

significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated February 19, 2003.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI-2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI-2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to

offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI-2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI-2 accident can be adequately met without reliance on a PASS.

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

The NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

Tennessee Valley Authority, Docket No. 50-390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: December 13, 2003.

Description of amendment request: The proposed amendment would allow

the use of Westinghouse leak-limiting Alloy 800 sleeves to repair defective steam generator tubes as an alternative to plugging the tube.

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 in accordance with the three standards set forth in 10 CFR 50.92(c), which are presented below:

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

No. The Westinghouse Alloy 800 leak-limiting repair sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applied stresses and fatigue usage for the repair sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by NRC's Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Burst testing of sleeve/tube assemblies has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The Alloy 800 repair sleeve depth-based structural limit is determined using the NRC guidance and the pressure stress equation of ASME Code, Section III with additional margin added to account for configuration of long axial cracks. A bounding detection threshold value has been conservatively identified and statistically established to account for growth and determine the repair sleeve/tube assembly plugging limit. A sleeved tube is plugged on detection of degradation in the sleeve/tube assembly.

Evaluation of the repaired steam generator tube testing and analysis indicates no detrimental effects on the sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Watts Bar Unit 1. Corrosion testing and historical performance of sleeve/tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the sleeve/tube assembly is bounded by the current steam generator tube rupture (SGTR) analysis described in Watts Bar Unit 1 Updated Final Safety Analysis Report. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates