

No. TVA's proposed TS amendment provides revised reactor pressure vessel P-T limits that are within the design capabilities of the pressure control systems for protection of the RCS. The limits are based on conservative design margins that ensure that plant operation is within the design capacity of the reactor vessel materials. Accordingly, the function of the RCS to provide a fission product barrier is not compromised.

TVA's proposed change to revise P-T limits does not result in a change to system design features. The proposed change does not affect plant conditions that result in precursors to accidents or cause degradation of accident mitigation systems. The plant system safety functions are not altered by the proposed change.

The proposed changes allow plant operation with different P-T limits while continuing to retain conservative margins for assuring integrity of the reactor vessel and the RCS. Consequently, the proposed TS revisions do not significantly reduce the 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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Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: June 5, 2003 (TSC 03-09).

Description of amendment request: The proposed change to the Updated Final Safety Analysis Report (UFSAR) would amend the design and licensing basis to identify that operator action may be necessary to ensure containment design pressure is not exceeded subsequent to a high energy line break (HELB) such as loss-of-coolant-accident.

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 procedure changes/additions being implemented to mitigate a SCSA [station control and service air] leak in containment will only be used following a HELB in containment, a consequential rupture of an SCSA line and a failure of the

outboard CIV [containment isolation valve] on the SCSA containment supply.

Operators isolate the SCSA leak on the accident unit by either manually closing a valve upstream of the stuck-open CIV or by shutting down the station air compressors. If the station air compressors are shut down prior to performing an emergency shut down of the non-accident unit or if an operator error results in an isolation of the control air supply to the non-accident unit, then at-worst, a UFSAR Condition II event is induced on the non-accident unit. For example a reactor trip from full power or a loss of normal feedwater—loss of control air to the feedwater regulator valves resulting in a loss of normal feedwater to the non-accident unit.

A UFSAR Condition II event has a frequency of one per year. Therefore, the proposed procedure changes/additions, including the potential for operator error do not result in more than a minimal increase in a previously evaluated Condition II event ($1+1/40 = 1.025$ less than 10 percent increase).

The operator actions being implemented to mitigate a SCSA leak in containment are performed after the occurrence of an accident on primarily non-safety-related systems, structures or components [SSCs] so they do not increase the likelihood of the occurrence of a malfunction of equipment previously evaluated in the UFSAR.

The air operated containment isolation valve is assumed to fail open due to single failure criteria and, containment isolation/integrity is maintained by the inboard check valve. The containment boundary is unaffected by the operator actions being implemented to mitigate a SCSA leak in containment. Therefore, the consequences of all accidents previously evaluated in the UFSAR remain unchanged.

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

No. This change implements new manual actions for accident failure modes not previously evaluated in the UFSAR. The manual actions are required to ensure containment design pressure is not exceeded.

Operators isolate the SCSA leak on the accident unit by either manually closing an upstream isolation valve on the accident unit's SCSA containment supply or by shutting down the station air compressors. The operator actions being implemented have been determined to meet the criteria for safety-related operator actions in NRC Information Notice (IN) 97-78/ANS-58.8 and; therefore, there are no credible operator actions which would prevent isolation of a SCSA leak prior to containment design pressure being exceeded. If the station air compressors are shut down prior to performing an emergency shutdown of the non-accident unit or if an operator error results in an isolation of the SCSA supply to the non-accident unit, then at-worst, a UFSAR Condition II event occurs. Because UFSAR Condition II events have been previously identified, the operator actions being added under this change do not create the possibility of an accident of a different type than previously evaluated.

The operator actions being implemented to mitigate a SCSA leak in containment are performed after the occurrence of an accident on primarily non-safety-related SSCs so they do not create a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the UFSAR.

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

No. The established limits for the fuel, reactor vessel or containment are not affected by the addition of operator actions to isolate a SCSA leak inside containment. Isolation of the air leak within two hours of a large break loss-of-coolant accident (LBLOCA) prevents containment pressure exceeding the peak calculated pressure. Consequently, this change does not represent a reduction in the 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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Tennessee Valley Authority (TVA), Docket No. 50-390, Watts Bar Nuclear (WBN) Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: May 14, 2003.

Description of amendment request: The proposed amendment would allow an alternate Westinghouse methodology for the measurement of reactor coolant system (RCS) total flow rate via measurement of the RCS elbow tap differential pressures. TVA stated that this methodology is similar to that reviewed and approved by the NRC for other utilities.

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. TVA's evaluation for WBN Unit 1 determined that the probability of an accident will not increase since adequate RCS flow will still be assured. Sufficient margin exists to account for all reasonable instrument uncertainties; therefore, no changes to installed equipment or hardware in the plant are required, thus the probability of an accident occurring remains unchanged. The initial conditions for all accident