

generator tube integrity commensurate with the requirements of R.G. 1.121.

R.G. 1.121 describes a method acceptable to the NRC staff for meeting GDCs [General Design Criteria] 2, 14, 15, 31, and 32 by reducing the probability of the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of outside diameter stress corrosion cracking at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The most limiting effect would be a possible increase in leakage during a steam line break event. Excessive leakage during a steam line break event, however, is precluded by verifying that, once the criteria are applied, the expected end of cycle distribution of crack indications at the tube support plate elevations would result in minimal, and acceptable primary to secondary leakage during the event and, hence, help to demonstrate radiological conditions are less than an appropriate fraction of the 10 CFR 100 guideline.

The margin to burst for the tubes using the voltage-based repair criteria is comparable to that currently provided by existing technical specifications.

In addressing the combined effects of LOCA [loss-of-coolant accident] + SSE [safe shutdown earthquake] on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to either the LOCA rarefaction wave and/or SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential the partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse or that short through-wall indications would leak at significantly

higher leak rates than included in the leak rate assessments.

Consequently, a detailed leak-before-break analysis was performed and it was concluded that the leak-before-break methodology (as permitted by GDC 4) is applicable to the Farley reactor coolant system primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed at Farley and were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Farley, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Based on analyses' results, no tubes near wedge locations are expected to collapse or deform to the degree that secondary to primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a LOCA + SSE is expected to be less than that previously allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to tighten under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing the R.G. 1.83 considerations, implementation of the tube repair criteria is supplemented by 100% inspection requirements at the tube support plate elevations having outside diameter stress corrosion cracking indications, reduced operating leakage limits, eddy current inspection guidelines to provide consistency in voltage normalization, and rotating pancake coil inspection requirements for the larger indications left in service to characterize the principle degradation mechanism as outside diameter stress corrosion cracking.

As noted previously, implementation of the tube support plate elevation repair criteria will decrease the number of tubes that must be taken out of service with tube plugs or repaired. The installation of steam generator tube plugs or tube sleeves would reduce the RCS flow margin, thus implementation

of the voltage-based repair criteria will maintain the margin of flow that would otherwise be reduced through increased tube plugging or sleeving.

Considering the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any bases of the plant Technical Specifications.

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.

*Local Public Document Room location:* Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

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

*NRC Project Director:* William H. Bateman.

*Southern Nuclear Operating Company, Inc., Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama*

*Date of amendments request:* January 9, 1995.

*Description of amendments request:* The requested changes to the Technical Specifications (TS) would implement the recommended changes from Generic Letter 93-05, "Line Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation." Specifically, the amendments would implement TS changes corresponding to the following GL 93-05 line-item improvement issues: Control Rod Movement Test for Pressurized Water Reactors, Radiation Monitors, Surveillance of Boron Concentration in the Accumulator/Safety Injection/Core Flood Tank, Containment Spray System, Hydrogen Recombiner, and Special Test Exemptions.

*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 changes do not involve any change to the configuration or method