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Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Information Related to NUREG-0737, Item II.B.3,
Post Accident Sampling System

In a letter dated September 8, 1982 we were requested to submit additional information related to NUREG-0737, Item II.B.3, Post Accident Sampling. Specifically, we were requested to provide additional information related to methods of compliance with the acceptance criteria established by the NRC Staff for this system.

The requested information is attached. Please contact us if you have any questions related to this material.

David Musolf
Manager - Nuclear Support Services

DMM/bd

cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
NRC Resident Inspector
G Charnoff

Attachment

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NRC 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.

Clarification: Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of off-site power (i.e. designate an alternative backup power source, not necessarily the vital (Class IE) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

Enclosure II request for information on Criterion (1):

Describe the provisions for sampling during loss of off-site power.

NSP Response: In the event of loss of off-site power, electrical power can be provided to necessary sampling, counting, and analystical equipment by running a temporary power cord from energized buses to the de-energized equipment. Discussion with plant personnel concluded that this could be accomplished within one hour from the time of request. In addition, provisions are being made to install a new non-vital bus. This bus will be provided with a non-safeguards diesel generators source in the event of loss of off-site power. All post-accident sampling and analytical chemistry equipment will be added to this bus. The backup counting facility is located in the EOF which will receive power from Dakota Electric Power Association and will also have power source from the plant substation.

NRC Criterion: (2) The licensee shall establish an on-site 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 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.

Clarification: 2 (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation to minimize personnel radiation exposure (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and nonvolatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information.)

2. Provisions to estimate the extent of core damage bases on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

2 (b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

2 (c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97, Rev. 2.

2 (d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Enclosure II request for information on Criterion (2):

> Provide procedure for relating radionuclides concentrations to core damage. Discuss on-site radiological and chemical analysis capability.

NSP Response: We have installed inline systems to allow recirculation, isolation and analysis of a pressurized liquid sample, from the reactor coolant system or the RHR system for hydrogen and radiogases. The sample panels and associated piping and valves are well shielded to allow sampling within limits set forth in 10CFR20. Remote analyzer readouts are provided to further reduce exposure levels. The pressurized liquid sample is depressurized and circulated through the hydrogen analyzer. The hydrogen analyzer has three range scales and is calibrated from 0 to 100% Hydrogen. The sample panel also has provisions for extracting a gas sample for radionuclide analysis. The radiogas sample can be diluted by a factor of 10 with nitrogen, prior to extraction, to reduce the sample activity for transport and counting. These inline sample systems have been installed on both units' sample trains and can be cross-connected to provide additional reliability for post-accident sampling. Additional hydrogen and radiogas analysis can be completed by utilizing installed grab sample capabilities and analysis with a gas chromatograph for further backup reliability.

Provisions have been installed to facilitate the drawing of a depressurized liquid sample for Boron, Chloride and pH analysis. These sample lines diameters have been reduced, lengths shortened and they have been well shielded to reduce the radiation levels in the sample area to allow sampling within 10CFR 20 limits. Additionally, sample recirculation drains from both the pressurized and depressurized samples may be routed to the affected unit's RHR sump and returned to containment during post-accident sampling. This depressurized grab sample can then be transported within a shield to a shielded, ventilated facility where pH can be taken on the sample using a standard pH meter. A one milliliter aliquot of sample is then diluted to reduce radiation exposure. That diluted sample (or similar sample) is used for the boron, chloride and isotope analysis. Boron is determined using a carminic acid procedure and spectrophotometer. The chloride analysis will be conducted on a liquid ion chromatograph. The isotopic analysis can be completed on either in-house analytical equipment or on a completely redundant off-site facility. The utilization of extended counting geometries and sample dilution allows determination of gross activity within the guidelines set forth in Regulatory Guide 1.97, Rev. 2. All equipment and instrumentation required for the boron and pH analysis can be backed up with our normal laboratory equipment.

A draft procedure has been written relating radionuclide concentrations to core damage. (Analysis of the samples by a Ge(HP)

detector will yield isotopic reports for short and long-lived volatile and nonvolitile radionuclides.) The results of the analysis of containment atmosphere and reactor coolant activity is used in this procedure to estimate core damage. This procedure will be in our Emergency Plan Implementing Procedures. The NRC will be receiving copies through the normal Emergency Plan Implementing Procedure distribution system.

NRC 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.

Clarification: Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is < 0.1 ppm by measurement of a dissolved hydrogen residual of ≥ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Enclosure II request for information on Criterion (4):

Provide discussion of method to verify that dissolved oxygen is less than 0.1 ppm if chloride exceeds 0.15 ppm.

NRC Response: We can accurately measure the reactor coolant hydrogen via methods stated in response to Criterion 2. This method of hydrogen analysis is accurate to ±10% down to a low range of 0.5 cc/kg. Therefore, we do not measure post-accident oxygen concentrations.

NRC 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.

Clarification: The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

In general, this analysis should be accurate within ±5% of the measured value (i.e. at 6,000 ppm B the tolerance is ± 300 ppm while at 1,000 ppm B the tolerance is ± 50 ppm). For concentrations below 1,000 ppm the tolerance band should remain at ± 50 ppm.

- Chloride: measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

- Hydrogen or Total Gas: monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of ± 10% is desirable between 50 and 2000 cc/kg but ± 20% can be acceptable. For concentration below 50 cc/kg the tolerance remains at ± 5.0 cc/kg.

- Oxygen: monitored to assess coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen the analysis should be accurate within ± 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at ± 0.05 ppm.

- pH: measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate with in ±0.3 pH units. For all other ranges ±0.5 pH units is acceptable.

To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituent	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs^+	250	Cesium Nitrate
Ba^{+2}	10	Barium Nitrate
La^{+3}	5	Lanthanum Chloride
Ce^{+4}	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li^+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^+	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

Enclosure II request for information on Criterion (10):

Turbidimetric chloride is not applicable on diluted reactor coolant sample due to lack of sensitivity and due to iodine

and other halogen fission product interference. Select an alternate chloride analytical procedure. Provide accuracy, range and sensitivity for hydrogen, oxygen, and total gas, and pH. Provide information demonstrating applicability of procedures in the post-accident water chemistry and radiation environment.

NSP Response: Chloride samples will be run on a liquid ion chromatograph. An aliquot of the depressurized coolant will be diluted 1 to 100 with demineralized water. The diluted sample will be allowed to decay for up to four days. A 25 ml volume of the diluted sample will then be injected into the ion chromatograph. The ionic species are then collected on a concentrator column and later eluted to the detector.

The accuracy, range, and sensitivity of the chemical analysis meets that required by Regulatory Guide 1.97, Rev. 2 as follows:

	Accuracy	Range	Sensitivity
Chloride	± 10% 50ppb below 0.5 ppm	10ppb - 20ppm	10ppb
Hydrogen	± 10%	0.5-200 cc/kg	0.5 cc/kg
pH	± 0.3	100-6000ppm	100ppm
Boron	50ppm below 1000 ppm		

NOTE: Chloride and boron analysis ranges may be increased through further dilution.

To provide procedure and instrument applicability in a post-accident environment, the test matrix detailed in Criterion (10) was prepared and the chemical analysis required in Regulatory Guide 1.97, Rev. 2 were run. The accuracy, range and minimum sensitivities set forth in this response were reproducable during testing. The test matrix was diluted the appropriate amount in accordance with Prairie Island procedures. The chemical analyses were conducted in both the standard test matrix and under appropriate spray additive conditions (2000 ppm NaOH). The test results were unchanged. Therefore, we will not support separate analysis for each accident condition.

These detailed sample results are on file and are available if further clarification is desired.

NRC 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-efficieny particulate air (HEPA) filters.

Clarification: (11) (a) A description of the provision which address each of the items in clarificaton 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(11) (b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

Enclosure II request for information on Criterion 11:

Confirm that post-accident reactor coolant and containment atmosphere samples will be representative

of the reactor coolant in the core area and the containment atmosphere following an accident. Address heat tracing of containment samples.

NSP Response:

The containment atmosphere is mixed with four Fan Coil Units and four dome recirculation fans with flows of 29,000 CFM and 3000 CFM respectively during an accident. The containment sample is taken at the discharge of the dome recirculation fans. With this large flow rate, the sample will be representative of the general containment atmosphere.

The containment sample leaves containment and flows to a condensing pot where the stream is cooled and the steam condensed. This water is sampled as well as a downstream silver zeolite cartridge, a particulate filter, and, if desired, a noble gas sample can be taken. Because we purposely cool the sample and condense the steam, no heat tracing is required. (The fact that the containment atmosphere could be at 275°F saturated with steam makes a sample stream heat tracing impractical at Prairie Island.)

The primary coolant system samples can be taken from two locations: Loop B Hot Leg or RHR pump discharge. The RHR pumps taken suction from the containment sump when the reactor coolant system is placed on recirculation following a loss of coolant accident to containment. The sample point on the RHR discharge is after the heat exchanger but before the containment vessel on a line that supplies the reactor vessel.

On a large break LOCA, the RHR pump suction is lined up to the containment sump shortly after the event. The primary sampling from the RHR discharge, as it is being pumped into the reactor vessel, should be a representative sample.

On a LOCA to containment that is large enough for voids in the hot legs with both safety injection pumps operating will be large enough to supply RHR pump suction from containment within about one hour after the break. Except for the degasing, the containment sump water would be representative of the primary coolant water. A smaller small-break-LOCA that does not void in the hot leg should make the Loop B sample point representative.

During a steam generator tube rupture, the hot legs should not be steaming. Because of a natural circulation or forced circulation, the Loop B Hot Leg sample point will be a representative sample.