

standards and applicable safety criteria limits are met. The proposed amendment would increase the time during which the equipment hatch and personnel air locks could be open during core alterations and movement of irradiated fuel. The proposed amendment does not involve changes in the operations of these containment penetrations. Having these penetrations open does not create the possibility of a new accident.

Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The assumptions and input used in the analysis are conservative as noted below. The design basis FHA has been defined to identify conservative conditions. The source term and radioactivity releases have been calculated pursuant to RG 1.183, Appendix B and with conservative assumptions concerning prior reactor operations. The control room atmospheric dispersion factor has been calculated with conservative assumptions associated with the release. The conservative assumptions and input noted above ensure that the radiation doses cited in the amendment request are the upper bound to radiological consequences of a FHA either in containment or in the spent fuel pool. The analysis shows that there is a significant margin between the TEDE radiation doses calculated for the postulated FHA using the AST and acceptance limits of 10 CFR 50.67 and RG 1.183. The proposed changes will not degrade the plant protective boundaries, will not cause a release of fission products to the public, and will not degrade the performance of any Structures, Systems, and Components important to safety. Therefore, there is no significant reduction in the margin of safety as a result of the proposed changes.

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

*Date of amendment request:* February 11, 2003.

*Description of amendment request:* The proposed amendments would allow a 40-month inspection interval for Farley, Unit 2 after the completion of the first post-replacement in-service

inspection, rather than the completion of two consecutive inspections resulting in a classification of C-1.

*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?

The proposed one-time change revises the steam generator (SG) inspection interval requirements in TS [technical specification] 5.5.9.3, "Inspection Frequencies," for the FNP [Farley Nuclear Plant] Unit 2 Spring 2004 refueling outage, to allow a 40[-]month inspection frequency after one inspection, rather than after two consecutive inspections with results that are within the C-1 category. C-1 category is defined as "less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective."

The proposed one-time extension of the FNP Unit 2 SG tube inservice inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the FNP's Final Safety Analysis Report (FSAR). As such, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would not go undetected. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last FNP Unit 2 refueling outage (*i.e.*, the first refueling outage following SG replacement) exceeded the TS requirements for the first two refueling outages after SG replacement. That is, more tubes were inspected than were required by TS. Currently, FNP Unit 2 does not have a SG damage mechanism, and will meet the current industry examination guidelines without performing SG inspections during the next refueling outage. Additionally, as part of the FNP SG Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the FNP Unit 2 Fall 2002 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Westinghouse replacement SGs, extending the SG tube inspection frequency does not

involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change revises the SG inspection frequency requirements in TS 5.5.9.3.a for the FNP Unit 2 Spring 2004 refueling outage, to allow a 40[-]month inspection interval after one inspection, rather than after two consecutive inspections, with inspection results within the C-1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during the last FNP Unit 2 refueling outage (*i.e.*, the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary-to-secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant systems or components.

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

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SGs is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded the scope required by the TS. These inspection results were comparable to similar inspection results for second generation alloy 690 models of replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

The NRC staff has reviewed the licensee's analysis and, based on this