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January 4, 1984

Docket Nos. 50-277
50-278

Mr. John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Accident Analysis Assumptions Regarding Reactor Building Leakage for Peach Bottom Atomic Power Station

Dear Mr. Stolz:

This letter formally responds to an issue recently identified by the Resident Inspector involving the Technical Specification acceptance criteria for reactor building leakage and the FSAR Accident Analysis assumptions.

The Peach Bottom Technical Specifications in Section 4.7.C.1 (page 176) specifies an acceptance limit of 10,500 cfm for reactor building leakage. An accident analysis presented in the Peach Bottom FSAR for both a LOCA and refueling accident (1983 Updated FSAR pages 14-6-23, 14-6-24, 14-6-30, 14-6-31) assumes a leakage of one reactor building air change per day. This leakage is equivalent to approximately 2,000 cfm.

The Resident Inspector expressed the concern that the FSAR did not evaluate off-site doses based on reactor building leakage up to the allowable level in the Technical Specification (10,500 cfm). The higher leakage results in a higher flow rate through the Standby Gas Treatment System (SGTS). Consequently, fission product holdup time is reduced and radioactivity releases are increased.

While the accident analysis in Section 14.6 referred to above is based on a leakage rate that is less conservative than the Technical Specification, there is another accident analysis in the FSAR that is more conservative than the Technical Specifications. This analysis is presented in Section 14.9.2

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of the FSAR. One of the assumptions in this analysis is that escaping fission products immediately flow through the SGTS and the stack without mixing in the secondary containment building. This assumption is equivalent to assuming a zero holdup time for the fission products, which results in a higher release rate to the environs. Therefore, the analysis in Section 14.9.2 results in calculated off-site dose levels that are higher than the dose level that would be obtained if the analysis assumed the allowable reactor building leakage in the Technical Specifications (10,500 cfm).

The accident analysis presented in Section 14.9.2 utilizes the assumptions presented in the AEC criteria (TID-14844), including source terms that are up to 600 times more conservative than the Section 14.6 analyses. The use of the AEC criteria to calculate the design basis accident doses shows that even with these conservative assumptions, the results, as shown in Table 14.9.7 of the FSAR, are still well within the limits of 10CFR100.

Accordingly, in assessing the potential consequences of reactor building leakage on off-site doses, the analysis presented in Section 14.9.2 should be considered in lieu of the analysis in Section 14.6, based on its upper bounding conservatisms and conformance to the AEC criteria. As previously stated, the results of the Section 14.9.2 analysis are well within 10CFR100 limits. For this reason, we conclude that the impact of allowable reactor building leakage on off-site doses is conservatively addressed in the FSAR accident analysis, and that further analysis is, therefore, not necessary.

Should you have any questions or require additional information regarding this matter, please do not hesitate to contact us.

Very truly yours,

cc: A. R. Blough, Site Inspector
R. W. Starostecki, Region I