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Docket No. 50-312

Mr. Ronald J. Rodriguez
Executive Director, Nuclear
Sacramento Municipal Utility District
6201 S Street
P. O. Box 15830
Sacramento, California 95813

Dear Mr. Rodriguez:

SUBJECT: Rancho Seco Nuclear Generating Plant - Post Accident Sampling System (NUREG-0737 Item II.B.3)

By letter dated February 15, 1983, we provided you with our Safety Evaluation on the Post Accident Sampling System proposed for the Rancho Seco Plant. In our evaluation we stated that you meet seven of the eleven criteria outlined in Item II.B.3 of NUREG-0737. By letters dated May 2, 1983 and June 17, 1983 you provided additional information to resolve the open items for the four remaining criteria. We have completed our review of the information and have concluded that you now meet all eleven of the criteria outlined in NUREG-0737 for the Post Accident Sampling System. Our updated Safety Evaluation is enclosed for your information. This completes this item for the Rancho Seco Plant.

In your May 2, 1983 letter you stated that your proposed method for estimating core damage will be ready for use by the end of the current outage. Please confirm in writing within two weeks of receipt of this letter that this procedure has been ready for use since you restarted from the current outage.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

OFFICE	ORB#4:DL	C-ORB#4:DL					
SURNAME	SMiner:cf	JStolz for					
DATE	9/2/83	9/2/83					

cc w/enclosure(s):

David S. Kaplan, Secretary and General Counsel
Sacramento Municipal Utility District
6201 S Street
P. O. Box 15830
Sacramento, California 95813

Sacramento County
Board of Supervisors
827 7th Street, Room 424
Sacramento, California 95814

Mr. John B. Martin, Regional Administrator
U.S. Nuclear Regulatory Commission
Region V
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Regional Radiation Representative
EPA Region IX
215 Fremont Street
San Francisco, California 94111

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Thomas Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Helen Hubbard
P. O. Box 63
Sunol, California 94586

Christopher Ellison, Esq.
Dian Grueuich, Esq.
California Energy Commission
1111 Howe Avenue
Sacramento, California 95825

Ms. Eleanor Schwartz
California State Office
600 Pennsylvania Avenue, S.E., Rm. 201
Washington, D. C. 20003

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Resident Inspector/Rancho Seco
c/o U. S. N. R. C.
14410 Twin Cities Road
Herald, CA 95638

Atomic Safety and Licensing Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Alan S. Rosenthal, Chairman
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. John H. Buck
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Christine N. Kohl
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Joseph O. Ward, Chief
Radiological Health Branch
State Department of Health Services
714 P Street, Office Building #8
Sacramento, California 95814

Updated Safety Evaluation by
Office of Nuclear Reactor Regulation
Related to the Operating License for
Rancho Seco Nuclear Generating Station
Sacramento Municipal Utility District
Docket No. 50-312

NUREG-0737, II.B.3 - Post-Accident Sampling Capability

Introduction

In our safety evaluation, we found that the licensee's post-accident sampling system (PASS) met seven of the eleven criteria for item II.B.3 in NUREG-0737. The remaining four criteria were unresolved. The four criteria which were not fully resolved are:

Criterion (1) provide backup power source.

Criterion (2) provide a core damage estimate procedure.

Criterion (3) verify that PASS valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Criterion (10) provide information demonstrating applicability of procedures and instrumentation in the post accident water chemistry and radiation environment.

Evaluation

By letters dated May 2 and June 17, 1983, the licensee provided additional 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 3 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. Power from standby station diesels can be manually switched on to energize the PASS in the event of off-site power failure during post-accident conditions. We 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 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 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 concentrations of liquids.

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

The PASS is capable of analyzing, by remotely controlled in-line monitors, noble gases, iodines and cesiums, non-volatile isotopes, boron, chloride, pH, and dissolved gases in the primary coolant. Hydrogen and gamma spectrum in the containment atmosphere can also be analyzed by in-line monitors. In addition, grab samples can be obtained from the containment atmosphere for hydrogen analysis.

The licensee provided a methodology to be used in developing a procedure for estimating core damage and stated that procedures would be in place for implementation by the end of the current outage. The methodology consists of:

A. Key plant parameter information (e.g., in-core temperature monitors, reactor coolant levels, containment hydrogen indicator, etc.) is formulated to establish preliminary indication(s) of possible core damage.

B. The PASS is capable of obtaining representative diluted or undiluted samples of the containment air and reactor coolant during post-accident conditions. A radionuclide analysis will be performed on the PASS samples and the results will be normalized, using equations in the procedure, for dilution, temperature, pressure, radioactive decay and plant power history to establish the extent of core damage. Normalized radionuclide concentrations will be developed for Xe133, I131, Cs134, Cs138, Ru103, Mo99, and other isotopes; with this data, an initial estimate of core damage can be made using a series of graphs.

If the results of the graphical analyses indicate that there is core damage, and the results are supported by information obtained from other plant indicators, the next step in the procedure is to calculate the extent of core damage. The first step is to determine the primary source, or sources, of activity by calculating the ratios of the key isotopes (e.g., Kr87 to Xe133 and I134 to I131). With the activity ratios established, the procedure provides equations to determine the contributions from clad rupture and fuel degradation (i.e., overheat and melt).

C. Finally, the calculated results are compared with the results of other plant indicators and an estimate of core damage based on all available plant information and radionuclide analyses is formulated.

We determined that these provisions meet Criterion (2) of Item II.B.3 in NUREG-0737 and are acceptable. By the end of the current outage, the licensee should provide confirmation that the procedure for estimating the degree of reactor core damage is in place.

Criterion: (3)

Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system [e.g., the letodwn 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 cold leg, the residual heat removal system, the containment sump, and the containment

atmosphere without using an isolated auxiliary system. The PASS valves which are not accessible after an accident are environmentally qualified for the conditions in which they need to operate. We find that these provisions meet Criterion (3) of Item II.B.3 in NUREG-0737 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 12, 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. Refresher training involving use of the PASS equipment will be provided for a sufficient number of designated equipment users to insure the availability of trained personnel to support post-accident sampling requirements. The frequency of such testing/training will be every six months $\pm$ 25%.

We determined that these provisions meet Criterion (10) of Item II.B.3 in NUREG-0737 and are, therefore, acceptable.

Conclusion

Based on our evaluation, we conclude that the post-accident sampling system now meets all of the eleven criteria of Item II.B.3 in NUREG-0737, and is, therefore, acceptable. By the end of the current outage, the licensee should verify that the procedure for estimating the extent of reactor core damage will be ready for use.