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Docket No. 50-336

JUN 14 1984

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Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141

Dear Mr. Counsil:

SUBJECT: NUREG-0737 ITEM II.B.3 - EVALUATION OF POST-ACCIDENT SAMPLING CAPABILITIES

Item II.B.3. of NUREG-0737 required licensees of operating reactors to submit a report to the NRC staff documenting the final design of their post-accident sampling capability. Your submittal of November 1, 1982 documented how you plan to satisfy each of the eleven criteria of NUREG-0737 Item II.B.3. Additional information was provided by you in your January 12 and April 19, 1984 letters.

In particular, these last two submittals responded to the following requests:

Criterion (1) Provide information regarding provisions for sampling in the event of loss of off-site power during an accident which requires post-accident sampling.

Criterion (2) provide a core damage estimate procedure.

Criterion (4) provide a capability for reactor coolant dissolved oxygen analysis in the event chlorides are determined to exceed 0.15 ppm.

Criterion (10) provide information demonstrating applicability of procedures and instrumentation in the post-accident water chemistry and radiation environment and the capability for radioactivity analysis accuracy that is within a factor of two.

Criterion (11) provide information regarding heat tracing of containment sample lines.

We have completed our review of your responses to the subject NUREG-0737 items. Our findings are contained in the enclosed Safety Evaluation (SE). Based upon our review, we find that all eleven criteria of NUREG-0737 Item II.B.3 are met, and the Post-Accident Sampling System is acceptable. This completes the staff's review of the subject NUREG-0737 items for your facility.

Sincerely,

Original signed by:
James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

ORB#3:DL	ORB#3:DL	CHEB	ORB#3:DL
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO OPERATION OF MILLSTONE UNIT NO. 2
NORTHEAST NUCLEAR ENERGY COMPANY
DOCKET NO. 50-336

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

Introduction

The post-accident sampling system (PASS) is evaluated for compliance with the criteria in NUREG-0737, Item II.B.3. The evaluation was based on acceptance criterion 5 in Section 9.3.2 of Standard Review Plan (NUREG-0800, July 1981) and the guidelines of Item II.B.3 in NUREG-0737. It was initially concluded that six of the eleven post-accident sampling criteria were acceptable. By letters dated January 12 and April 19, 1984, the licensee provided additional information to resolve the remaining five criteria.

Evaluation

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 which is considered a reliable power source by the licensee. The licensee will rely on the dependability of the off-site power sources and on the ability of the operators to assess the condition of the plant and to reach a decision on the desirability of bringing onsite power to the PASS. 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 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., nobel gases; iodines and cesiums, and nonvolatile 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 inline monitoring capabilities to perform all or part of the above analyses.

The PASS provides the means to remotely dilute reactor coolant samples by a factor of 1,000 to reduce personnel radiation exposure. The licensee methodology relates post-accident core damage with measurements of radionuclide concentrations in the reactor coolant and containment atmosphere and with other plant indicators. A realistic differentiation between four major fuel conditions (no damage, cladding failure, fuel overheating, and core melt) can be assessed by qualified personnel. Fission product isotopes which can characterize the extent of core damage and which can be effectively measured are identified. The licensee has committed in the next revision of the procedure to utilize specific ratios of selected nuclides to indicate whether the activity is released from the gap or from fuel overheating/melt. The radionuclide measurements, used in conjunction with other plant indicators of core exit thermocouple temperatures, pressure vessel water level, containment radiation monitors, and hydrogen production from metal/water reaction, are used to develop an estimate of the type and extent of core damage.

An inline hydrogen analyzer is provided for the reactor containment to determine hydrogen concentrations in the containment atmosphere. Reactor coolant total dissolved gases in pH can be determined with inline monitors. Reactor coolant boron, radionuclide gamma spectrum, and gross radioactivity are determined by laboratory analyses on small grab samples obtained via septum and syringe. The chloride content in post accident reactor coolant is done by pologrаphic analyzer.

We find that the licensee meets Criterion (2) by establishing an onsite radiological and chemical analysis capability and an acceptable methodology for estimating core damage.

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 (RWCUS)] 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 each reactor coolant hot leg, each reactor coolant cold leg, the RHR system, the containment sump, and the containment atmosphere without using an isolated auxiliary system. PASS valves which are not accessible after an accident are environmentally qualified for the conditions in which they need to operate. The licensee's response to Criterion (3) is acceptable since PASS sampling is performed without requiring operation of an isolated auxiliary system.

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 samples are cooled and degassed to obtain representative total dissolved gas samples. If chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is less than 0.1 ppm by measurement of a dissolved hydrogen residual of greater than 10 cc/kg is achievable for up to 30 days after the accident. We determined that these provisions meet Criterion (4) 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 licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

A polographic analyzer is utilized for reactor coolant chloride content down to 0.1 ppm. Criterion (5) is met by the licensee by having the capability to perform the chloride analysis on undiluted samples and therefore, is 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 exposure 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 has performed a shielding analysis to ensure that operator exposure while obtaining and analyzing a PASS sample is within acceptable limits. This operator exposure includes entering and exiting the sample panel area, operating sample panel manual valves, positioning the grab sample into the shielded transfer carts, and performing manual sample dilutions, if required, for isotopic analysis: PASS personnel radiation exposures from reactor coolant and containment atmosphere sampling and analysis are within 5 rem whole body and 75 rem extremities which meet the requirements of GDC (19) and 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).

Boron analysis of the reactor coolant will be performed on grab samples by photometric analysis with a measurement capability of 1 ppm to 3,000 ppm under accident conditions. We find this provision meets Criterion (7) and is, therefore, acceptable.

Criterion: (8)

> If inline 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 day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

The PASS has the capability to obtain both undiluted and diluted reactor coolant grab sample and undiluted containment atmosphere grab samples. In addition, an inline chemical analyses panel is provided for measuring reactor coolant pH, total dissolved gas concentrations, as well as containment hydrogen concentrations. Also, a backup (diluted and undiluted) reactor grab sample can be obtained for the offsite analysis at the Haddam Neck Plant which has analysis capabilities similar to that of the licensee. We find that 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 the isotopes of the nuclide categories discussed above to levels corresponding to the source term 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 concentration in the range from approximately 1 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 primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. Counting room 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 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 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 September 2, 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. The standard test matrix and radiation effect evaluation indicated no interference in the PASS analyses.

The licensee plans to conduct quarterly surveillance tests on the PASS as well as emergency drills. Operator retraining will occur 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 the 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 adsorbers and high-efficiency particulate air (HEPA) filters.

The licensee has addressed provisions for purging to ensure samples are representative, remote operated isolation valves to limit reactor coolant loss from a rupture of the sample line, and ventilation exhaust from the PASS is filtered through HEPA filters in the auxiliary building exhaust system. Our intent for Criterion 11b is to minimize discharges or radioactivity during the sample flushing procedure, when the purge water is drained to an open sink. By purging to an enclosed system which has effluent treatment and

radiation monitoring provisions during the sampling process, the licensee is minimizing discharges of radioactivity during sampling. Therefore, this is acceptable.

The licensee has not provided heat tracing on the containment atmosphere sample lines since this sample will be analyzed for noble gases. These results are used for an estimate of core damage from the release of fission products into the containment. Noble gases do not plateout on containment atmosphere sample lines surfaces, therefore, heat tracing, which is used to limit plateout of ionic species, is not needed. We find that this position meets Criterion (11) and is, therefore, acceptable.

Conclusion

On the basis of our evaluation, we conclude that the Post-Accident Sampling System meets all eleven criteria of Item II.B.3 in NUREG-0737, and is, therefore, acceptable.