SUPPLEMENTAL SAFETY EVALUATION BY
THE OFFICE OF NUCLEAR REGULATION
RELATED TO OPERATION OF
FORT ST. VRAIN NUCLEAR GENERATING STATION
PUBLIC SERVICE COMPANY OF COLORADO
DOCKET NO. 50-267

Post-Accident Sampling System (NUREG-0737), Item II.B.3)

I. Introduction

In our safety evaluation and a previous supplement, we concluded that the licensee's proposed Post-Accident Sampling System (PASS) met eight of the nine criteria in Item II.B.3 of NUREG-0737 which are relevant for a gas-cooled reactor. The one criterion which was not fully resolved was Criterion (2), for which the staff requested the licensee to provide a plant-specific core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage.

II. Evaluation

By letters dated October 28, 1983, July 2, 1984, July 16, 1984, September 20, September 30, 1985, and November 25, 1985, the licensee provided additional information in response to the above concern.

Criterion (2) "The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three-hour time frame established above, quantification of the following:

a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes);

b) hydrogen levels in the containment atmospheres;

c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids;

d) alternatively, have in-line monitoring capabilities to perform all or part of the above analyses."

The PASS provides in-line monitoring for noble gas activity, CO and moisture in the helium coolant, as well as for radioactivity in the reactor building stack gas. The PASS also provides the capability to collect grab samples of the coolant and of the reactor building atmosphere that can be transported to the radio-chemical laboratory for CO, CO_2, H_2, CH_4, N_2 and radionuclide analyses. These species are indicators of core damage in a gas-cooled reactor, and their relative magnitudes indicate core temperature extremes , fuel particle failure, air ingress or water ingress.

In our SER, we found that the licensee partially met Criterion (2) by establishing an onsite radiological and chemical analysis capability. However, we stated that the licensee should provide a procedure, consistent with the clarification of NUREG-0737, Item II.B.3, Post-Accident Sampling System, transmitted to the licensee on July 9, 1982, to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as the concentrations of other gases and core temperature date. Guidance for the procedure to estimate core damage for water-cooled reactors was provided. The procedure for estimating core damage was to be consistent with those portions of these recommendations which are applicable to a gas-cooled reactor.

The procedure for estimating core damage presented in the letter of July 2, 1984, is not acceptable because it is based solely on the Xe^{133} concentration in the coolant. An acceptable procedure would included consideration of 1) the concentrations of other volatile radionuclides such

as additional xenons, krypton and iodines, 2) the concentration of other gaseous species, such as H_2O, CO, CO_2, H_2, CH_4 and N_2 and (3) core temperature.

The procedure should indicate how these additional considerations would 1) confirm the core damage estimated based on Xe^{133}, (2) provide an estimate of core damage due to water or air ingress, and (3) provide an estimate of extremes of core temperature.

In response to the above, by letter dated November 25, 1985, the licensee provided the results of an analysis of other parameters for use in evaluating core damage. The licensee concluded that:

(1) The concentration of water, carbon monoxide, carbon dioxide and other gaseous impurities in the primary coolant can only be used to provide an upper bound estimate of the total amount of graphite that has been oxidized. The distribution of oxidation within the core, reflector and core support structures cannot be discerned and the amount of oxidation would be expected to vary significantly among these graphite components due to variations in graphite types within the core.

(2) Core temperatures during Loss of Forced Circulation (LOFC) at locations other then the core support blocks can only be inferred from core outlet thermocouple information and additional calculations. The core outlet thermocouples are primarily of value in indicating to the operators whether the LOFC transient is proceeding in a manner similar to that analyzed and described in the FSAR.

(3) Fission product activity is the best indicator of core damage. To this end, procedure RERP-CORE is in place and contains guidance for determining core damage based on Xe-133 concentration can be independently confirmed by utilizing the Kr-88 concentration in primary coolant. Procedure RERP-CORE was revised to incorporate the

use of Kr-88 concentration in primary coolant to confirm the core damage estimate obtained from the Xe-133 concentration in primary coolant.

Conclusion

Based on our review of the above, we conclude that the licensee has satisfactorily addressed our concern regarding compliance with Criterion (2) dealing with establishment of a plant-specific core damage estimate procedure. Thus, the licensee is now in compliance with the nine criteria of Item II.B.3 of NUREG-0737 which are applicable to a gas-cooled reactor. We therefore find the post-accident sampling system at Fort St. Vrain to be acceptable.