DMB

JUL 08 1985

Docket: 50-267

Mr. O. R. Lee, Vice President
Electric Production
Public Service Company of Colorado
P. O. Box 840
Denver, Colorado 80201

Dear Mr. Lee:

Our September 22, 1983, letter provided the Preliminary Safety Evaluation Related to the Post Accident Sampling System (NUREG-0737, Item II.B.3) for the Fort St. Vrain Station (FSV). Your October 28, 1983 (P-83352), and July 2 and 16, 1984 (P-84192 and P-84216) letters provided additional information to resolve the open items in our evaluation. We have reviewed your submittals and have determined that additional information is still required for us to resolve this issue. The results of our review are contained in the enclosed Supplemental Safety Evaluation (SSE).

We have found that you now meet eight of the nine pertinent criteria contained in Item II.B.3 of NUREG-0737 and that some revision is required in the procedures used for estimating the extent of core damage in order for us to find the procedure to be acceptable.

Therefore, we request that you review the attached SSE and provide your comments on resolving our concern within 60 days of the date of this letter.

If you have any questions on this matter, contact the NRC Project Manager - P. Wagner - at (817) 860-8127.

Since this reporting requirement relates solely to FSV, OMB clearance is not required under P.L.96-511.

Sincerely,

Original Signed By
E. H. Johnson

Eric Johnson, Chief
Reactor Project Section 1

Enclosure: SSE on II.B.3

cc: (see next page)

RIV:PM	C:SP&ES	RPB1	D:DRSS
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Supplemental Safety Evaluation by
the Office of Nuclear Reactor Regulation
Related to Operation of
Fort St. Vrain Nuclear Generating Station
Public Service of Colorado
Docket No. 50-267

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

I. Introduction

In our safety evaluation, we concluded that the licensee's proposed Post-Accident Sampling System (PASS) met six of the nine criteria in Item II.B.3 of NUREG-0737 which are relevant for a gas-cooled reactor. The three criteria which were not fully resolved were:

Criterion (2) Provide a plant-specific core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage.

Criteria (9) and (10) Provide information on the procedure for taking samples of highly radioactive coolant for gamma spectrometry in such a manner that the activity of the sample does not exceed the measurement capability of the spectrometer.

II. Evaluation

By letters dated October 28, 1983 and July 2, 1984, the licensee provided additional information.

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, fuel particle failure, air ingress or water ingress. We find that the licensee partially meets Criterion (2) by establishing an onsite radiological and chemical analysis capability. However, 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 data. Guidance for the procedure to estimate core damage for water-cooled reactors was provided. The procedure for estimating core damage should 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 should include consideration of (1) the concentrations of other volatile radionuclides such as xenons, kryptons 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 estimate based on Xe^{133}, (2) provide an estimate of core damage due to water or air ingress, and (3) provide an estimate of core temperature.

Criterion (9):

The licensee's radiological and chemical sample analysis capability shall include provisions to:

a) Identify and quantify the isotopes of the nuclear categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentrations in the range from approximately 1μ Ci/g to 10 Ci/g.

b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

The radionuclides in both the helium coolant and the reactor building atmosphere samples will be identified and quantified using the onsite gamma spectrometer. By letter dated July 2, 1984, the licensee provided information on the procedure to take small low pressure samples of the highly radioactive coolant during the period of maximum activity between approximately 5 hours and 7 days after the onset of a loss-of-cooling accident. By controlling the sample size, the measurement capability of the gamma spectrometer will not be exceeded. Radiation background levels will be restricted by shielding. Ventilated radiological and chemical analysis facilities are provided to obtain results within an acceptably small error (approximately a factor of 2). We find that these provisions meet Criterion (9) and are, therefore, acceptable.

Criterion (10):

> Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe the radiological and chemical status of the reactor coolant systems.

The accuracy, range, and sensitivity of the PASS instruments and analytical procedures are consistent with the recommendations and the clarifications of NUREG-0737, Item II.B.3, Post-Accident Sampling Capability, transmitted to the licensee on June 30, 1982. Therefore, they are adequate for describing the radiological and chemical status of the reactor. The analytical methods and instrumentation are capable of operation in the post-accident sampling environment. No additional training of chemistry personnel is required because the same systems are used for normal and post-accident sampling and analysis. The letter of July 2, 1984, describes provisions to limit sample size, enabling the onsite measurement of radionuclide concentrations in the helium coolant in the post-accident period of maximum coolant radioactivity.

We find that these provisions meet Criterion (10) and are, therefore, acceptable.

Conclusion

We conclude that the post-accident sampling system partially meets the criteria of Item II.B.3 of NUREG-0737. Two of the eleven criteria are not applicable to a gas-cooled reactor. The licensee's proposed methods to meet eight of the remaining nine criteria are acceptable. The criterion which has not been fully resolved is:

Criterion (2): Provide a core damage estimate procedure to include consideration of coolant concentrations of volatile radionuclides and gaseous chemical species together with other physical parameters as indicators or core damage.