

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.

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

*NRC Section Chief:* John A. Nakoski.

*Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama*

*Date of amendment request:* March 31, 2003.

*Description of amendment request:* The proposed amendments would modify Surveillance Requirement (SR) 3.4.11.1, for Farley, Unit 2 only by the addition of the following note that states, "Not required to be performed for Unit 2 for the remainder of operating cycle 16 for Q2B31MOV8000B." In addition, a temporary Technical Specification SR 3.4.11.4 is added to provide compensatory action for this block valve while SR 3.4.11.1 is suspended. Further, this SR requires that power to the Farley, Unit 2 Power Operated Relief Valve Q2B31MOV8000B be checked at least every 24 hours for the remainder of operating cycle 16.

*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 changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to Surveillance Requirement (SR) 3.4.11.1 suspends the requirement to cycle test the Unit Two pressurizer power operated relief valve (PORV) block valve Q2B31MOV8000B for the remainder of operating cycle 16. This change will eliminate the remaining scheduled cycle tests for the PORV block valve during operating cycle 16. SR 3.4.11.4 is added to provide compensatory measures for verifying power available to the block valve at least every 24 hours. At the end of cycle 16, the proposed changes will no longer be in effect. Suspension of the cycle tests for the PORV block valve Q2B31MOV8000B may result in a small decrease in assurance that the block valve would cycle if required to isolate a stuck open PORV. However, experience with these valves has shown them to be very reliable and suspension of the remaining tests will not appreciably reduce reliability of the valve. There is no relationship between packing leakage on the PORV block valve and a postulated stuck open PORV. The proposed compensatory measure of verifying block

valve power available on a 24 hour basis adds additional assurance that the block valve will close if demanded. Therefore, the probability of a previously evaluated accident remains acceptable is not significantly increased.

The proposed changes do not affect the consequences of a previously analyzed accident since the magnitude and duration of analyzed events are not impacted by this change. The dose consequences of the proposed change are bounded by LOCA [loss-of-coolant accident] analyses. Therefore, the consequences of a previously evaluated accident are unchanged.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes involve no change to the physical plant. They allow for suspension of the PORV block valve Q2B31MOV8000B cycle tests for a limited time and provide for compensatory action to verify power to the PORV block valve. This valve provides an isolation function for a postulated stuck open or leaking pressurizer PORV. This condition is an analyzed event since it is bounded by the FNP [Farley Nuclear Plant] LOCA analyses. In addition to the isolation function, the block valve is required to remain open to allow the associated PORV to function automatically to control reactor coolant system (RCS) pressure. These changes do not impact the open function of the block valve since the normal position is open.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The physical plant is unaffected by these changes. The proposed changes do not impact accident offsite dose, containment pressure or temperature, emergency core cooling system (ECCS) or reactor protection system (RPS) settings or any other parameter that could affect a margin of safety. The elimination of cycle testing of the PORV block valve Q2B31MOV8000B for the remainder of the Unit Two operating cycle and the addition of the proposed compensatory action that enhances assurance of valve operation are somewhat offsetting.

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.

*Attorney for licensee:* M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

*NRC Section Chief:* John A. Nakoski.

*Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia*

*Date of amendment request:* February 26, 2003.

*Description of amendment request:* The proposed amendments would

revise Technical Specifications Section 5.5.17, "Containment Leakage Rate Testing Program," to reflect a one time deferral of the Type-A Containment Integrated Leak Rate Test (ILRT). The 10-year interval between ILRTs is to be extended to 15 years from the previous ILRTs that were completed in March 2002 for Unit 1 and March 1995 for Unit 2.

*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 Technical Specifications change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to Technical Specifications 5.5.17, "Containment Leakage Rate Testing Program," involves a one-time extension to the current interval for Type A containment leak testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specifications change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG-1493 ["Performance-Based Containment Leak-Test Program"], that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. VEGP [Vogtle Electric Generating Plant] test history supports this conclusion. NUREG-1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanism which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural