

of coolant accident]. The Containment Spray System is capable of performing its function effectively with the single failure of any active component in the system, any of its subsystems, or any of its support systems. A plugged nozzle would have negligible impact on the capability of the Containment Spray System to respond to a Loss of Coolant Accident.

Therefore, the consequences of an accident previously evaluated are not significantly affected by the proposed change.

2. Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change will not physically alter the plant (no new or different type of equipment will be installed) or change the methods governing normal plant operation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The system is not susceptible to corrosion-induced obstruction or obstruction from sources external to the system. Maintenance activities that could introduce foreign material into the system would require subsequent verification to ensure there is no nozzle blockage. The spray header nozzles are expected to remain unblocked and available in the event that the safety function is required. Therefore, the capacity of the system would remain unaffected. Hence, this 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.

STP Nuclear Operating Company, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 22, 2003.

Description of amendment request: The proposed amendments revise Technical Specification (TS) 3.4.2.2, "Reactor Coolant System," to relax the lift setting tolerance of the pressurizer safety valves from ± 2 percent to ± 2 percent, -3 percent.

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. Does the proposed change involve a significant increase in the probability or

consequences of an accident previously evaluated?

Response: No.

The proposed TS change takes credit for the assumptions made in the reanalysis of the rod withdrawal from power event already evaluated in the UFSAR [Updated 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. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed TS change takes credit for the assumptions made in the reanalysis of the rod withdrawal from [the] power event already evaluated in the UFSAR. Therefore, the 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.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed TS change takes credit for the assumptions made in the reanalysis of the rod withdrawal from power event already evaluated in the UFSAR. That analysis demonstrated that the fuel design limits were maintained by the reactor protection system since the DNBR [Departure from Nucleate Boiling Ratio] was maintained above the limit value. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment involves no significant hazards consideration under the standards set forth in 10 CFR 50.92 and, accordingly, a finding of "no significant hazards consideration" is justified.

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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Tennessee Valley Authority (TVA), Docket No. 50-328, Sequoyah Nuclear Plant (SQN), Unit 2, Hamilton County, Tennessee

Date of amendment request: June 5, 2003 (TSC 03-08).

Description of amendment request: The proposed amendment would revise the reactor coolant system (RCS) heatup and cooldown curves (pressure-temperature (P-T) limits). The revision

replaces the P-T limits that are currently analyzed for 14.5 Effective Full Power Years (EFPYs) with new limits analyzed for 32 EFPYs. In addition, the amendment includes corresponding changes to the Technical Specification (TS) figure associated with the Low Temperature Over Pressure Protection and the TS Bases.

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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed revision does not affect plant equipment, test methods or operating practices. The modification to SQN TSs is consistent 10 CFR 50, Appendix G in conjunction with alternative methods provided in American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1." The proposed change continues to provide controls for safe operation within the required limits. The proposed changes do not contribute to events or assumptions associated with postulated design basis accidents (DBA). The proposed revisions continue to maintain the required safety functions. Accordingly, the probability of an accident or the consequences of an accident previously evaluated is not increased.

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

No. The proposed revision is not the result of changes to plant equipment, test methods, or operating practices. The proposed revision to the SQN Unit 2 P-T limits continues to ensure that conservative fracture toughness margins are maintained to protect against reactor pressure vessel failure. In addition, SQN's current setpoints for low-temperature overpressure protection were evaluated and are bounding for the proposed 32 EFPY P-T limits. The updated P-T limits are based on NRC approved methodology in conjunction with alternative methods provided in American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Requirement Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1."

The reactor vessel P-T limits are operational limits and are not considered to be contributors to the generation of postulated accidents. The safety functions of the associated systems remain unchanged and do not affect the assumptions of DBAs. The operational limits continue to be governed within the TSs. Accordingly, the proposed change does not create the possibility of a new or different kind of accident.

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