ATTACHMENT C

FOR INFORMATION ONLY

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

RADIOLOGICAL AND ENVIRONMENTAL SERVICES DEPARTMENT

PROCEDURE NO.: RTP-34

TITLE: CONTAINMENT RADIATION MONITORING - POST ACCIDENT*

PORC Review No./Date Meeting No. 84-020 Date 4/4/84

Approved By: by div. R Ahn
Resident Manager

Approved By: Eric Mulcahey
Radiological and Environmental
Services Superintendent

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Date 03/84

# CONTAINMENT RADIATION MONITORING - POST ACCIDENT*

## 1.0 PURPOSE

The purpose of this procedure is to describe the containment high range radiation monitors, operation and use.

## 2.0 REQUIREMENTS AND ACCEPTANCE CRITERIA

### 2.1 Technical Specifications

None

### 2.2 Plant

2.2.1 Containment high range radiation monitors should be operable during operation of the plant.

2.2.2 During each refueling outage, the containment high range radiation monitors should be source checked.

## 3.0 SPECIAL EQUIPMENT

### 3.1 Containment High Range Radiation Monitors

Two radiation monitors (27-RM-104A and B) provide gamma radiation indication of the primary containment. These monitors were designed to meet the requirements of NUREG-0578 and NUREG-0737.

The General Atomics Model RD-23 gamma radiation detectors are widely separated and located in the large volume portion of the drywell. Detector "A" (27-RE-104A) is located on the south side of the drywell at elevation 290 feet four inches, above 10-MOV-18 near the ladder to the 292 foot elevation. Detector "B" (27-RE-104B) is located near the east recirculation discharge line at elevation 288 feet four inches, above the east unit cooler.

Each detector has an internal U-234 "bug" source to provide a minimum detector current corresponding to 1 R/hr.

The General Atomics Model RP-2C readout modules are located in the main control room on process radiation monitor panel (09010). The readout log ratemeter displays the detector indication over a range of 1E+1 to 1E+8 R/hr. Front panel controls include three alarm lights/reset buttons, and a spring loaded, three-position function switch for normal operation, trip level adjustments, and internal check signal actuation. In the CHECK position, an electric current is

generated which produces a mid scale indication. Test points and adjustment potentiometers are accessible from the front panel.

Each containment radiation monitor has "FAIL", "ALERT", and "HI-HI" annunciator alarms located on the main control room panel (09-4). Computer alarms F545, F546, F547 for 27-RE-104A are FAIL, HI, and HI-HI alarms respectively. Computer alarms A564, A565, A566 for 27-RE-104B are FAIL, HI, and HI-HI alarms respectively. Analog computer points E026 and E027 are available for "A" and "B" monitors respectively.

3.2 General Atomics RT-11 Calibrator

The General Atomics RT-11 Calibrator, weighing 56 pounds, consists of a Cs-137 gamma source in a removable housing and a shielded carrier. The source is positioned next to the RD-23 gamma detectors to verify that the detector responses to radiation. An activity of about 100 mCi is distributed in a line which allows this source to produce an on scale reading of approximately 10 R/hr.

When not in use, the RT-11 is contained in the portable shielded holder which reduces surface exposure rates to reasonable levels. The source is basically a lead brick with attenuators, and must be handled carefully.

4.0 PROCEDURE

4.1 Containment Exposure Rate Evaluation

4.1.1 A document entitled "Procedures for the Determination of the Extent of Core Damage Under Accident Conditions" identified by General Electric as NEDO-22215 provides a method to estimate core damage based on containment radiation levels. This procedure uses the above document to interpret containment radiation monitor data under accident conditions.

4.1.2 Obtain the containment radiation monitor reading from the control room in rem/hr for each available monitor and record data on Figure 1.

4.1.3 Determine the elapsed time from plant shutdown to the monitoring reading in hours as follows:

a. Record time of reading on Figure 1.

b. Record time of shutdown on Figure 1.

c. Calculate hours since shutdown and record on Figure 1.

4.1.4 Normalize the radiation monitor to the reference plant using the following formula:

(rem) normalized = rem (1,670÷P) (V÷237, 450) (6÷D)

Where

P = FitzPatrick reactor power, 2,436

V = Total containment free volume, 267,000

D = Distance of detector to shield wall;

Approximately five feet for 27-RE-104A

Approximately ten feet for 27-RE-104B

a. For 27-RE-104A

(rem) normalized = rem x 0.925

b. For 27-RE-104B

(rem) normalized = rem x 0.463

c. Record the normalized radiation data on Figure 1.

4.1.5 Using Figure 2, determine the fuel inventory release. Record the best estimate for each detector on Figure 1.

4.2 Detector Source Check

4.2.1 This section of the procedure shall be coordinated with the Instrument and Control Department surveillance test of this system. Each detector is exposed to a radiation source by the Radiological and Environmental Services Department and the Instrument and Control Department verifies that the monitor is responding to radiation.

a. Assure that the Instrument and Control Department is fully prepared to obtain the necessary data for verification of monitor response to radiation.

b. Two technicians transport the radiation source and holder to the 292 foot platform in the drywell and clear the area around and below the detector to minimize radiation exposure to other workers.

c. One technician removes the source from the holder and transports it to the appropriate detector.

d. The other technician leaves the drywell and notifies the control room that the source is being placed on the detector.

e. The source is to be held with the source line parallel to and touching the cylindrical part of the detector housing.

f. The control room personnel obtain necessary data as rapidly as possible.

g. The control room personnel notify the drywell technicians immediately when the data has been collected.

h. The source is then removed and placed back into the shielded carrier.

i. If the second detector is to be checked, leave the source secured in the drywell. Leave the drywell until the control room is fully set up and then repeat the check procedure for the other detector.

j. When complete, notify an RES Supervisor of the results.

## 5.0 REFERENCES

5.1 Procedures for the Determination of the Extent of Core Damage Under Accident Conditions, NEDO-22215, General Electric.

5.2 J.A.F. Modification F.I. 80-16 Containment High Range Radiation Monitors.

5.3 High Range Gamma Radiation Monitoring System Operation and Maintenance Manual No. E-115-876, General Atomics Company, 1981.

5.4 RT-11 Calibrator Description and Operating Instructions No. E-255-1040, General Atomics, October 1981.

## 6.0 ATTACHMENTS

6.1 Figure 1, Containment Radiation Monitor Data/Work Sheet

6.2 Figure 2, Approximate Source and Damage Estimate

Figure 1

CONTAINMENT RADIATION MONITOR DATA/WORK SHEET

Name: _____

Date: _____

Time: _____

A. Radiation Monitor Data

17-RE-104A Rem: _____ Date: _____ Hour: _____

17-RE-104B Rem: _____ Date: _____ Hour: _____

B. Elapsed Time (hours)

Shutdown Date: _____ Time: _____

Hours Since Data Obtained: _____ Hours

C. Normalized Radiation Data

17-RE-104A _____ rem x 0.925 = _____ rem (normalized)

17-RE-104B _____ rem x 0.463 = _____ rem (normalized)

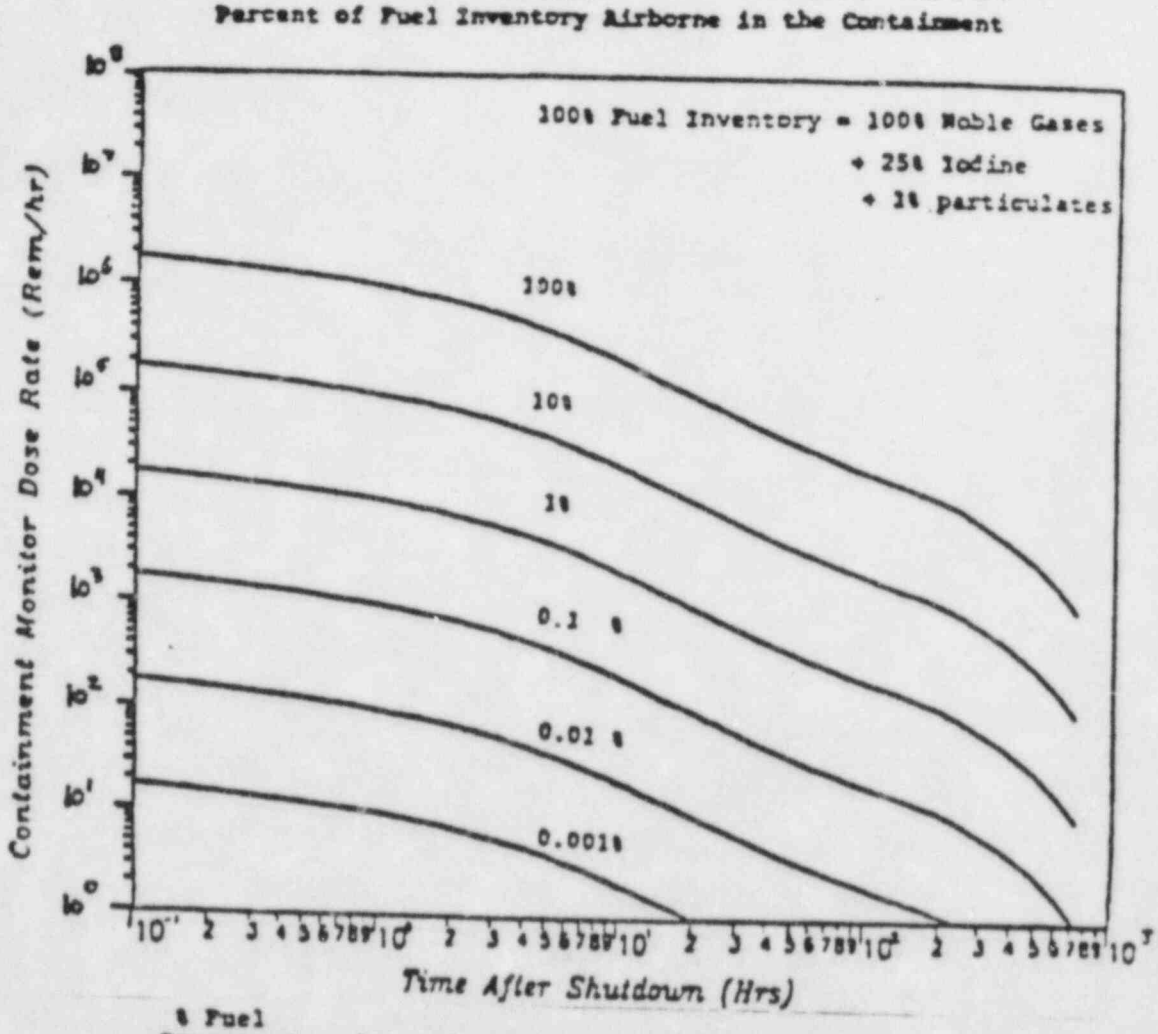
D. Best Estimate of Fuel Inventory Release from Figure 2

17-RM-104A _____

17-RM-104B _____

Figure 2

APPROXIMATE SOURCE AND DAMAGE ESTIMATE




| % Fuel Inventory Released | Approximate Source and Damage Estimate |
|---|---|
| 100. | 100% TID-14844, 100% fuel damage, potential core melt. |
| 50. | 50% TID noble gases, TMI source. |
| 10. | 10% TID, 100% NRC gap activity, total clad failure, partial core uncovered. |
| 3. | 3% TID, 100% WASH-1400 gap activity, major clad failure. |
| 1. | 1% TID, 10% NRC gap, Max. 10% clad failure. |
| .1 | .1% TID, 1% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies. |
| .01 | .01% TID, .1% NRC gap, clad failure of 3/4 fuel element (36 rods). |
| $10^{-3}$ | .01% NRC gap, clad failure of a few rods. |
| $10^{-4}$ | 100% coolant release with spiking. |
| $5\times10^{-6}$ | 100% coolant inventory release. |
| $10^{-6}$ | Upper range of normal airborne noble gas activity in containment. |

NEW YORK POWER AUTHORITY

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

JPN-84-23

ATTACHMENT D

ATTACHMENT 1

GENERAL ELECTRIC

NUCLEAR POWER SYSTEMS DIVISION
GENERAL ELECTRIC COMPANY • 175 CURTNER AVENUE • SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

MFN-006-84
FRH-003-84

January 18, 1984

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D.C. 20555

Attention: Darrell G. Eisenhut, Director

Gentlemen:

SUBJECT: ACCURACY OF DISSOLVED GAS MEASUREMENT FOR GE POST-ACCIDENT SAMPLING SYSTEM (PASS)

Reference: Meeting, between NRC management (R. Vollmer, et.al.) and GE (J. Quirk, et.al.) dated December 12, 1983 on the subject of need for dissolved gas measurement in BWR.

The purpose of this letter is to document the agreements reached between the NRC Staff and GE at the reference meeting and to justify the adequacy of the currently installed PASS for measurement of dissolved gas.

At the reference meeting, GE provided a technical justification (Attachment 1) supporting our contention that post-accident monitoring for dissolved gases in a BWR is not necessary. In addition, GE is providing the expected hydrogen partitioning between primary water, reactor vessel steam dome and containment (Attachment 2). These calculations assume the reactor to be at high pressure, which maximizes the possible dissolved hydrogen. They also assume that a sample is taken very soon after a postulated significant metal-water reaction event, before continued safety/relief valve actuation completely flushes the primary system of hydrogen.

Based on these results, GE continues to feel that such measurements are not needed in order to address NRC's concerns of estimation of degree of core damage and assessment of coolant corrosion potential. Despite this position, GE has supplied a dissolved gas measurement capability as part of the PASS design.

The GE PASS quantifies the concentration of total dissolved gas by allowing the expansion of the dissolved gas from a known volume of liquid into a partially evacuated gas collection region and measuring the pressure rise in that gas collection region. The accuracy of this measurement for the GE PASS has been determined by GE to be at least ±50%

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22020 345
22020 355

for dissolved gas concentrations between 25 cc/kg and 50 cc/kg and at least ±30% for dissolved gas concentrations greater than 50 cc/kg.

GE feels that these accuracies are sufficient for dissolved gas measurement in the BWR for the following reasons. At low pressure following an accident, the gases evolved from the core will be essentially all released to the containment atmosphere, and the dissolved gas concentration for these conditions will not be useful. Even under postulated high pressure conditions following an accident, Attachment 2 shows that, for a postulated dissolved gas concentration of 50 cc/kg: (1) the weight of hydrogen in the reactor pressure vessel is at most 5% of the total hydrogen generated by the metal-water reaction due to discharge of the safety/relief valves to maintain system pressure and (2) the dissolved hydrogen is that associated with the reaction of only 0.5% of the cladding with water. Consequently, even if the dissolved gas contribution is totally ignored under postulated high pressure conditions and a dissolved gas concentration of 50 cc/kg, the absolute error in the total hydrogen release for estimation of core damage would be at most 0.5% metal-water reaction.

GE analyses have demonstrated that, for high pressure conditions (~1050 psig) and total dissolved gas concentrations of greater than 25 cc/kg, the dissolved gas composition is essentially all hydrogen. As noted previously, at low pressure conditions, the dissolved gas concentration would be too small to be useful as a core damage indicator. The NRC staff position as given at the reference meeting was that monitoring for dissolved hydrogen was sufficient to preclude the need for monitoring of dissolved oxygen.

GE will be providing procedures to all of its PASS users for measurement of dissolved gas.

A prompt response to this letter would be appreciated. If you have any questions, please call J. F. Quirk at (408) 925-2606 or F. R. Hayes at (408) 925-2140 of my staff.

Sincerely,

Glenn G. Sherwood, Manager
Nuclear Safety and Licensing Operation

GGS:csc/I12143

cc: R. Johnston (NRC) T. Sullivan (NRC) F. Witt (NRC)
R. Vollmer (NRC)

NEW YORK POWER AUTHORITY

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JPN-84-23

ATTACHMENT E

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The following table of "Integrated Dose Rates" was previously submitted in the Authority's phase 1 responses to NUREG 0737 Item II.B.3 (Reference 2). This revised table is being submitted to incorporate calculated background doses with Reg. Guide 1.4 source terms one (1) hour after an accident. Computer codes developed for response to NUREG 0578 Item 2.1.6.b were used as the basis for these results. A 2 hour background dose was applied for the isotopic analysis due to the approximate time sequence for obtaining and transporting the sample.

As stated in Reference 2, the times shown are estimated times the whole body or extremity is being exposed to its respective dose and not the actual time required for the particular task. Sample dose rates were obtained from Reference 3, with plant unique correction factors applied for JAF as follows:

Coolant Inventory = 0.927 X Reference Plant
Drywell and Torus Air Volume = 0.839 X Reference Plant

The FitzPatrick Plant, with these revisions, still complies with the personnel radiation exposure limits of 5 rem whole body and 75 rem to the extremities.

Note: Some of the times have been increased to allow for unanticipated equipment malfunctions.

INTEGRATED DOSE RATES (1 hr. after accident) Rev. 1

Notes: E = Extremity doses calculated at 10 cm.
W = Whole body doses calculated at 60 cm.

| 1. Liquid Sample (small volume) | Time (min) | Background (mr/hr) | Sample Dose (mr/hr) | Integrated Dose (mr) |
|---|---|---|---|---|
| (W) Sample Sink pre-op | 15 | 103 | - | 26 |
| (W) obtain sample | 45 | 103 | 93 | 147 |
| (E) handle sample | .5 | 103 | 1300 | 11.7 |
| (W) transport cask | 10 | 100 | 5.7 | 17.6 |
| (W) sample preparation | 6 | 640 | 220 | 86 |
| (E) sample preparation | 4 | 640 | 7920 | 571 |
| (W) sample analysis | 8.5 (boron) | 640 | 44 | 97.5 |
| (E) sample analysis | 2.0 (boron) | 640 | 1580 | 74 |
| (W) sample analysis | 20 (isotopic) | 400* | -- | 133 |
| 2. Gas Sample (Drywell atm.) | | | | |
| (W) Sample sink pre-op | 15 | 103 | | 26 |
| (W) obtain sample | 30 | 103 | 302 | 203 |
| (W) handle sample | 1 | 103 | 344 | 7.5 |
| (W) transport cask | 10 | 100 | 45.3 | 24.2 |
| (E) transport cask | 10 | 100 | 20,100 | 3366.7 |
| (W) sample preparation | 6 | 640 | 520 | 116 |
| (E) sample preparation | 4 | 640 | 20,100 | 1383 |
| (W) sample analysis | 30 (Hydrogen) | 640 | 18.5 | 329.3 |
| (E) sample analysis | 1.5 (Hydrogen) | 640 | 663 | 32.6 |
| (W) sample analysis | 20 (isotopic) | 400* | -- | 133 |

Total integrated whole body dose for liquid sample = 507.1 mr.
Total integrated whole body dose for gas sample = 839.0 mr.

Total integrated whole body dose for samples = 1346.1 mr.

Total integrated extremity dose for liquid samples = 656.7 mr.
Total integrated extremity dose for gas samples 4782.3 mr.

Total integrated extremity dose for samples = 5439.0 mr.

*assumed 2 hrs. after accident