DCS MS-016

Docket No. 50-335

SEP 28 1983

Dr. Robert E. Uhrig
Vice President
Advanced Systems & Technology
Florida Power & Light Company
P. O. Box 14000
Juno Beach, Florida 33408

Dear Dr. Uhrig:

SUBJECT: POST ACCIDENT SAMPLING SYSTEM (NUREG-0737, ITEM II.B.3)

The staff has completed its review of your Post Accident Sampling System (PASS), NUREG-0737, Item II.B.3. Your submittals of December 6, 1982 and August 15, 1983 supplemented your submittal provided in conjunction with the licensing review of the PASS for St. Lucie Plant, Unit No. 2 that describes the system being utilized in both Units at the St. Lucie Plant site.

As a result of our review, we find that you meet ten of the eleven criteria. Your response to the criterion requiring a procedure for estimating the extent of core damage is acceptable on an interim basis. You should provide a core damage estimate procedure that will include comparison of the PASS data with other plant parameters as discussed in our Safety Evaluation. You are requested to provide a schedule for the submittal of this revised procedure within 30 days of the receipt of this letter.

In addition, you should confirm that heat tracing has been installed on the containment atmosphere sampling lines by June 30, 1984 and that unqualified PASS valves have been replaced with qualified valves prior to restart from your next refueling outage (cycle 7). In view of the extended nature of the cycle 6 refueling outage, you are encouraged to exert every effort to complete these actions prior to cycle 6 restart.

We consider NUREG-0737 Item II.B.3 complete for St. Lucie 1 and any further action associated with the PASS will be handled on a plant specific basis.

Our Safety Evaluation is enclosed.

The information requested in this letter affects fewer than 10 respondents, therefore OMB clearance is not required under P.L. 96-511.

Sincerely,

Original Signed by J. R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Safety Evaluation

cc: See next page

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SAFETY EVALUATION BY
THE OFFICE OF NUCLEAR REACTOR REGULATION
ST. LUCIE PLANT, UNIT NO. 1
FLORIDA POWER & LIGHT COMPANY
DOCKET NO. 50-335

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 licensee should provide information on 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 in accordance with the requirements of NUREG-0737, II.B.3.

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

Evaluation

By letters dated December 6, 1982 and August 15, 1983, 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 PASS has 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. Upon loss of offsite power, the instrument portion of the PASS is automatically picked up on emergency diesel loading. Power to the PASS System solenoid valves and pumping systems is manually loaded after the auto sequence. The staff has determined that these provisions meet Criterion (1) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Criterion: (2)

The licensee shall establish an onsite radiological and chemical analysis capability to provide, within 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 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 in-line monitoring for pH, dissolved oxygen and hydrogen. The PASS also provides the capability to collect diluted or undiluted liquid and gaseous grab samples that can be transported to the radiochemical laboratory for boron, hydrogen, total dissolved gas, chloride, and radionuclide analyses.

In addition, in-line redundant monitoring of containment hydrogen is provided.

The proposed procedure for core damage assessment utilizes only the radiological analysis of samples obtained from the PASS. Sample locations (RCS hot leg, RCS pressurizer, containment sump, containment atmosphere, shutdown cooling system, and steam generator secondary) that are appropriate for core damage assessment are identified. The extent of damage is estimated by characterizing the fission products, calculating the isotopic ratios, and comparing them to estimated gas gap and pellet ratios. The degree of core damage is expressed in terms of the percentage of the total core inventory available for release. This procedure is being revised to include comparisons with additional PASS data (e.g., hydrogen concentrations and total gas content in the samples) and other instrumentation data which relate to core integrity, including RCS pressure, core exit thermocouple temperatures, and containment radiation levels. The staff finds that these provisions partially meet Criterion (2). The procedure for estimating core damage is acceptable on an interim basis.

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, the low pressure safety injection pump discharge, the containment sump, and the containment atmosphere without using an isolated auxiliary system.

In the PASS, the valves V-55001 through V-55006 and a drain valve V-55009 for liquid and gas separator are inaccessible after an accident. These valves were specified to operate for the specified design conditions but do not have the required qualification paperwork. The valves will be replaced with environmentally qualified valves during the next refueling outage. The staff has determined that these provisions meet Criterion (3) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

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 dissolved gas samples (H_2 and O_2). If chlorides exceed 0.15 ppm verification that dissolved oxygen is less than 0.1 ppm is possible. 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. Within 30 days, consistent with ALARA, direct monitoring for dissolved oxygen is provided. The staff has 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 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.

Primary system component cooling at St. Lucie Unit 1 is provided by the Component Cooling Water System. The Secondary Component Cooling Water System provided two barriers against in-leakage from the plant's seawater systems. Consequently, the criteria for a 4-day analysis are met. The sensitivity for chloride analysis is 0.002 ppm. The staff has determined that 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 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 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, meeting 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 diluted grab samples with a measurement capability from 0 ppm to 6,000 ppm under accident conditions.

The staff finds that this provision meets the recommendations of Regulatory Guide 1.97, Rev. 2 and Criterion (7) and is, 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 day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

An in-line chemical analysis panel is provided for reactor coolant pH, oxygen and hydrogen concentrations, as well as containment hydrogen concentrations. Also, a backup (diluted and undiluted) reactor coolant grab sample can be obtained for these analyses. The staff finds 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 1μ Ci/g to 10 Ci/g.

b) Restrict background levels or 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. The PASS can perform radioisotope analyses at the levels

corresponding to the source term given in Regulatory Guides 1.4 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). The staff finds 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 PASS has the analytical ranges and accuracies that are consistent with the recommendation of Regulatory Guide 1.97, Rev. 2, and the clarification of NUREG-0737, Item II.B.3, Post-Accident Sampling Capability, transmitted to the licensee on July 13, 1982. The materials used in the instrumentation have been selected on the basis of their ability to withstand the radiation effects of the post accident primary coolant. The analytical procedure has been developed to ensure that the PASS is maintained in a high degree of reliability as well as maintaining the proficiency of the PASS operators. The procedure (based on a semi-annual frequency) consists of the necessary equipment operability and calibration checks and provides for the training and requalification of the PASS operators. The staff finds 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 line, 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

a) coolant and containment atmosphere samples should be representative of the reacor 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, size of sample line to limit reactor coolant loss from a rupture of the sample line, and ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters.

The PASS atmosphere sample line is not presently heat traced. The licensee intends to install heat tracing on the atmosphere sample line and is currently scheduled to install the heat tracing by June 30, 1984. The staff has determined that these provisions meet Criterion (11) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Conclusion

On the basis of the staff's evaluation, it is concluded that the Post Accident Sampling System meets ten of the eleven criteria of Item II.B.3 in NUREG-0737. The procedure for estimation of reactor core damage is acceptable on an interim basis. The licensee should provide a procedure to estimate the extent of core damage which will include comparisons of the PASS data with other plant parameters such as the RCS pressures, core exit thermocouple temperatures, and containment radiation levels. Prior to restart from the next refueling outage, the licensee should confirm that the present unqualified PASS valves have

been replaced with valves qualified for the appropriate environment. In addition, the licensee should confirm that heat tracing has been installed on the containment atmosphere sampling lines by June 30, 1984.

Principal contributor:
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