

applicable only to valves that are not isolable from the primary containment free air space.

*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. The proposed change does not involve a significant increase in probability or consequences of an accident previously evaluated.

The proposed change will revise LaSalle County Station, Units 1 and 2, Technical Specification (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program" by identifying a specific exception to the testing guidance contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program."

The function of the primary containment is to isolate and contain fission products released from the reactor Primary Coolant System (PCS) following a design basis Loss of Coolant Accident (LOCA) and to confine the postulated release of radioactive material to within limits. The probability of an accident previously evaluated is not dependent on the test-frequency of the primary containment Type A, B or C testing. The test interval associated with primary containment testing is not a precursor of any accident previously evaluated. The proposed specific exception to the testing guidance contained in RG 1.163 will continue to test all potential valve atmospheric leakage paths and will not be a precursor to a Design Basis Accident (DBA). Containment testing does provide assurance that the LaSalle County Station primary containments will not exceed allowable leakage rate values specified in the Technical Specifications and will continue to perform their design function following an accident.

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

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

3. The proposed change does not involve a significant reduction in a margin of safety.

The integrity of the primary containment is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify

the essentially leak tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a specific exception to the testing guidance contained in Regulatory Guide (RG) 1.163 will continue to test all potential valve atmospheric leakage paths and does not effect the test acceptance criteria for Type A, B or C testing. Therefore, LSCS has determined that the proposed change provides an equivalent level of protection as that currently provided.

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 requested amendments involve no significant hazards consideration.

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*NRC Section Chief:* Anthony J. Mendiola.

*Nine Mile Point Nuclear Station, LLC, Docket No. 50-410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York*

*Date of amendment request:* November 20, 2003.

*Description of amendment request:* The licensee proposes to revise the safety limit minimum critical power ratio (SLMCPR) values in section 2.1.1.2 of the Technical Specifications (TSs). The SLMCPR values are based on cycle-specific calculations done for the next fuel cycle, Cycle 10, using methodology previously approved by the Nuclear Regulatory Commission (NRC).

*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. The NRC staff has reviewed the licensee's analysis against the three standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed SLMCPR values, calculated using an NRC-approved methodology, will be made in a manner such that conservatism is maintained through compliance with applicable NRC regulations and guidance. No hardware design change is involved with the proposed amendment, thus there will be

no adverse effect on the functional performance of any plant structure, system, or component (SSC). All SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the consequences of postulated accidents. SLMCPR values were not previously factored into the probability of accidents, nor were they factored into scenarios of previously analyzed accidents. Accordingly, the revised SLMCPR values will lead to no increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods the unit is operated. As a result, all SSCs will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the licensee did not propose to exceed or alter a design basis or safety limit, the proposed amendment will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the NRC staff's analysis, 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:* Richard J. Laufer.

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

*Date of amendment request:* November 21, 2003.

*Description of amendment request:* The proposed amendment would allow the position of a rod to be monitored by