

performed during 1RE10 [Refueling Outage 10 for Unit 1], the first refueling outage following SG [steam generator] replacement, exceeded the TS requirements for the first two refueling outages after replacement combined. That is, more tubes were inspected than were required by TS. Currently, South Texas Project Unit 1 does not have an active SG damage mechanism and will meet the current industry examination guidelines without performing inspections during the next refueling outage. The results of the Condition Monitoring Assessment after 1RE10 demonstrated that all performance criteria were met during 1RE10. The results of the 1RE10 Operational Assessment show that all performance criteria will be met over the proposed operating period.

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

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

*Response:* No.

The proposed change does not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during 1RE10, the first refueling outage following SG replacement, significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement combined.

The proposed change does not affect the design of the SGs, the method of operation, or reactor coolant chemistry controls. No new equipment is being introduced and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection interval, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

*Response:* No.

Steam generator tube integrity is a function of design, environment, and current physical condition. Extending the SG tube inservice inspection frequency [interval] by four months does not alter the function or design of the SGs. Inspections conducted prior to placing the SGs into service (preservice inspections) and inspection during the first refueling outage following SG replacement demonstrate that the SGs do not have fabrication damage or an active damage mechanism. The scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for the same model of RSGs [replacement steam generators] installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the

replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

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*NRC Section Chief:* Robert A. Gramm.

**TXU Generation Company LP, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas**

*Date of amendment request:* September 23, 2003.

*Brief description of amendments:* The proposed change would revise Technical Specification (TS) 3.6.3 entitled, "Containment Isolation Valves," to extend the frequency of Surveillance Requirement 3.6.3.7 for containment and hydrogen purge valves and containment pressure relief valves with resilient seats.

*Basis for proposed no significant hazards consideration determination:* As required by Title 10 of the Code of Federal Regulations (10 CFR), Section 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

*Response:* No.

Operability and leakage control effectiveness of the containment purge, hydrogen purge and containment pressure relief system isolation valves have no effect on whether or not an accident occurs. Consequently, increasing the interval between surveillances of isolation valve leakrate does not involve a significant increase in the probability of an accident previously evaluated. The consequences of a non-isolated reactor containment building at the time of a fuel-handling accident or LOCA [loss-of-coolant accident] is release of radionuclides to the environment. Analyses have conservatively assumed that a containment pressure relief system line is open at the time of an accident, and release to the environment continues until the isolation valves are closed. In addition, LOCA analyses assume containment leakage of 0.1% of the containment volume per day for the first 24 hours and 0.05% per day for the duration of the accident. Consequently,

increasing the interval between surveillances of isolation valve leakrate does not involve a significant increase in the 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.

The proposed changes do not involve a modification to the physical configuration of the plant (*i.e.*, no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. The functions of the containment purge, hydrogen purge and containment pressure relief systems are not altered by this change. Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

*Response:* No.

This proposed change only increases the interval between surveillance tests of the containment purge, hydrogen purge and containment pressure relief system valves. Analyses have conservatively assumed that the containment purge valves are open at the time of a fuel handling accident, and that the containment pressure relief valve is open at the time of a loss-of-coolant accident. In addition, LOCA analyses assume containment leakage of 0.1% of the containment volume per day for the first 24 hours and 0.05% per day for the duration of the accident. The radiological consequences of both an fuel handling accident and a LOCA are unchanged and remain within the 10 CFR 100 limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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*NRC Section Chief:* Robert A. Gramm.

**Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas**

*Date of amendment request:* October 17, 2003.

*Description of amendment request:* The licensee is proposing to revise Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program," for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The proposed revision to