for ambient conditions, test duration per valve is approximately six hours. When these as-found tests are performed at the beginning of an outage (2-3 days after shutdown), the RCS heatup rate would be in excess of 20 degrees F/minute with a loss of the operable RHR train of shutdown cooling. This places the unit in a condition of increased risk.

The deletion of SR 3.6.3.12 and 3.6.3.13 is acceptable because the penetrations are not a potential containment atmosphere leakage path during and following a Design Basis Accident (DBA). Clarification on the closed system outside containment, comparison with similar Westinghouse 4-loop plants, and changes in CPSES design and maintenance since original licensing, is attached.

This communication contains no new licensing basis commitments.

Should you have any questions, please contact Mr. Jack Hicks at (254) 897-6725.

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

Executed on August 14, 2003.

Sincerely,

TXU Generation Company LP
By: TXU Generation Management Company LLC,
Its General Partner

C. L. Terry
Senior Vice President and Principal Nuclear Officer

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

JCH/jch

Attachment

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ATTACHMENT to TXX-03143

TECHNICAL ANALYSIS

Section I: Summary of Design Basis and Safety Analysis Consideration

TXX-03040 dated March 18, 2003, provided a detailed design basis and safety analysis of containment isolation relative to the valves subject to SR 3.6.3.12 and 3.6.3.13. Failure modes and effects analysis (FMEA) were provided for each penetration. In addition, failure modes and effects analysis were provided for each closed system outside containment. The closed systems outside containment are credited in the original licensing basis for providing the secondary containment isolation boundary when there are no active failures. In this case, the system is pressurized by pumps connected to the containment emergency sump. However, the results of these FMEAs also show that the closed system outside containment is credited in the original licensing basis for providing the secondary containment isolation boundary when there is a single active failure of Train A or Train B power when the pressure in the closed system is not in excess of containment design. In this case, the containment boundary is provided by a closed check valve inside containment and the closed system outside containment. SR 3.6.3.12 and 3.6.3.13 have no effect on this failure mode and effect. The closed systems outside containment are designed and tested to meet NUREG-0800 SRP 6.2.4 and NUREG-0737 Section III.D.1 to satisfy this function. An expanded description of the Radioactive System Leakage Inspection (RSLI) Program is provided in a section below.

The FMEAs show that the only failure in which the valves subject to SR 3.6.3.12 and 3.6.3.13 function as containment isolation valves, they are providing the third containment barrier. The first barrier is the closed check valve inside containment, the second is the closed system outside containment, and the third is the motor operated valves at the containment penetration. The failure which results in this configuration is the active failure of an RHR or Containment Spray Pump which is a low probability event. The probability of this failure for each pump coincident with a large break LOCA is on the order of 1.0E10-8 per year.

The deletion of SR 3.6.3.12 and 3.6.3.13 is acceptable because the penetrations are not a potential containment atmosphere leakage path during and after a LOCA. The maintenance of a water seal for 30 days is assured by a closed system outside containment which is not drained or vented post-LOCA. This secondary boundary would be available even in the event of any single active failure. Furthermore, even after the worst single active failure, there are a minimum of three containment isolation barriers between the containment atmosphere and the outside atmosphere when these valves are closed after a LOCA as opposed to a single barrier assumed for Appendix J leak rate testing. Barriers in series have been previously accepted as the basis for exceptions to Type C testing. For example, local vent and drain connections are locked closed and capped providing an effective labyrinth seal as the basis for not requiring Type C LLRT. The pressure drop across each barrier is reduced or eliminated by each subsequent barrier. In addition, "closed systems" inside reactor containment (with requirements similar to the closed systems outside containment) have a single active valve outside containment, and these valves are not subject to Type C testing.

The pressure boundary components and piping design ratings are in excess of the 50 psig containment design pressure. Design pressure varies from 325 to 2485 psig. Because containment pressure peaks at 48 psig for only a few seconds and is required to be reduced below 5 psig within 24 hours post-LOCA, post-accident containment pressures do not present a significant challenge to the pressure boundary and leakage limiting components.

Leakage in the closed system outside containment is minimized by the RSLI and maintenance programs. Operating experience and work history demonstrates that the closed system is a suitable barrier and ensures the secondary containment isolation function is maintained. Detailed descriptions of the RSLI and maintenance programs and their operating experience and work history is provided in sections below.

Therefore, the surveillance required by Technical Specifications SR 3.6.3.12 and SR 3.6.3.13 is not commensurate with the design and licensing basis. These surveillance requirements are not required to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. These valves are not part of the primary success path for containment isolation and do not meet 10CFR50.36 *Criterion 3* for 10CFR50 Appendix J requirements. These components are not part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. In addition, these valves do not meet 10CFR50.36 *Criterion 4.* Neither operating experience nor probabilistic risk assessment has shown LLRT of these valves to be significant to public health and safety.

Removal of the surveillance required by Technical Specifications SR 3.6.3.12 and SR 3.6.3.13 meets the requirements of 10CFR50.92.

Section II: The Radioactive System Leakage Inspection (RSLI) Program

The overall objective of the RSLI program is to monitor and reduce leakage from those portions of systems outside containment that contain highly radioactive fluids during post accident operation to as low as reasonably achievable levels. Leakage from radioactive systems outside containment are monitored to meet the commitments in CPSES FSAR Section III.D.1.1 (CPSES Response to the NRC Action Plan for the TMI Accident) and the requirements of Section 5.5.2, Primary Coolant Sources Outside Containment of the Technical Specifications.

The RSLI program includes the following:

a. Preventive maintenance and periodic visual inspection requirements; and
b. Integrated leak test requirements for each system at refueling cycle intervals or less.

The leakage criteria for the RSLI Program are as follows: The limiting leakage value based on a cumulative amount from all liquid systems tested under the RSLI Program is 1.0 gpm per unit. An additional criterion for liquid leakage on individual systems is administratively applied.

Total leakage greater than 1.0 gpm is the established fluid system leakage criteria. The 1.0 gpm is based on accident analysis assumptions for radiological consequences of engineered safety features equipment leakage outside containment [See CPSES FSAR Section 15.6.5.4]. All abnormal leakage is evaluated and corrected under the Appendix B corrective action program in accordance with NRC Generic Letter 91-18, Revision 1.

Each RSLI system is inspected at intervals not to exceed each refueling cycle. Testing is performed at normal system operating pressures. In order to have appropriate portions of systems pressurized, inspection of the Containment Spray, Residual Heat Removal, and Safety Injection systems are scheduled to coincide with the operability tests of those systems, when possible.

The test method for the RSLI inspection tests (as found in ETP (Equipment Test Procedure) procedures) specifies that floor drains are to be observed in every space for the presence of liquid and boron. Additionally, the procedure specifies inspection of the system for leaks at packing leakoffs, flanges, fittings, valves, pumps, etc. With the inspection process checking for leaks at the floor drains where the stem leakoff piping is directed, and checking flanges, fitting, etc., while the system is pressurized, significant leakage would definitely be detected during the test.

For example, for the Unit 2 Train A RHR system, the inspection walkdown list directs attention to the following valves/pipe penetrations: 2-8856A, 2SI-1008, 2SI-1003, 2SI-0001, 2-8809A, 2SI-0007, 2-8842, 2-8840, 2-HV-4178, and 2RH-0018. With the exception of instrument (temperature) fittings these valves are the only penetrations to the RHR line above the top of the ECCS sump.

During leak testing of the RHR system the system pressure is > 200 psig therefore any leakage during an accident (at a lower pressure) would be considerably less. Historical leakage for this test has been very low, less than 0.1 gpm. Currently the leakage for both units/trains of RHR is 0 gpm.

For example, for the Unit 2 Train A CT system inspection, the walkdown list directs attention to the following valves/pipe penetrations: 2-HV-4776, 2CT-0051, 2CT-0053. These valves are the only penetrations to the CT line above the top of the ECCS sump.

During leak testing of the CT system the system pressure is > 150 psig therefore any leakage during an accident (at a lower pressure) would be considerably less. Historical leakage for this test has been very low, less than 0.1 gpm. Currently the only noted CT system leakage is for unit 2 train B and is 0.002 gpm.

Based on the present maximum observed values of 0.002 gpm and the procedures used to collect that data, it can reasonably be shown that stem leakoff of the 8809, 8840 and CT system isolation valves is inconsequential with relation to containment leakage.

Operations, Engineering, and Maintenance personnel perform tests, walkdowns and inspections on a frequent basis and identify/quantify leakage and initiate corrective actions (e.g. work orders, SmartForms) as necessary.

System engineering reviews RSLI test data and other significant leakage data and applicable corrective action documents on RSLI system components to maintain a RSLI Program Leakage Table for each unit. This will ensure that the unit's cumulative leakage for portions of systems covered by this program remain within the leakage criteria.

Maintenance personnel implement corrective actions as soon as reasonably possible on leakage identified by RSLI tests or other inspections. These corrective actions include adjusting packing or replacement of seals, gaskets, o-rings, etc., on RSLI system components.

The RSLI program ensures that the closed systems outside containment provide an acceptable secondary containment boundary during and after a LOCA.

Section III: Comparison with Similar Westinghouse 4 Loop PWRS

Safety Injection Valves 8809A, 8809B, and 8840

In order to benchmark testing practices at similar designs, Westinghouse engineers reviewed the flow diagrams for the following eight plants focused on the configurations from the discharge of the SI and RHR pumps to the cold and hot legs.

- The following plants are the same as CPSES : Vogtle, Byron, Braidwood, Callaway, Wolf Creek, and Catawba
- Seabrook is the same except they have two parallel RHR hot leg isolation valves 8804A and B (as opposed to a single valve 8840).
- Diablo Canyon is the same except their RHR hot leg isolation valve 8703 (equivalent to 8840) is inside containment.

For all of these plants, Westinghouse has confirmed that design and operation require the cold leg injection paths to be open for cold leg injection but closed for switchover to cold leg recirculation specifically to prevent runout of one pump should the other have failed to start. Switchover to cold leg recirculation aligns the RHR pump discharge to supply the suction of all high head and intermediate head ECCS pumps as well as continuing the cold leg injection.

Based on the similarity in design, it was determined that the only significant differences between these plants and CPSES would be the relative elevation of the subject valves and the containment emergency sump. Plants were contacted to obtain a comparison to CPSES for the ECCS systems with the following plants responding: Wolf Creek, Callaway, McGuire, Vogtle, and Diablo Canyon. These plants were found to be essentially the same as CPSES. The isolation valves in the penetrations in the discharge of the SI and RHR pumps to the cold and hot legs are above the emergency sump. For all portions of the system below the sump, the sump can provide a virtually unlimited supply of makeup for any anticipated leakage.

CPSES has confirmed that at each of these eight plants, a) they are not required to Type C LLRT the penetrations in the discharge of the SI and RHR pumps to the cold and hot legs, b) their licensing basis is simply the closed system outside containment, and c) this was the original licensing basis at the time the operating license was received.

No Westinghouse 4-loop PWR contacted by CPSES indicated that they do Type C LLRTs on their comparable valves. Therefore, CPSES was unable to identify any comparable plant which does Type C test these valves for a comparison of design and licensing basis.

Therefore, it can be concluded that there is substantial precedence for the request in this LAR that the closed system outside containment be approved as the basis for the elimination of Type C testing.

Containment Spray Valves HV-4776, HV-4777, CT-142 and CT-145

No plant design comparison was made for the containment spray system since this design varies significantly from plant to plant. However, it was found that the following plants do not Type C test the corresponding containment isolation valves based on the closed system outside containment: Wolf Creek, Callway, and McGuire/Catawba. The following plants do Type C test the corresponding containment isolation valves: Vogtle, Byron/Braidwood, and Diablo Canyon. This is a clarification to Section 3.0 of Attachment 1 to Reference 1 where we stated that "The Byron/Braidwood Stations, Diablo Canyon Station, and Wolf Creek/Callaway Stations do not perform LLRTs for these RHR and Containment Spray Valves." The CPSES Maximum Allowable Leakage Rage is 4734 cc/hr based on a large water seal, required for the spray system design function independent of containment isolation. This large volume provides a substantial barrier between the containment atmosphere and the environment and does not need to be verified by test because of the additional barrier of the closed system outside containment.

Section IV: Changes in CPSES Design and Maintenance Since Original Licensing

There have been significant changes to the plant design and maintenance since original licensing which directly affect this request. In addition, there is significant maintenance history which is relevant to the subject isolation valves.

During the original licensing process, an internal technical issue was raised about the potential for leaking from the valve stem packing. Briefly, the issue was related to the gate valve design. With post-accident containment pressure being applied on the inboard side and failure to maintain pumped pressure on the outboard side, the disk would experience seating force on the outboard seat of the valve and a reduction of seating force on the containment side of the valve. This issue did not affect the validity of the closed system outside containment; however, it could result in containment pressure in the bonnet of the isolation valve. At that time, our valves had traditional packing with leak off connections. Packing leaks were not uncommon. The resolution of this issue was the commitment to Type C test these valves and a request by the applicant to include testing with water as an option in Technical Specifications in accordance with the NUREG-0800 Standard Review Plan 6.2.6. These were conservative decisions and resolved the technical concern.

Changes to the plant design and maintenance process implemented since original licensing include:

- Issuance of Design Specification CPES-M-1070, "Alternate Valve Stem Packing Replacement"
- Revisions to Maintenance Procedures MSM-CO-8803, "Borg-Warner Bolted Bonnet Gate Valves" and MSM-CO-8824, "Westinghouse Gate Valves"
- Formation of a maintenance department "Valve Team"
- Design Improvements.

CPSES has implemented an Alternate Valve Stem Packing Replacement Program. This program provides the Engineering direction for the Maintenance department to improve valve packing performance by use of alternate packing configurations and materials in safety and non-safety applications. The program is based on EPRI NP-5697, "Valve Stem Packing Improvements" and is consistent with current valve Original Equipment Manufacturer (OEM) methodologies for stem sealing. It should be noted the original valve packing provided with the valves did not use these new methodologies currently specified. In order to reduce/eliminate packing problems, the program implements the recommended corrective actions from the EPRI report to:

- Reduce the packing stack height to decrease in-service consolidation and maintain gland load more effectively
- Employ improved stem packing configuration to transmit axial gland loads to radial sealing forces more effectively
- Provide a means of continual gland adjustment through live-loading to maintain sufficient pressure on the gland follower for effective long-term sealing.

Implementation of the alternate valve stem packing replacement program has in practice revealed improved stem sealing performance. A large reduction in packing leaks was realized through

implementation of this program. The reduction realized in operational leakage provides a high degree of confidence that the valves stem sealing improvements will remain post accident.

Valve work and test histories for the 10 valves have been reviewed to determine their condition and historical performance. No packing leaks have ever been identified on nine of the ten valves; however, reports of potential leaks were investigated and no leakage found. The only exception was an insignificant leak. Valve 1-8809B experienced an insignificant leak in 1998 which was corrected. The work order reflected that the only impact of the leakage was minor contamination of the valve around the packing area. The packing was repaired under the new program and no leakage has been identified since. Nine of the ten valves have had the original packing replaced under the new program. The only valve that has not been re-worked with new packing, 1-8840, has never had an identified packing leak.

The results of the work order history review is summarized as follows:

1-8809A	Packing replaced in the fall of 1993. None of the work orders performed any valve internal work and no packing leakage has ever been identified.
1-8809B	Packing replaced in 1992. Minor packing leak corrected in April of 1998. Work orders indicated no other valve internal work or packing leakage has been identified.
1-8840	Review of work order history does not indicate packing upgrade has been performed. None of the work orders performed any valve internal work and no packing leakage has ever been identified.
1-HV-4776	Valve internals were re-worked in 1999 and alternate packing installed. Review of work orders did not indicate any packing leakage work orders requiring corrective action.
1-HV-4777	Packing replaced in 1991. No packing leakage has been identified following installation. Body to bonnet leakage was identified with work orders performing bonnet fastener tightening and valve internal corrective actions.
2-8809A	Packing replaced in 1992. Review of work order history did not indicate any valve internal work or packing leakage has ever been identified.
2-8809B	Packing replaced in Feb. of 1993. Review of work order history did indicate seat leakage corrective work orders. No packing leak corrective work orders were implemented.
2-8840	Packing data sheet indicates packing replacement performed during start-up activities in 1991. Review of work history did not indicate any packing leakage or valve seat leakage work orders.
2-HV-4776	Corrective work order performed valve internal work in 1992 which installed the alternate packing material. No further packing problems identified in work history.
2-HV-4777	Corrective work orders performed valve internal work in 1992 which installed the alternate packing material. No further packing problems identified in work history.

Engineering and the Maintenance department determined following initial startup that valve maintenance practices required additional information to effectively receive the desired valve performance. Early 1990's work processes involved valve disassembly, identification of items to be corrected, contacting the OEM for repair guidance, and processing documentation to allow rework to continue. This method impacted the re-assembly process prompting Engineering and Maintenance enhance the process. The improved process considered a variety of valve related maintenance issues such as body to bonnet surface defect repair, disc and seat sealing surface repair, stem scoring, etc. The OEM Engineering departments were requested to provide the desired information prior to valve disassembly. The goal of the effort was to obtain the necessary maintenance information to provide effective maintenance on service induced defects without impacting the valve design function. Guidance provided by the OEM have been added to the installation and maintenance instruction manuals at CPSES, and the associated maintenance procedures. The following procedures applicable to the subject valves have incorporated the OEM information which includes all critical sealing functional areas on the valves.

MSM-CO-8824, "Westinghouse Gate Valves" was upgraded to maximize the valve maintenance process. This procedure applies to the 8809A, 8809B and 8840 valves.
MSM-CO-8803, "Borg-Warner Bolted Bonnet Gate Valve Maintenance" was upgraded to maximize the valve maintenance process. This procedure applies to HV-4776 and HV-4777 valves.
The CPSES Valve Team was formed by the maintenance organization to provide a specialized group focusing only on valve maintenance. The team is a multi-discipline organization dedicated only to valves and actuators and include mechanics, electricians, and I&C technicians. This group receives specialized training on valve repair techniques from a variety of sources. Component specialists (for gate, globe, check, and relief valves, MOV's and AOV's) are provided by the Maintenance Support Group at CPSES to support the Valve Team and monitor valve component health.

The Valve Team methodology uses a philosophy of returning the valve to a "like new" condition. The detailed maintenance activities involves close dimensional verifications and inspection activities in order to ensure the work performed restores the design sealing functions. For example, the as-found inspection per MSM-C0-8803 contain 15 inspection sections (bonnet, body, wedge, stem runout, bolting, etc) with 36 required inspections. MSM-CO-8824 contains similar level of detailed as-found inspection containing 19 inspection sections containing 44 required inspections. Dimensional information obtained from the OEM has been incorporated into their procedures providing for easy correction of problems identified during valve rework.

Contract personnel brought in during outages to perform maintenance are qualified valve technicians using the maintenance department procedures contained in the work orders. Valve Team members supervise the contract personnel during outages. Critical steps in the repair process (such as blue check of the disc to seat interface) are inspected by Valve Team personnel to ensure the desired level of quality is realized.

Prior to start-up, 2-HV-4776 and 2-HV-4777 experienced excessive seat leakage which upon disassembly identified body guide wear/damage due to contact with the wedge. This condition is one of the disadvantages of the flexible wedge gate design if the body guide clearances are wide enough to allow the mating disc guide to hit the inside surface and bind. Excessive clearances in this location can also cause excessive wear on the valve seating surfaces. This condition was corrected on Unit 2 prior to initial start-up. On Unit 1, this same condition was identified during the performance of a corrective work order for body to bonnet leakage. Conversations with the Flowserve Engineering provided Engineering input to tighten the guide clearances to limit the movement of the wedge. The design changes have been implemented on both Unit 1 valves as a part of corrective actions. The design change will be implemented on the Unit 2 valves the next time valve maintenance is performed on them.

EPRI testing for motor operated valves revealed sharp edges on the disc, seat and disc guide could during the valve closure stroke result in a less than smooth transition into the seat. In order to ensure a smoother and more predictable operation, the OEM was requested to review their design, and recommend radius or chamfer instructions per the guidance in EPRI-TR-103237. Flowserve review of these surfaces revealed the seat ring I.D. per their design was consistent with EPRI guidance. The radius on the disc O.D. and the chamfer on the disc guide bottom edge were recommended to be increased or machined to a specified dimension. Both Unit 1 valves were inspected and machined for these design features. Unit 2 valves will have this enhancement installed during their next maintenance activity.