REQUEST FOR ADDITIONAL INFORMATION

POINT BEACH NUCLEAR PLANT

EXTENDED POWER UPRATE (LAR 261)

POSTULATED STEAM GENERATOR TUBE RUPTURE EVENT

Background

The postulated steam generator tube rupture (SGTR) event, as analyzed for the Point Beach Nuclear Plant (PBNP) extended power uprate (EPU) request, is based on a conservative evaluation of the total reactor coolant mass released, through the steam generator, to the environment. The analyses supporting this evaluation rely on, among others, an assumption that operators can terminate the leakage of reactor coolant into the steam generator shell side, and ultimately, into the environment.

The validity of this assumption relies, in part, on a supplemental analysis that demonstrates that there is adequate margin to steam generator overfill. Although not a part of the PBNP licensing basis, this analysis attempts to validate the mass release evaluation by demonstrating, using a separate set of initial conditions, that following an SGTR, primary to secondary break flow can be terminated and the steam generator can be isolated before it fills with liquid water.

This supplemental analysis, however, is based on assumptions that are non-bounding of permissible plant operation, do not consider uncertainties, and do not include a limiting single active component failure in the mitigating safety system. The analyses also demonstrate a very small available margin to steam generator overfill.

Generic studies documented in WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," indicate that final liquid volume in the steam generator shell side will be greater, by about 12-13 percent when employing conservative input values. The NRC staff accepted the technical basis underlying WCAP-10698-P-A, since it described an acceptable methodology for performing a conservative margin-to-overfill analysis for a postulated SGTR.

Adding 12-13 percent to the Point Beach margin-to-overfill analysis results would predict overfill of the ruptured steam generator.

As documented in WCAP-11002-P[1], the postulated consequences of a steam generator overfill could include water relief through a safety valve, which then fails due to liquid flow. The valve could either fail to reseat, or fail fully open. In either case, the ruptured steam generator would be unisolable and would continue to discharge effluent from the ruptured steam generator for a period of time that far exceeds the release time assumed in the licensing basis mass release analyses for PBNP EPU. The continued discharge would also impede efforts to depressurize the reactor coolant system to a pressure below the steam generator shell side pressure

[1] Note that the staff discussed WCAP-11002-P in its evaluation of WCAP-10698-P-A, but did not find that it provided an acceptable method for performing a licensing basis safety analysis.

Consideration of input parameter variability, uncertainties, and limiting single failures, demonstrate that the postulated SGTR event at Point Beach may result in a steam generator overfill. This information challenges the assumption, employed in the mass release analyses, that the ruptured steam generator is isolable.

Applicable Regulatory Guidance

Title 10 of the *Code of Federal Regulations*, Part 50, Section 36 (10 CFR 50.36) promulgates requirements for facility technical specifications.

10 CFR 50.36 contains requirements for limiting safety system settings (LSSS), which are settings for automatic protective devices related to those variables having significant safety functions. Where an LSSS is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Certain LSSS, including those for steam generator water level, provide a basis for and/or establish an allowable range of initial conditions from which postulated accidents are assumed to initiate, and at which automatic equipment actuations are assumed to occur.

10 CFR 50.36 also requires the establishment of limiting conditions for operation (LCOs) of a nuclear reactor, which include, among other things, process variables, design features, and operating restrictions that are initial conditions of design basis accident or transient analyses that either assume the failure of or present a challenge to the integrity of a fission product barrier.

The Point Beach design basis, as described in PBNP Final Safety Analysis Report, Section 1.3, states that each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Since the radiological consequences of a postulated SGTR at PBNP are evaluated using an alternative source term, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," is applicable. Regulatory Position 5.1.3 states that the numeric values that are chosen as inputs to the analyses should be selected with the objective of determining a conservative postulated dose.

Issue

Based on the NRC staff's independent assessment, consideration of the full range of permissible plant operation, consistent with the above regulatory and licensing basis requirements, and consideration of parametric uncertainties, for a postulated steam generator tube rupture would result in a predicted overfill of the steam generator.

In order for the NRC staff to accept the proposed mass release analysis, the licensee must provide an acceptably conservative evaluation of margin to steam generator overfill to validate the assumption that the steam generator will not fill with water prior to the termination of atmospheric release.

Request

Demonstrate that the radiological consequences predicted using input from the licensing basis SGTR mass release analysis are characteristic of the limiting SGTR scenario. Provide appropriate consideration of the possibility and consequences of a water-filled steam generator with a ruptured tube, and ensure that supporting thermal-hydraulic analyses are consistent with the above regulatory and licensing basis requirements, including a plant-specific evaluation of the limiting single failure.

If it is necessary to revise the mass release analysis, address the effects of the revised analysis on (1) any downstream safety analyses and (2) any affected licensing actions currently under NRC staff review.