Docket No. 50-346

February 10, 1987

DCR 016

Mr. Joe Williams, Jr.
Senior Vice President, Nuclear
Toledo Edison Company
Edison Plaza - Stop 712
300 Madison Avenue
Toledo, OH 43652

Dear Mr. Williams:

SUBJECT: SAFETY EVALUATION - POST ACCIDENT SAMPLING SYSTEM, NUREG-0737, ITEM II.B.3

The staff has completed its evaluation of the Post Accident Sampling System (PASS). This evaluation is based upon information submitted by Toledo Edison Company by letters dated April 25, 1983, August 14, 1984, and October 2, 1986. In our evaluation, we have compared the information provided in your submittals against the eleven (11) criteria specified for Item II.B.3 of NUREG-0737, Post-Accident Sampling System.

With the exception of the revised procedure for estimating the degree of core damage which will be reviewed by the staff as a separate licensing action, we find that the design of the PASS meets the criteria specified for Item II.B.3 NUREG-0737, and is, therefore, acceptable. The Safety Evaluation Report supporting our finding is enclosed.

Sincerely,

/s/

John F. Stolz, Director
PWR Project Directorate No. 6
Division of PWR Licensing-B

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO NUREG-0737, ITEM II.B.3, POST-ACCIDENT SAMPLING SYSTEM

TOLEDO EDISON COMPANY

AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

INTRODUCTION

Subsequent to the TMI-2 incident, the need was recognized for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

To comply with NUREG-0737, Item II.B.3, the licensee should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary and (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the criteria are met.

EVALUATION

By letters dated April 25, 1983, August 14, 1984 and October 2, 1986, the licensee provided information on the PASS.

Criterion (1):

> The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be three hours or less from the time a decision is made to take a sample.

The licensee has provided sampling and analysis capability to promptly obtain and analyze reactor coolant samples and containment atmosphere samples within three hours from the time a decision is made to take a sample. The PASS electrical power supply is from the normal station service power supply. In the event that offsite power is lost, an emergency power supply is available to operate the sampling and analysis panels. We find that these provisions meet Criterion (1) and are, therefore, acceptable.

Criterion (2):

> The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three-hour timeframe 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 non-volatile isotopes);
>
> b) hydrogen levels in the containment atmosphere;
>
> 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 the capability to collect diluted and undiluted samples of reactor coolant and to analyze them for pH, chloride, boron, radionuclide and dissolved hydrogen and oxygen. It also provides the capability to take diluted grab samples of the containment atmosphere for analysis by gas chromatograph for hydrogen. In addition, in-line monitoring capability of the coolant is provided for dissolved hydrogen by an electrochemical sensor and for dissolved oxygen by polarography. In-line monitoring of hydrogen in the containment atmosphere is also provided.

We find that the licensee has met Criterion (2) by establishing an onsite radiological and chemical analysis capability. The licensee has developed an interim procedure for a core damage assessment based on measurements of radionuclide concentrations in the coolant and containment radiation levels. However, the licensee's procedure to estimate the extent of core damage should also take into consideration other physical parameters such as core temperature data, pressure vessel liquid level and hydrogen concentrations. A final plant specific procedure taking the other physical parameters into consideration was submitted October 2, 1986. This procedure will be reviewed as a separate action by the staff.

Criterion (3):

> Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system) to be placed in operation in order to use the sampling system.

Reactor coolant and containment atmosphere sampling during post-accident conditions does not require an isolated auxiliary system to be placed in operation in order to perform the sampling function. The PASS provides the ability to obtain samples from the reactor coolant system, the pressurizer liquid and vapor spaces, the RHR system, the containment sump and the containment atmosphere without using an isolated auxiliary system. However, portions of the existing Nuclear Sampling System need to be placed in service to obtain post-accident samples. All valves used in the PASS are qualified to function in the post-accident environment in which they operate. Therefore, we find that the licensee meets Criterion (3) of Item II.B.3 of NUREG-0737.

Criterion (4):

> Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Pressurized reactor coolant is cooled and analyzed for dissolved hydrogen by a continuous on-line dissolved hydrogen monitor. The hydrogen concentration in pressurized grab samples is measured off-site by gas chromatography. The dissolved oxygen content can also be measured by an in-line polarographic method to concentrations below 0.1 ppm. We have determined that these provisions meet Criterion (4) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion (5):

> The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the applicant shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the applicant shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Chloride analysis of reactor coolant is performed within 96 hours on a diluted sample by ion chromatograph for an initial scoping chloride concentration. Additional accuracy will be obtained by analyzing an undiluted sample that has been decayed sufficiently to be in conformance with ALARA. These provisions meet Criterion (5), and are, therefore, acceptable.

Criterion (6):

> The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC-19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC-19 criterion (October 30, 1979 letter from H.R. Denton to all licensees).)

The licensee, in the August 14, 1984 submittal, indicated that an analysis using a time-motion study and radiation zone maps based upon the NRC-specified source term indicated that PASS personnel radiation exposures would be within the requirements of GDC-19 and Criterion (6). The time-motion study was done using preliminary procedures which did not incorporate all the required sampling steps.

In early 1986, as part of the Design Review Program commitment, the licensee conducted another time-motion study. This latest study, however, used operating procedures which have been approved for use. As in the previous study, the specified source term was used. The results indicated that the dose to the operator in the post-accident situation would exceed GDC-19 guidelines. The licensee reported this to the NRC in Licensee Event Report No. 86-20, and in a letter dated October 2, 1986, informed the NRC that the following corrective modifications have been implemented:

1. The process piping on the PASS skid, where the sample is drawn, is separate from the PASS control panel. A shield wall has been constructed between the skid and the control panel to reduce significantly the dose to the operator.

2. Repeater gauges for monitoring sample flow and temperature have been mounted in a low dose area adjacent to the PASS control panel. This will allow the operator to remain away from the panel while monitoring system function during long sample purges.

With these modifications, the estimated operator doses by the licensee would be below GDC-19 guidelines and meet the requirements of Criterion (6), and are, therefore, acceptable.

Criterion (7):

> The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

Onsite boron analysis will be performed using an automatic titrator with an accuracy of ±5%. Grab samples will be analyzed offsite by the carminic acid or mannitol methods. These provisions meet the recommendations of Regulatory Guide 1.97, Rev. 2, and Criterion (7) and are, therefore, acceptable.

Criterion (8):

> If in-line monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per week until the accident condition no longer exists.

Continuous monitoring capability is provided for dissolved hydrogen in the coolant and for hydrogen concentration in the containment atmosphere. Diluted or undiluted grab samples of coolant and containment atmosphere will be taken for backup analyses. We find these provisions meet Criterion (8) and are, therefore, acceptable.

Criterion (9):

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

a) Identify and quantify isotopes of the nuclide categories discussed above to levels corresponding to the source term given in Regulatory Guides 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 concentration in the range from approximately 1 μCi/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 primary coolant and the containment atmosphere are identified and quantified. Provisions are available for diluted reactor coolant and containment atmosphere samples to minimize personnel exposure. The PASS can perform radioisotopes analyses at the levels corresponding to the source terms given in Regulatory Guides 1.4, Rev. 2 and 1.7. Radiation background levels will be restricted by shielding and ventilation in the radiological and chemical analysis facilities such that analytical results can be obtained within an acceptably small error (approximately a factor of 2). We find 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 radiological and chemical status of the reactor coolant systems.

The accuracy, range, and sensitivity of the PASS in-line instruments and analytical procedures for chloride and boron are consistent with the recommendations of Regulatory Guide 1.97, Rev. 2, 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 coolant. The analytical methods and instrumentation were selected for their ability to operate in the post-accident sampling environment. Equipment used in post-accident sampling

and analyses will be tested approximately every six months. Retraining of operators for post-accident sampling is scheduled at a frequency of once every six months. We find that these provisions meet Criterion (10) and are, therefore, acceptable.

Criterion (11):

In the design of the post-accident sampling and analysis capability, consideration should be given to the following items:

a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

The licensee has addressed provisions for purging to ensure samples are representative, for limiting reactor coolant loss from a rupture of a sample line, and for ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters. The reactor coolant system sampling locations were selected to provide coolant samples that are representative of core conditions. The containment air sample line is not heat traced to limit iodine plateout. However, the iodine concentration in containment air is not one of the parameters used in the licensee's procedure for estimating core damage. Therefore, a provision to reduce plateout in air sample lines is not needed.

We determined that these provisions meet Criterion (11) and are, therefore, acceptable.

CONCLUSION

On the basis of our evaluation, we conclude that the proposed post-accident sampling system meets all of the 11 criteria in Item II.B.3 of NUREG-0737, and therefore, the post-accident sampling system is acceptable. The revised procedure for estimating the degree of core damage, submitted October 2, 1986, will be reviewed by the staff as a separate licensing action.

Dated: February 10, 1987

Principal Contributor:

S. Kirslas