TOLEDO EDISON

RICHARD P. CROUSE
Vice President
Nuclear
(419) 259-5221

Docket No. 50-346

License No. NPF-3

Serial No. 1066

August 14, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz
Operating Reactor Branch No. 4
Division of Operating Reactors
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

This is in response to your Mr. G. W. Rivenbark's letter dated July 19, 1984 (Log No. 1553) concerning Post-Accident Sampling System, NUREG-0737, Item II.B.3 Request for Additional Information. Attached is Toledo Edison's response for the Davis-Besse Nuclear Power Station Unit No. 1.

Very truly yours,

RPC:GAB:lah
attachments

cc: DB-1 NRC Resident Inspector

A046
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REQUEST FOR ADDITIONAL INFORMATION

NUREG-0737, ITEM II.B.3

Question 1: Identify the method used to determine dissolved hydrogen in the reactor coolant sample. State the accuracy and sensitivity of the method or instrument used and whether the sample is pressurized or unpressurized and if sampling is continuous and diluted. Also identify how long the instrument takes to obtain a steady reading.

Response: The dissolved H_2 is measured by direct measurement of the H_2 partial pressure, transmission of this electrochemically generated signal to the electronics, and subsequent conversion of this signal to concentration of H_2 in cc/kg of H_2O. The sample is pressurized, continuous, and undiluted.

Accuracy:	±2% of full scale
Sensitivity:	cc of H_2 per kg of H_2O
Response Time:	90% response to step change in <5 minutes

Question 2: Verify that samples for offsite laboratory analysis will be transported using licensed carriers. Identify the offsite laboratory or laboratories expected to be used.

Response: All radioactive materials that are sent offsite are shipped in accordance with station procedure HP 1607.01 "Shipping Radioactive Materials". It is anticipated that all post-accident samples will meet the requirements of a "Type A" quantity of radioactive material. We currently maintain purchase orders with Hittman Nuclear Corporation and Chem-Nuclear Systems, Inc., giving us access to a large inventory of radioactive material transportation containers and vehicles on a demand basis. The samples will be sent to the Babcock and Wilcox Lynchburg Research Center, Mount Athos Road, Lynchburg, Virginia for analysis. The laboratory holds a current and appropriate materials license numbered SNM-778.

Question 3: Describe the sampling point for the Containment Air Sample and verify that such a sample would be representative of the Containment Atmosphere.

Response: The emergency Containment grab sampler is installed to collect a sample from the inlet to Containment Monitor RE4597AA on which the valves can be lined up to sample the dome, the top of the secondary shield walls or the lower

region of containment. In addition to RE4597AA, a sample may be collected from RE4597BA. Noble gas, particulate, and iodine samples can be collected. With the use of shielding the special handling tools, the samples can be collected and transferred without exposing personnel to radiation levels that would exceed exposure limits.

The attainability of representative samples is based on the natural recirculation of the containment atmosphere.* In addition to the natural recirculation, the collection of representative samples is further enhanced by the four (4) different regions from which containment air can be sampled.

Question 4: Are the valves used in the PASS environmentally qualified for the post-accident sampling conditions, e.g., temperature, pressure, activity, etc., in which they must operate?

Response: The valves of the Post Accident Sample System (PASS) meet the criteria of ASME Section III, Class 2, M-601 piping class sheets or M-611 master valve list, where required. In all cases, the valves are of proper design and materials to withstand the post-accident pressure, temperature and activity levels for their specific location in the system.

Question 5: Describe the portable oxygen monitor used to determine dissolved oxygen in the reactor coolant sample. State the accuracy and sensitivity of the monitor.

Response: The portable oxygen monitor used to determine dissolved oxygen in the reactor coolant system is the Rexnord Model 340 analyzing system. The monitor is designed for direct measurement of dissolved oxygen in liquid streams. The oxygen diffuses through a gas permeable membrane which retains an electrolyte. This potassium iodide electrolyte surrounds a cathodic platinum electrode and an anodic lead electrode. The platinum reduces the oxygen entering the system and the resulting chemical reaction with the lead anode produces a current flow through the electrolyte. The current flow is measured as a function of oxygen. The driving force for the diffusion through the membrane is the partial pressure of oxygen.

The monitor consists of the Model 340 analyzer, the Model 60 probe, and a thermistor for automatic temperature compensation. The monitor has three direct output scales:

0-20 mg/l, (ppm)
0-200 ug/l, (ppb)
0-20 ug/l, (ppb)

* Reference Study, reported in NUREG/CR-0304 "Mixing of Radiolytic Hydrogen Generated in Containment".

The power supply can be 115V A.C., 230V A.C., or 4-9 volt batteries. The monitor's operation is unimpaired by hydrogen sulfide, sulfur dioxide, ammonia, hydrogen, and carbon dioxide and is interfered with by molecular chlorine only if the reactor coolant system pH is less than 5.

Accuracy: ±1% full scale
Sensitivity: ppb

Question 6: State the method used for chloride analysis at the offsite laboratory and the associated accuracy and sensitivity.

Response: The method for chloride analysis is ion-chromatograph with a minimum level detection of 5 ppb.

Accuracy: ±1, ppb
Sensitivity: ppb

Question 7: Provide the radiation exposure results from your time and motion studies of post-accident sampling and analysis. Confirm that the results include the effects of all radiation sources to which an operator would be exposed, that the exposures would be as-low-as-reasonably-achievable, and that the requirements of GDC 19 are met.

Response: Using NRC source terms, Radiation Zone Maps (Bechtel Drawings 7749-A-11 through 7749-A-17) have been prepared for areas throughout the station. The PASS is designed to be accessible and available for use during post-LOCA plant conditions including subsequent cold shutdown of the reactor thereby meeting GDC 19 Criteria. Using the values from the Radiation Zone Maps and determining the time spent in each area by walking through the collection procedure, personnel exposures have been determined as shown below:

	Personnel Exposure
Station Vent Sample	1.0 rem
Containment Atmosphere Sample	0.5 rem
Reactor Coolant Liquid	0.2 rem

Question 8: Describe the offsite laboratory method used to determine dissolved hydrogen in the reactor coolant. Provide details of the type sample and indicate if it is transported pressurized.

Response: Gas chromatograph will be used to determine dissolved H_2 with accuracies of ±20% at H_2 concentration of <0.5% and ±5% at H_2 concentration of ≧0.5%. The sample is transported pressurized in a stainless steel flask, with manual inlet/outlet valves and encased in 2½" of lead.

Question 9: Confirm there is a back-up counting facility for post-accident samples at Davis-Besse and indicate its location.

Response: Gamma spectroscopy equipment has been taken to the lobby area of the station, set up and operated. This equipment could also be set up in the Personnel Process Facility or the Emergency Operations Center.

Question 10: Describe backup provision for backup power to PASS in the event of loss of site power.

Response: Should there be a loss of power to the PASS, power could be restored using a path from the diesel generators back through the 13.8 KV busses. There is a station procedure to power nonessential unit substations from the emergency diesel generators if needed.

Question 11: Identify the method for boron analysis at the offsite laboratory and state the applicable accuracy and sensitivity.

Response: The boron content of each liquid sample will be measured over a range of 0.1 to 10,000 ppm by one of two methods:

1) Formation of a colored complex with carminic acid followed by spectrophotometric measurement of solution absorbance, or

Accuracy:	±5% from 10 ppm to 100 ppm
Sensitivity:	ppm

2) Titration of the boric acid with standardized NaOH after addition of mannitol.

Accuracy:	±5% from 0.1 ppm to 100 ppm; ±1% from 100 ppm to 10,000 ppm
Sensitivity:	ppm

Question 12: Confirm that the Post Accident Sample Analytical methods and instrumentation has been tested using a Standard Test Matrix for the type of contamination anticipated in post-accident samples. Confirm that the analytical methods and instrumentation will yield reliable results in the presence of anticipated radioactivity.

Response: A PASS test matrix verification program has been contracted to Babcock and Wilcox to perform. Initial results from the Babcock and Wilcox batch type analysis are due in mid August 1984. We currently anticipate reliable results from this testing program. Work is expected to be completed prior to startup following the 1984 refueling outage.

Question 13: Provide your procedure for estimating the extent of core damage.

Response: A procedure for estimating core damage has been prepared and was implemented on July 27, 1984.

Docket No. 50-346
License No. NPF-3
Serial No. 1066
August 14, 1984
Attachment II

T-8145

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TSC
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SECTION 1

PROCEDURE TITLE AND NUMBER
TECHNICAL SUPPORT CENTER ACTIVATION EI 1300.07

REASON FOR CHANGE

Included to incorporate Core Damage Assessment as per NRC commitment letter Serial #931.

CHANGE

See attached sheets

IS PROCEDURE REVISION REQUIRED Yes [X] No [ ] If no, this modification is valid until ______

	DATE
PREPARED BY Ron Dudel	7-26-84
APPROVED BY D. W. Briden	7/26/84
APPROVED BY W. J. O'Connor	7/26/84
SUBMITTED BY (Section Head) Judith Hirsch	7/26/84
RECOMMENDED BY (SRB Chairman) S M Quennoz	7/27/84
QA APPROVED BY (Manager of Quality Assurance) NA	—
APPROVED BY (Station Superintendent) T.D. Murray/SMQ	7/27/84

(DADS) terminals in the TSC to provide sufficient plant data for personnel to evaluate and diagnose plant conditions.

NOTE: The DADS can monitor plant transients during and following most events expected to occur during the life of the station.

6.8.2 Coordinating the assessment activities of the Technical Engineering staff.

6.9 Utilize the Radcon Operations Manager to direct radiological assessment activities during the initial stages of an emergency and to coordinate the radioactive waste and radiological controls aspects of the recovery including:

6.9.1 Keeping the Plant Operations Manager and Emergency Duty Officer informed of radcon and radwaste activities pursuant to the emergency.

6.9.2 Coordinating the activities of the Health Physics Monitoring Room portion of the Operations Support Center.

6.9.3 Relaying health physics information over the NRC Health Physics Network phone.

6.9.4 Supervising the onsite radiation surveys and survey results analysis.

6.9.5 Assuring that TSC and ECC radiological monitoring is accomplished in accordance with AD 1850.06, Radiation, Contamination and Airborne Radioactivity Monitoring during emergencies at the DBAB.

6.10 Access the condition of the reactor and essential safety-related systems to assure necessary steps are taken for protection of Station personnel and the public.

NOTE: Plant conditions requiring protective actions should be discussed with the Emergency Duty Officer for implementation.

6.11 Analyze plant conditions to determine reactor core status.

6.11.1 See attached sheet

6.12 Ensure that directives issued to the Control Room are assessed for potential adverse consequences before issuance - this includes all offsite directives from government or company management organizations.

6.11.1 The Radcon Operations Manager is responsible for coordinating the necessary manpower to perform core damage assessment utilizing Attachment 4, Core Damage Evaluation From Radiochemistry Analyses, and Attachment 5, Core Damage Evaluation From Containment Radiation Plots.

NOTE: If the values determined in attachment 4, and attachment 5 do not correlate, the Radcon Operations Manager should use his judgment in determining which values should be used.

7.2 All records generated during the operation of the TSC have been forwarded to the Emergency Planning supervisor.

7.3 All equipment and still useable supplies have been returned to their normal storage location.

8. ATTACHMENTS

The following are attachments for use with this procedure:

1. Technical Support Center Floor Plan
2. Technical Support Center Equipment List
3. Emergency Conditions Turnover Data Sheet
4. CORE DAMAGE EVALUATION FROM RADIOCHEMISTRY ANALYSES
5. CORE DAMAGE EVALUATION FROM CONTAINMENT RADIATION PLOTS

CORE DAMAGE EVALUATION FROM RADIOCHEMISTRY ANALYSES

1. Purpose

The purpose of this attachment is to evaluate the extent of core damage based on the reactor coolat activity for I-131, I-133, and Ba-140.

2. References

a. DBNPS Updated Safety Analysis Report

b. B&W Training Course on Post Accident Radiochemistry

c. Post Accident Sampling Analyses, AD 1850.04

d. Post Accident Sampling System Operation, SP 1103.00

3. Procedure

NOTE: The Radcon Operations Manager is responsible for coordinating the necessary manpower to complete the following steps and to report the results to the TSC Manager and Emergency Duty Officer.

a. Using the post accident sampling system, collect a reactor coolant sample as per SP 1103.00, Post Accident Sampling System Operation.

b. Determine I-131, I-133, and Ba-140 concentrations as per Ad 1850.04, Post Accident Sampling Analyses.

c. Decay back to the time when the reactor was shutdown for I-131, I-133, and Ba-140.

d. Calculate the percent of melted fuel using the following equation:

Percent of melted fuel = 0.002 x Ba-140

Percent of melted fuel = 0.002 x ________ = ____________

Where: Ba-140 is in uCi/gm

e. Calculate the percent of overheated fuel using the following equation:

Percent of fuel (F) = 1.74E-4 x I-133 - 3.66E-5 x I-131

Percent of fuel (F) = 1.74E-4 x _____ - 3.66E-5 x _______

Percent of fuel (F) = _______ - ________ = ____________

Where: I-131 and I-133 are in uCi/gm

f. Calculate the percent of fuel activity released from the gap using the following equation:

Percent of fuel (G) = 0.0264 x I-131 - 0.0136 x I-133

Percent of fuel (G) = 0.0264 x ______ - 0.0136 x ______

Percent of fuel (G) = _______ - _______ = ______________

Where: I-131 and I-133 are in uCi/gm

4. <u>Discussion</u>

a. Extent Of Melting

The Ba-140 concentration is based on only 10% of the activity in the fuel is released.

b. Total concentration of I-131 and I-133 in the reactor coolant system water if all the activity were released from the fuel and gap region was calculated from the following data.

Fission product activity in the fuel at 240 EFPD (from USAR):

I-131 = 7.44E+7 curies
I-133 = 1.44E+8 curies

Fission product activity in the rod gap region at 240 EFPD (from USAR):

I-131 = 9.55E+5 curies
I-133 = 2.00E+5 curies

Reactor Coolant System volume at system temperature and pressure is 83,028 gal from PP 1101.01, Section 1.1.2.3, thus:

$$\text{I-131} = \frac{(7.44E{+}7\ \overset{\text{Ci}}{\cancel{\text{cc}}})(1.0E{+}6\ \text{uCi/Ci})}{(83{,}028\ \text{gal})(3785\ \text{cc/gal})(0.713\ \text{gm/cc})}$$

I-131 = 3.32E+5 uCi/gm

FUEL

$$\text{I-133} = \frac{(1.44E{+}8\ \text{Ci})(1.0E{+}6\ \text{uCi/Ci})}{(83{,}028\ \text{gal})(3785\ \text{cc/gal})(0.713\ \text{gm/cc})}$$

I-133 = 6.43E+5 uCi/gm

$$\text{I-131} = \frac{(9.55\text{E+5 Ci})(1.0\text{E+6 uCi/Ci})}{(83{,}028\text{ gal})(3785\text{ cc/gal})(0.713\text{ gm/cc})}$$

$$\text{I-131} = 4.26\text{E+3 uCi/gm}$$

GAP

$$\text{I-133} = \frac{(2.0\text{E+5 Ci})(1.0\text{E+6 uCi/Ci})}{(83{,}028\text{ gal})(3785\text{ cc/gal})(0.713\text{ gm/cc})}$$

$$\text{I-133} = 8.93\text{E+2 uCi/gm}$$

c. Derivation of the equation to calculate the percent of failed fuel due to the release of activity from overheated fuel.

1% of I-131 and I-133 activity released from overheated fuel

$$\text{I-131 uCi/cc} = (3.32\text{E+3})\ F + (4.26\text{E+1})\ G$$

$$G = \frac{\text{I-131 uCi/cc}}{(4.26\text{E+1})} - \frac{(3.32\text{E+3})F}{(4.26\text{E+1})}$$

$$= \frac{\text{I-131 uCi/cc}}{(4.26\text{E+1})} - (7.79\text{E+1})F$$

$$\text{I-133 uCi/cc} = (6.43\text{E+3})\ F + (8.93\text{E+0})\ G$$

$$G = \frac{\text{I-133 uCi/cc}}{(8.93\text{E+0})} - \frac{(6.43\text{E+3})F}{(8.93\text{E+0})}$$

$$= \frac{\text{I-133 uCi/cc}}{(8.93\text{E+0})} - (7.20\text{E+2})\ F$$

$$\frac{\text{I-131 uCi/cc}}{(4.26\text{E+1})} - (7.79\text{E+1})\ F = \frac{\text{I-133 uCi/cc}}{(8.93\text{E+0})} - (7.20\text{E+2})\ F$$

$$(6.42\text{E+2})\ F = \frac{\text{I-133 uCi/cc}}{(8.93\text{E+0})} - \frac{\text{I-131 uCi/cc}}{(4.26\text{E+1})}$$

$$F = 1.74\text{E-4 (I-133 uCi/cc)} - 3.66\text{E-5 (I-131 uCi/cc)}$$

Derivation of the equation to calculate the percent of failed fuel due to the release of activity from the fuel rod gap.

1% I-131 and I-133 activity released from the gap region at 240 EFPD

$$\text{I-131 uCi/cc} = (3.32\text{E+3})\ F + (4.26\text{E+1})\ G$$

$$F = \frac{\text{I-131 uCi/cc}}{(3.32E+3)} - \frac{(4.26E+1)}{3.32E+3} G$$

$$= \frac{\text{I-131 uCi/cc}}{(3.32E+3)} - (1.28E-2)\ G$$

$$\text{I-133 uCi/cc} = (6.43E+3)\ F + (8.93E+0)\ G$$

$$F = \frac{\text{I-133 uCi/cc}}{(6.43E+3)} - \frac{(8.93E+0)}{6.43E+3}$$

$$= \frac{\text{I-133 uCi/cc}}{(6.43E+3)} - (1.39E-3)\ G$$

$$\frac{\text{I-133 uCi/cc}}{(6.43E+3)} - (1.39E-3)\ G = \frac{\text{I-131 uCi/cc}}{(3.32E+3)} - (1.28E-2)\ G$$

$$(1.14E-2)\ G = \frac{\text{I-131 uCi/cc}}{(3.32E+3)} - \frac{\text{I-133 uCi/cc}}{(6.43E+3)}$$

$$G = 0.0264\ (\text{I-131 uCi/cc}) - 0.0136\ (\text{I-133 uCi/cc})$$

CORE DAMAGE EVALUATION FROM CONTAINMENT RADIATION PLOTS

1.0 PURPOSE

The purpose of this attachment is to evaluate the extent of core damage based on the identified Containment radiation levels.

2.0 REFERENCES

2.1 DBNPS Emergency Plan

2.2 Bechtel to Toledo letter dated June 20, 1980, Log No. BT-10483

3.0 PROCEDURES

3.1 Determine time elapsed since reactor shutdown (hours).

3.2 Obtain Figure 1 and draw a vertical line, originating from the proper time value on the TIME (horizontal) axis, upward through all plots on the graph.

3.3 There will be 4 points of intersection along the vertical line drawn in step 3.2, and plots already on the Figure.

3.3.1 Label the intersection on curve "100% Fuel Melt" as point A.

3.3.2 Label the intersection on curve "100% Cladding Failure" as point B.

3.3.3 Label the intersection on curve "10% Cladding Failure" as point C.

3.3.4 Label the intersection on curve "<1% Cladding Failure" as point D.

3.4 Determine the average radiation monitor reading (mr/hr) from RE-2004, RE-2005, RE-2006 and RE-2007 (eliminate the obviously erroneous readings from the averaging such as, zero or pegged over scaled readings).

3.5 Locate this average value on the Figure (on the vertical line drawn in step 3.2). Label it as point "X".

3.6 Identify the region that point "X" is in (i.e. A-B, B-C, C-D, or below D).

3.7 Use the proper equation as listed below, for the particular region applicable, to determine the extent of core damage.

If X is in region A-B:

% of Fuel Melt (beyond 100% cladding failure) = $\left[\frac{X-B}{A-B}\right]$ x 100% = ____%

If X is in region B-C:

% of Cladding Failure = $\left[\frac{X-C}{B-C}\right]$ x 90% + 10% = ____%

If X is in region C-D:

% of Cladding Failure = $\left[\frac{X-D}{C-D}\right]$ x 9% + 1% = ____%

If X is below D:

% of Cladding Failure = < 1%

4.0 DISCUSSION

4.1 The curves represent direct readings from monitors RE-2004, RE-2005, RE-2006, and RE-2007 located at elevation 585', outside Containment (in the annulus). The calculation of monitor response did not include any particulates or iodine since the noble gases are the most significant contributors to dose rate in the Containment. At the worst, neglecting the particulates adds a slight amount of conservatism since the actual presence of particulates would result in a higher monitor reading.

4.2 The curves account for the finite containment volume seen by the detector, but do not account for any monitor's physical or shielding characteristics or calibration uncertainties.

4.3 The curves assume that only airborne noble gases are significant. Sprays, if used, would make the iodine and any particulate contribution insignificant. However, particulate plateout on surfaces and direct shine doses from components may make the readings unreliable.

4.4 Curve uncertainties are on the order of a factor of 2 to 5.

4.5 When referring to the Percent (%) Fuel Inventory Released, 100% fuel inventory is equal to 100% noble gas.

4.6 The curves are of theoretical gross gamma dose rates versus time and are given for a range of potential source terms. They correlate to:

Curve No.	% Fuel Inventory Released	% Fuel Melt	% Cladding Failure	Approximate Source and Damage Estimate
1	100	100	100	100% Regulatory Guide 1.4
2	10	0	100	10% Regulatory Guide 1.4 or 100% NRC Gap Activity per Regulatory Guide 1.25
3	1	0	10	1% Regulatory Guide 1.4 or 10% NRC Gap Activity per Regulatory Guide 1.25
4	0	0	<1	100% Coolant Release

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Attachment E

FIGURE 1 (REV. 1)

CONTAINMENT RADIATION MONITOR READINGS FOR RE-2004, RE-2005, RE-2006 AND RE-2007 VS. TIME FOLLOWING AN ACCIDENT FOR RELEASES OF VARIOUS AMOUNTS OF CORE ACTIVITY

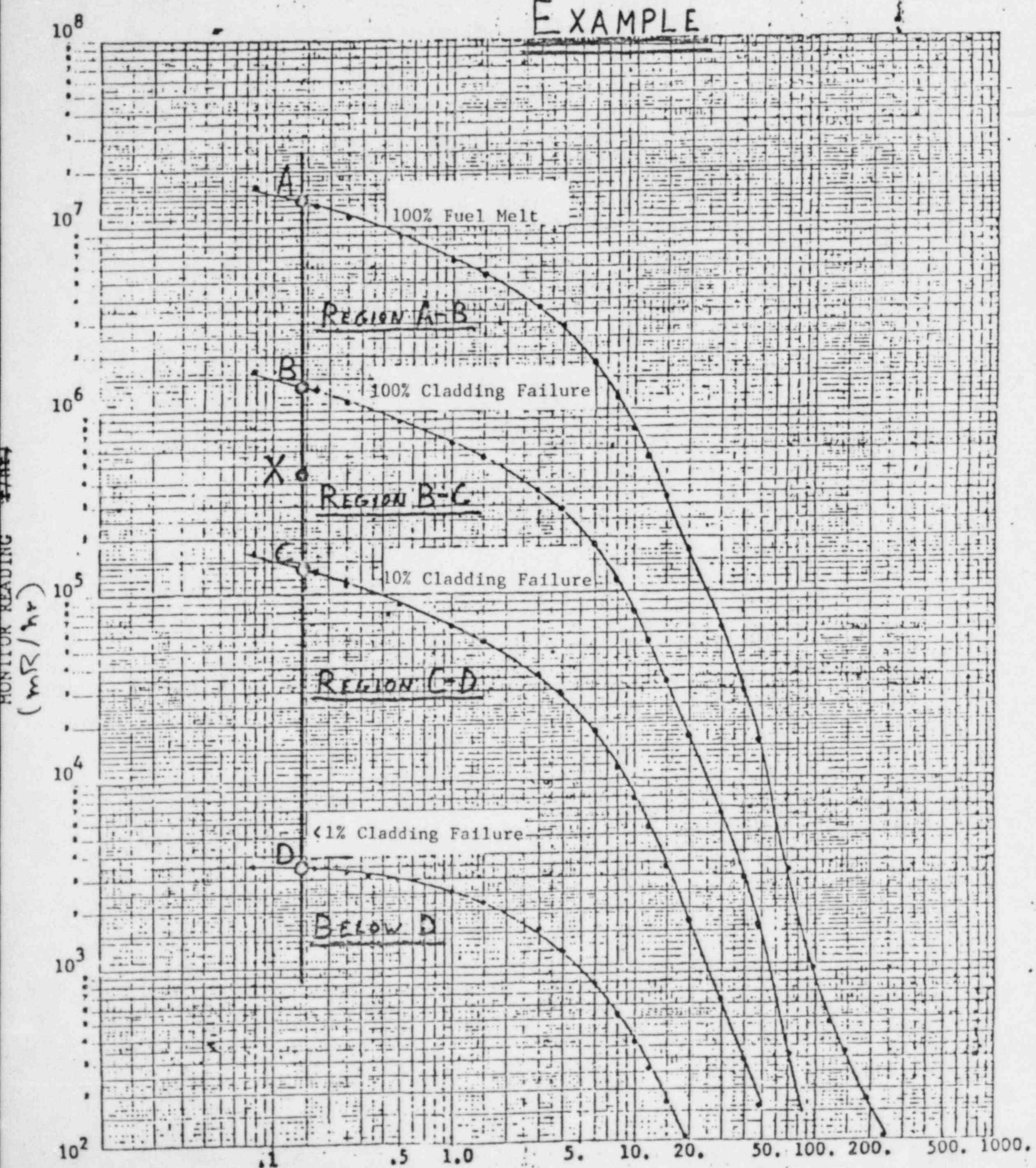
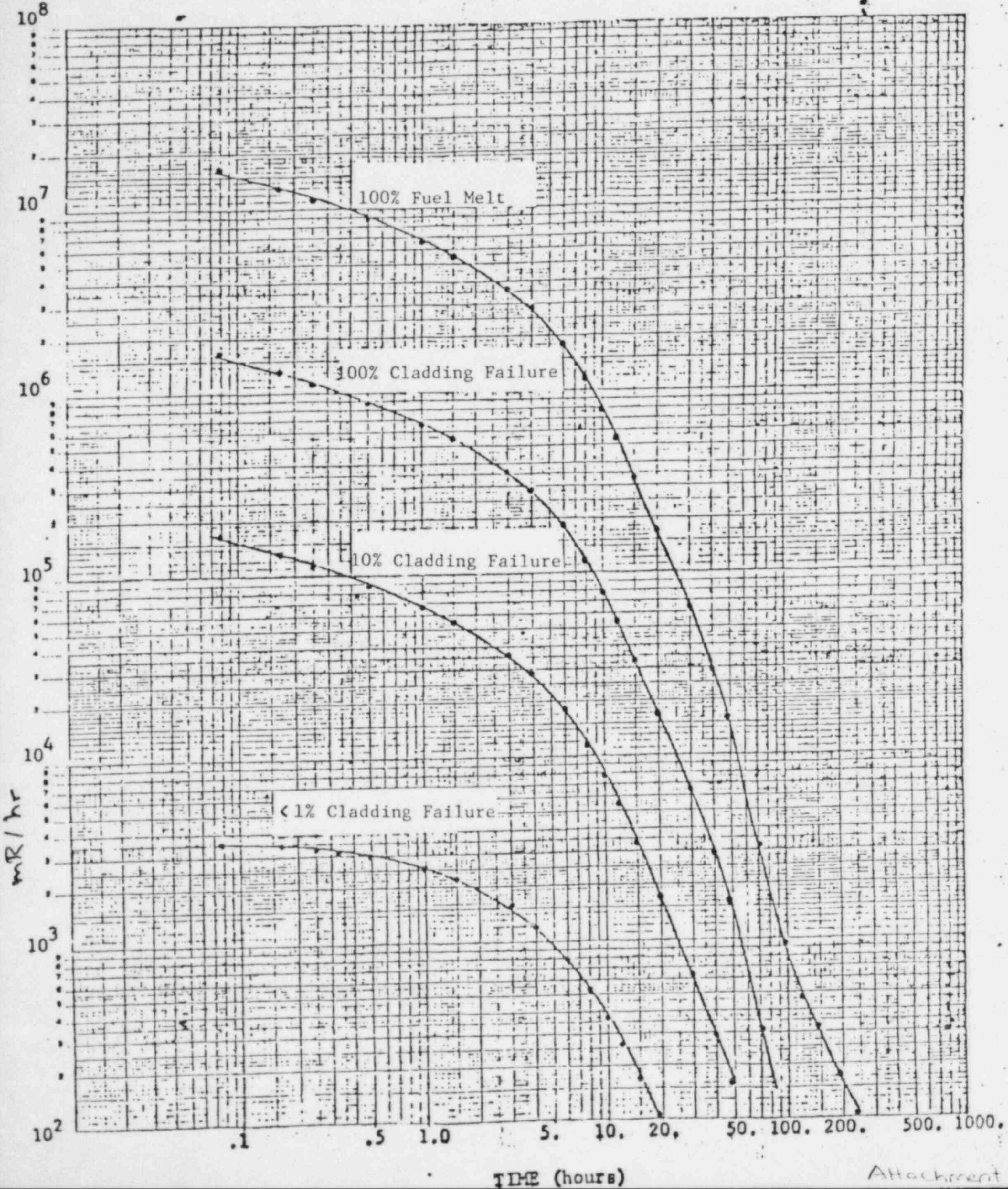


FIGURE 1 (REV. 1)

CONTAINMENT RADIATION MONITOR READINGS FOR RE-2004, RE-2005, RE-2006 AND RE-2007 VS. TIME FOLLOWING AN ACCIDENT FOR RELEASES OF VARIOUS AMOUNTS OF CORE ACTIVITY



Attachment 5