

Operator Actions, including Response Times," provided guidance to the industry concerning use of operator actions in place of automated system or component actuation. IN 97-78 states: In those instances where licensees consider temporary or permanent changes to the facility which credit operator actions, the NRC has relied on the guidance provided in \* \* \* ANSI/ANS 58.8, "Time Response Design Criteria for Safety-related Operator Actions," \* \* \* for evaluating such changes. The American Nuclear Society (ANS)-58.8, establishes the requirements for safety-related operator actions, which are summarized as follows: (1) The specific operator actions required, (2) the potentially harsh or inhospitable environmental conditions expected, (3) ingress/egress paths taken by the operators to accomplish functions, (4) procedural guidance for required actions, (5) operator training and qualifications to carry out actions, (6) any additional support personnel and/or equipment to carry out actions, (7) information required by the control room staff to determine whether action is required, including qualified instrumentation to diagnose the situation and to verify that the action is successfully, (8) ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery, and (9) consideration of risk significance of operator actions.

The manual actions implemented by this change can be completed within the guidance and criteria provided in IN 97-78 and ANS-58.8. Consequently, the manual actions can be credited in the mitigation of the specific accidents. With credit for the manual actions to isolate the affected lines subsequent to an accident inside containment, the type of accidents and consequences currently evaluated in the UFSAR, remains the same. Therefore, the proposed change does not create the possibility of new or different kinds of accidents from any accident previously evaluated.

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

*Response:* No.

This change establishes requirements for manual actions to isolate one air line and three water lines subsequent to an accident inside containment concurrent with a single failure of a containment isolation valve to close. The manual actions ensure air or water cannot continue to enter containment with a single failure of an outboard containment isolation valve when the line pressure boundary inside containment is lost due to an accident and associated pipe interactions. The safety-related configuration of the lines (outboard motor operated valve and inboard check valve) continues to ensure the containment environment is automatically prevented from exiting the line to outside the containment. Safety-related instrumentation is available to inform operators that the

manual actions are required, and operators have been trained in the requirements for addressing the failures of valves to close. In addition, adequate time and resources are available to perform the manual actions. The manual actions meet the criteria for safety-related operator actions contained in NRC IN 97-78 and ANS-58.8. Further, the proposed change to allow credit for the manual actions does not affect the offsite and Main Control Room dose consequences of the accidents currently reported in UFSAR Chapter 15, Accident Analyses. 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:* Allen G. Howe.

*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:* March 6, 2003.

*Brief description of amendments:* Technical Specifications Section 1.1 "Definitions" for Engineered Safety Feature (ESF) Response Time and Reactor Trip System (RTS) Response Time require U.S. Nuclear Regulatory Commission (NRC) review and approval of any methodology used to allocate response times in lieu of measuring them. The application requests NRC review and approval of a topical report to allow the use of allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 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.

The proposed change does not result in a condition where the design, material, and construction standards that were applicable

prior to the change are altered. The same RTS and ESFAS [Engineered Safety Feature Actuation System] instrumentation are being used and the time response allocations and modeling assumptions in the Chapter 15 safety analysis are unchanged. Only the method of verifying the time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade, or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR [Final Safety Analysis Report]. Therefore, the proposed change does 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.

The proposed change does not alter the performance of the process protection racks, the nuclear instrumentation, or the logic systems used in the plant protection systems. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time characteristics to exceed the total allowance. Changing the method of periodically verifying instrument response for these systems from response time testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time characteristic to exceed the total allowance. The total time response allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Therefore, 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 affect the total system response time assumed in the safety analysis. The periodic response time verification method for the Process protection racks, the nuclear instrumentation and the logic systems is modified to allow the use of actual test data or engineering data. The method of verification still provides assurance that the total system response time is within that defined in the safety analysis, since calibration tests will continue to be performed and may be used to detect any degradation which might cause the response time to exceed the total allowance. The total response time allowance for each function bounds all degradation that cannot be detected by