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February 9, 1983
5211-83-016

Office of Nuclear Reactor Regulation
Attn: J. F. Stolz, Chief
Operating Reactor Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Post Accident Sampling System (NUREG 0737 Item II.B.3)

This letter (enclosure 1) provides responses to your letters of July 8, 1982 and October 7, 1982 and addresses open items from NUREG 0680. For your convenience we have provided (enclosure 2) the criteria/clarifications from the above NRC letters.

Sincerely,

RJ Toole for

H. D. Hukill
Director, TMI-1

HDH:CWS:vjf

Enclosures: (1) Responses to NRC criteria/clarifications for NUREG 0737 Item II.B.3
(2) Attachment No. 1 to Post Accident Sampling System NUREG 0737 II.B.3 Evaluation Criteria Guidelines and NRC letter dated October 7, 1982.

cc: R. C. Haynes
J. Van Vliet

A046

ENCLOSURE I

Response to Criterion (1) as Clarified:

See Sections 2.1.2.4.1 and 2.1.2.4.2 of the TMI-1 Restart Report for details on the Reactor Coolant Sampling and Containment Atmospheric Sampling systems. Additionally pressurized samples can now be obtained and analyzed (H_2, N_2, and total gas). During Loss of Offsite Power (LOOP) the following equipment is used in obtaining an RCS sample within the 3 hour time frame.

a. Emergency ventilation system (AH-E-90 and AH-E-91 fans) is powered by a diesel backed system. However, these fans are manually loaded after diesel start.

b. The new Gamma spectrometer with multichannel analyzer and the vacuum pump are being placed on the emergency power system (diesel backed or inverter backed).

c. Cooling water to the sample cooler is part of the nuclear services closed cooling water system and has an emergency power backup.

d. Remote operated valves (CA-V2 and CA-V13) are backed by emergency power.

e. The gas chromatograph and other analytical accessories can be powered from other alternate available power sources to meet the 3 hour limit in the case of a LOOP.

During Loss of Offsite Power the following equipment is used in obtaining and analyzing a containment atmospheric grab sample within the 3 hour time frame.

Containment sampling valves (CMV7, 8, 9, 10, 12, 13) are capable of being operated by an inverter in the event of loss of offsite power.

Gas Chromatagraph (see a. above).

Response to Criterion (2a) as Clarified:

TMI-1 has sufficient radiological analysis capability. Current instrumentation is listed in attached Table 1. Based on our post accident shielding evaluation of the TMI-1 FSAR Update Section 11.A.6 and additional shielding added to system (2 inches of lead to the supply and return lines; 4 in. of lead in front of new sampling sink) radiation exposures will be minimized. Using guideline information supplied by the NRC Materials and Qualification Division, plant specific procedures are being developed to correlate radionuclide type/concentration/activity, core temperature and sampling location with the degree of core damage.

Response to Criterion (2b) as clarified:

See Restart Report Section 2.1.2.4.2. Additionally, a continuous hydrogen (post accident) monitor is installed which indicates (0-10% H_2) in the Control Room. The system is nuclear safety related Class IE and redundant. The analyzers, (which are located in the Intermediate Building) control units and associated equipment are qualified to IEEE 3.23-1974. The sample lines are seismically supported and all other equipment is seismically installed. The system is powered from vital buses. A common alarm for both channels is annunciated in the Control Room at H_2 concentrations above 2%.

Response to Criterion (2c) as clarified:

The sampling system is capable of obtaining a post accident reactor coolant sample at 2500 psig and 600°F. This sample will then be cooled, depressurized and analyzed by the gas chromatograph for total gas or hydrogen.

Chloride and boron samples are discussed in Section 2.1.2.4.1 of the TMI-1 Restart Report.

Furthermore, the units has the capability of measuring gamma spectrum, pH and dissolved O_2 (see response to Criterion 4) as described in the TMI-1 Restart Report.

Response to Criterion (2d) as clarified:

TMI-1 does not have a liquid inline monitoring system. The containment Hydrogen in line monitoring system will be described in our final response to NUREG 0737 Item II.F.1.6.

Response to Criterion (3) as clarified:

Section 2.1.2.4 of the Restart Report discusses the valves necessary for the operation of the sampling system. As described in Table 2.1-1 and 2 of the TMI-1 Restart Report, sufficient override capability exists to take a sample. As shown on schematics 302-673 (gas) and 302-671 (liquid), no isolated auxiliaries are required to place the RCS sampling system in operation. As shown on schematic 302-721, no isolated auxiliaries are required to place the containment sampling system in operation.

Isolation valves CAV 1, 2, 3 and 13, which are not accessible after an accident, are environmentally qualified as described in GPUN response to IEB 79-01B dated August 28, 1981 (RB Isolation Section).

Response to Criterion (4) as clarified:

In the event of an accident a pressurized sample will be taken and then depressurized to permit degassing. The gases evolved will be collected and analyzed for Hydrogen. If the chloride level of the liquid sample taken exceeds .15 ppm and either (1) the hydrogen level cannot be maintained or returned to greater than 10cc/kg; or (2) 30 days have elapsed, then a sample will also be analyzed for dissolved oxygen.

Response to Criterion (5) as clarified:

TMI-1 cooling water is from the Susquehanna River and is not brackish. The chloride analysis will be performed offsite within 4 days. Since the sample set for chloride analysis will be diluted, an undiluted sample will be retained at the unit for analysis within 30 days consistent with ALARA.

Response to Criterion (6) as clarified:

Personnel exposure based on person motion studies are include in Section 2 of the Restart Report. Predicted personnel exposure levels are currently being reviewed against as-built conditions. This review is targeted for completion for May, 1983.

Response to Criterion (7) as clarified:

The capability to perform a boron analysis is discussed in section 2.1.4.1 of the Restart Report.

Response to Criterion (8) as clarified:

TMI-1 does not have an inline liquid monitoring system. The containment atmosphere grab sample is addressed in Section 2.1.2.4.1 of the TMI-1 Restart Report.

Response to Criterion (9a) as clarified:

Post Accident reactor coolant and containment atmosphere source terms are discussed in section 2.1.2.4.3 of the Restart Report. Recently, a new Multichannel Analyzer/data processing system has been delivered to the site; this increases the plant's capability to analyze and process data.

Response to Criterion (9b) as clarified:

The initial post accident plant shielding study is contained in section 11.A.6 of the TMI-1 FSAR Update. Further, the background levels of the Count Room are being evaluated specifically and the results are scheduled to be completed in May 1983.

Response to Criterion (10) as clarified:

GPUN is currently evaluating and performing demonstrations of instrumentation procedures to show analysis capability in a post accident radiation environment.

Response to Criterion (11a) as clarified:

Flushing

The TMI-1 liquid and gas sampling systems are described in section 9.2.2 of the FSAR Update. The RCS sample flows from the B loop cold leg to a letdown line through CAV 13 to the sample line. During steady state natural circulation flow through loop B with minimum decay heat (about .01 of ANS 5.2 with a 1.0 multiplier) would be about 120 lbm/sec. By purging the sampling system with a minimum of 2 volumes, a representative sample can be obtained.

Flushing (cont.)

The containment atmosphere sample shares the same Reactor Building penetrations as the existing RM A2 and is drawn by an eductor through heat traced lines both the mixing provided by the eductor and height of the sample location provide adequate assurance of representativeness.

Loss

Sample loss is prevented by the use of bellows type isolation valves which prevent stem leakage. The system is also checked for leaks once per refueling cycle, as part of the Surveillance and Inservice Inspection program.

Blockage

RCS sampling lines can be blown through with high pressure (2250 psig) nitrogen. For the containment atmosphere, instrument air or compressed air/gas from bottles would be used.

Disposal

Excess samples of reactor coolant liquid will be flushed into the sample sink drain with demineralized water. Remaining material stays in sampling lines, and is ultimately passed upstream of the makeup filters when subsequent sampling is done.

Response to Criterion (11b) as clarified:

All sample hoods and the gas chromatograph exhaust to the ventilation system for the Control Building, which is routed through charcoal and HEPA filters.

POST ACCIDENT SAMPLE ANALYSIS INSTRUMENTATION

Make/Model	Type	Range	Instrument Accuracy	Use
GOW-MAC 69-750	Gas Chromatograph	0.1 to 100.0% for O_2 0.2 to 100.0% for N_2 0.005 to 100.0% for H_2	± 5% of reading	Gas analyses
GOW-MAC 69-700	Strip recorder	±0.25% of scale		"
Orion 701A	pH meter	0-13.99	± .002pH, ± 1 mv, or ± 0.1% of reading, whichever is greater.	Boron analyses
Princeton/ Gammatech. Intrinsic Germanium Detector, Canberra Series 85 MCA, PDP 1124 CPU	Gamma Spectrum-Detector, Multi-Channel Analyser, Computer	Selectable	Isotope dependent	Gamma spectrum
Orbisphere Model 2713	Dissolved Oxygen Measuring System	0.001-20 ppm	± 20% in the 100 ppb range	Dissolved Oxygen Analysis
Beckman DU-7U UV/VIS (Onsite Analysis)	Spectrophotometer	0.05 ppm - 5 ppm	0.05 - 0.1 ± 0.01 ppm 0.10 to 5 ppm .054 x Chloride Concentration	CL^- Analysis

ATTACHMENT NO. 1 TO
POST ACCIDENT SAMPLING SYSTEM
NUREG-0737, II.B.3 EVALUATION
CRITERIA GUIDELINES

The post accident sampling system will be evaluated for compliance with the criteria from NUREG-0737, II.B.3. These eleven items have been copied verbatim from NUREG-0737. The licensees submittal should include information equivalent to that which is normally provided in an FSAR. System schematics with sufficient information to verify flow paths should be included, consistent with documentation requirements in NUREG-0737, with appropriate discussion so that the reviewer can determine whether the criteria have been met. Further information pertaining to the specific clarifications of NUREG-0737, which will be considered in the reviewers evaluation are listed below. Technically justified alternatives to these criteria will be considered.

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).

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 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 exposures (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 non volatile 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 based 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).

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.

Clarification: System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

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 chlorides 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.

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.

Clarification: BWR's on sea or brackish water sites, and plants which use sea or brackfish water in essential heat exchangers (e.g. shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform a chlorida analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

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).

Clarification: Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

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).

Clarification: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Criterion: (8) If inline monitoring in 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 samplies. 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.

Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

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 terms 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 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.

Clarification: (9) (a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(9) (b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

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 ± 50ppm). 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 within ±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

Constituient	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.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance wit Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

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.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification 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.

Rec 10/12/82

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

October 7, 1982

Docket No. 50-289

CUS/82-203

Mr. Henry D. Hukill
Vice President
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

By letter dated July 8, 1982 we sent you a request for additional information concerning NUREG-0737 Item II.B.3 Post Accident Sampling System. There has been confusion concerning our request and the purpose of this letter is to provide some clarification. First, the requirements of this item are as described in NUREG-0737. The clarification section of our previous letter provides staff guidelines not requirements on how to meet the NUREG-0737 criteria.

Additional guidance concerning our initial letter is as follows:

1. NUREG-0737 states that the licensee should be able to perform sampling and analysis within 3 hours of deciding to take the sample. Our clarification section asks how the 3 hour limit is to be met during a loss of offsite power. It was not meant to imply that the sampling system had to be operational during a loss of offsite power. Rather the intent was if there is a loss of offsite power, can you meet the three hour limit.

2. Clarification 2(d) of our original request asked for a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. A detailed reliability analysis is not required to satisfy the staff concerns in this area. The staff needs enough data to provide reasonable assurance that the on-line instrument will function when needed.

3. Clarification 3 of our original request discussed environmental qualification of certain valves. These valves should already be on the list of equipment that must be environmentally qualified by previous Commission Order. Therefore a statement that the valves are in the previously submitted environmental qualification program will satisfy staff concerns for this review.

4. Clarification 6 requested information on the predicted man-rem exposures based on person-motion sampling, transport and analysis of all parameters. This information is necessary to confirm that the licensee has made adequate provisions to meet GDC 19 requirements.

5. Finally, several portions of the clarification section refer to Regulatory Guide 1.97 Revision 2. For purposes of this information request, Regulatory Guide 1.97 is recognized as a recommendation and not a requirement.

The above should resolve the present concerns in this area. If you have further questions please contact your assigned NRC Project Manager.

Sincerely,

John F. Stolz

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

cc: See next page