List of Enclosures

Information Provided by Department of Energy (DOE) and its Contractors on Three Mile Island.

1. Memorandum, DOE Albuquerque Operations, Sandia Area Office, Gil E. Cordova, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/24/79. Assistance provided by Sandia Laboratories. (Contact: Gil E. Cordova, Sandia Area Office, P.O. Box 5400, Albuquerque, NM 87115)

   5

2. Memorandum, DOE Chicago Operations, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 6/6/79. Assistance provided by Chicago Operations and contractors. (Contact: Edward J. Jascewsky, Chicago Operations Office, 9800 South Cass Avenue, Argonne, IL 60439)

   6

3. Teletype, DOE Dayton Area Office, Harry V. Hill, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/24/79. Assistance provided by Mound Laboratory and Monsanto Research Corporation. (Contact: Edward J. Jascewsky, Chicago Operations Office, 9800 South Cass Avenue, Argonne, IL 60439)

4. Memorandum, DOE Idaho Operations Office, Charles E. Williams to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/25/79. Assistance provided by DOE Idaho Operations Office and contractors. (Contact: C. E. Williams, Idaho Operations Office, 550 2nd Street, Idaho Falls, ID 83401)

   4

5. Teletype, DOE Nevada Operations Office, Mahlon E. Gates to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/25/79. Assistance provided by Nevada Operations and its contractors to Emergency Action Coordination Team (EACT). (Contact: Mahlon E. Gates, Nevada Operations Office, P. O. Box 14100, Las Vegas, NV 89114)

6. Memorandum, DOE Oak Ridge Operations Office, Joseph A Lenhard, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/25/79. Assistance provided by Oak Ridge Operations and Oak Ridge National Laboratory. (Contact: R. L. Egli, Oak Ridge Operations Office, P. O. Box E, Oak Ridge, TN 37830)

   2

7. Teletype, DOE Richland Operations Office, [illegible] Standerfer, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/25/79. Assistance provided by Richland Operations Office, Pacific Northwest Laboratory and Hanford Engineering Development Laboratory. (Contact: J. D. White, Richland Operations Office, 825 Jadwin Avenue, P. O. Box 550, Richland, WA 99352)

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8. Teletype, Lawrence Livermore Laboratory to DOE, Office of Nuclear Energy Programs, Robert Ferguson, 5/23/79.
Assistance provided by Lawrence Livermore Laboratory.
(Contact: Joe LaGrone, San Francisco Operations Office, 1333 Broadway, Oakland, CA 94612)

9. Memoranda, DOE Savannah River Operations, N. Stetson, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/25/79 and
3 4/25/79.
Assistance provided by Savannah River Laboratory and Savannah River Plant of E.I. duPont de Nemours and Company.
(Contact: Clifford Webb, Savannah River Operations Office, P. O. Box A, Aiken, SC 29801)

10. Memorandum, DOE Division of Naval Reactors, H. G. Rickover to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/24/79.
1 Assistance provided by DOE Division of Naval Reactors, Pittsburgh Naval Reactors Office, Schenectady Naval Reactors Office, Bettis Atomic Power Laboratory and the Knolls Atomic Power Laboratory.
(Contact: H. G. Rickover, Division of Naval Reactors, U.S. Department of Energy, Washington, D.C. 20545)

11. Memorandum, DOE Office of Nuclear Waste Management, Sheldon Meyers, to DOE Office of Nuclear Energy Programs, Robert Ferguson, 5/23/79.
Assistance provided by Naval Research Laboratory.
(Contact: Sheldon Meyers, Office of Nuclear Waste Management, U.S. Department of Energy, Washington, D.C. 20545)

12. List of titles of TMI related activities at Los Alamos Scientific
8 Laboratory (LASL) and Sandia supplied by DOE Albuquerque Operations Office (telephone conversation with James Morley, Special Programs Division, AL, 6/15/79). (Contact: J. A. Morley, Albuquerque Operations Office, P. O. Box 5400, Albuquerque, NM 87115)

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Department of Energy
Washington, D.C. 20545

May 24, 1979
NR:D:HGR R#76

Robert L. Ferguson
Program Director for Nuclear Energy

THREE MILE ISLAND EMERGENCY RESPONSE—INFORMATION REQUESTED BY THE PRESIDENT'S COMMISSION; FORWARDING OF

The Naval Reactors program provided emergency response assistance to the Commonwealth of Pennsylvania, the Nuclear Regulatory Commission and General Public Utilities following the March 28, 1979 reactor accident at the Three Mile Island plant. Department of Energy message R151741Z May 79 requested information concerning this assistance for transmittal to the President's Commission on Three Mile Island.

A major part of the Naval Reactors program support to the Commonwealth of Pennsylvania was provided by the Pittsburgh Naval Reactors Office, the Schnectady Naval Reactors Office, the Bettis Atomic Power Laboratory and the Knolls Atomic Power Laboratory through the Emergency Action Coordination Team function. Information concerning this assistance has not been included in this letter in accordance with the guidelines of the above request which states that this information would be provided separately.

Additional assistance provided by the Naval Reactor program included performing radiochemical analyses as detailed in enclosure (1), consultation on radiolysis considerations as reported in enclosure (2), temporary loan of 6 radiological control supervisors to assist General Public Utilities in their initial radiological recovery operations, and consultation with the Interagency Dose Assessment Group composed of representatives from the Nuclear Regulatory Commission, the Environmental Protection Agency, and the Department of Health Education and Welfare.

If additional information or details are required, I would be happy to cooperate further.

H G Rickover
H. G. RICKOVER
Director, Division of
Naval Reactors

Enclosure

Copy to:
R. J. Catlin, EV-10
Major General, J. K. Bratton, DP-20

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From TECHNICAL DIRECTOR

WIN

Date MAY 21, 1979

Subject

THREE MILE ISLAND UNIT 2 ASSISTANCE

To File

This memorandum summarizes the assistance provided to the Nuclear Regulatory Commission during the Three Mile Island Unit 2 accident. This does not include assistance provided to the Emergency Action Coordination Team that is reported elsewhere.

I. Analyses were made on several reactor coolant and containment gas samples. These included:

   a. Reactor coolant sample #1 - Sampled March 29, 1979

   b. Containment gas sample #1 - Sampled March 31, 1979

   c. Containment gas sample #2 - Sampled April 2, 1979

   d. Waste gas tank sample #1 - Sampled April 2, 1979

   e. Reactor coolant sample #2 - Sampled April 10, 1979

   The most extensive analyses were performed on reactor coolant sample #1 for selected fission products as well as for core structural materials. The results of all analyses performed are appended to this memorandum as Attachment 1.

II. Evaluations of the initial coolant and containment gas samples were made to estimate the extent of damage to the reactor core.

   Based on the results of the radiochemical analyses as well as data presented in the Final Safety Analysis Report of Three Mile Island Unit 2, preliminary conclusions were that a large fraction of the fuel rods experienced loss of cladding integrity, 10 to 40% of the fuel experienced high temperatures (in excess of 3200°F), the fuel did not experience temperatures in excess of the 5000°F melting temperature and melting of core structurals and control rods did not occur.

Enclosure (1)
R#76

Discard Date

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III. Consultation was provided on the adequacy of oxygen recombination in the reactor coolant environment.

Based on calculations of generation rates of oxygen from radiolysis and upon estimated hydrogen concentration in the coolant, it was concluded that oxygen formed would recombine with hydrogen.

IV. Assistance was also provided in obtaining needed equipment for recovery operations at Three Mile Island.

Approximately 8 tons of lead bricks for shielding were delivered to Harrisburg via Pittsburgh area Air Force Reserve aircraft. In addition, flame arrestors were procured from the Kent Manufacturing Company of Glen Burrie, Maryland and delivered to Three Mile Island.

V. The major portion of this work, chemistry analyses, was performed by personnel of the Development Laboratories Operations and Materials Technology activities of the Bettis Laboratory. Personnel from many other projects and activities assisted as required for technical and administrative support.

The information developed by the Laboratory was provided to the following Nuclear Regulatory Commission personnel; V. Stello, G. Gibson, M. Barrett and S. Bland, et al.

JWoodruff
D. J. Woodruff
Technical Director

Attachment 1

TABLE I TMI-2 Reactor Coolant Sample #1
Sampling Date: March 29, 1979
All radioactivities corrected
for decay to 1200 March 30, 1979

	$T_{1/2}$ Used	µCi/ml at 1200 3/30/79	
I-133	20.8 h	6750 (1)	
I-131	8.04 d	13500 (1)	
Cs-136	13.1 d	176 (2)	
Cs-134	2.06 y	63 (2)	
Cs-137	30.2 y	275 (2)	
Sr-89, 90	-	5.4 (2)	
Ba-140	12.79 d	21.2 (2)	
Te-132	78 h	203 (2)	
Mo-99	66.02 h	1530 (2)	
Ru-103	39.4 d	0.72 (2)	
Ru-106	368 d	0.36 (2)	
Ce-141	32.5 d	0.37 (2)	
Ce-144	285 d	0.38 (2)	
Rb-86	18.65 d	< 185 (2)	MDA (6)
In-114m	49.5 d	< 0.17 (2)	MDA
Cd-115m	44.6 d	< 1.4 (1)	MDA
Ag-110m	252 d	< 0.022 (2)	MDA
Zr-95	64 d	< 0.54 (2)	MDA
Mn-54	312 d	< 0.037 (2)	MDA
Fe-59	44.6 d	< 0.034 (2)	
Co-60	5.27 y	< 0.30 (2)	MDA

Others

Gross α (3)	$(7.2 \pm 2.4) \times 10^{-4}$ µCi/ml (no decay correction)
Total Uranium (4)	2.6 µg/ml
Total Boron (5)	1750 ppm

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Revised April 30, 1979

TABLE II TMI-2 Containment Gas Sample #1
Sampling Date: 0700 March 31, 1979

	μCi/ml(1) decay corrected to 0700 March 31, 1979
Xe-133 g	743
Xe-133 m	8.0
Xe-135	3.5
I-131	0.025
I-133	< 0.01

TABLE III TMI-2 Containment Gas Sample #2
Sampling Date: 1030 April 2, 1979

	μCi/ml(1) on April 3, 1979 Aliquot 1	Aliquot 2	Average	
Xe-133 g	28.6	17.7	23.2	
Xe-133 m	0.33	0.21	0.27	
Xe-135	0.01	0.0023	0.0062	
I-131	0.0074	0.012	0.0097	
I-133	< 0.0061	< 0.0061	< 0.0061	MDA(6)
Cs-137	< 0.0054	< 0.0054	< 0.0054	MDA(6)

Water in gas samples, μCi/ml decay corrected to 1030 April 2, 1979

Cs-136	0.028(2)	
Cs-134	0.014(2)	
Cs-137	0.055(2)	
I-131	3.9(1)	
Gross α	$< 9.2 \times 10^{-8}$	MDA(6)

Others(2): N_2 79.0%, O_2 21.0%, H_2 < 0.1%

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TABLE IV TMI-2 Waste Gas Tank Sample #1
Sample Date: April 2, 1979

	μCi/ml on April 3, 1979			
	Aliquot 1	Aliquot 2	Average	
Xe-133 g	4200	6100	5200	
Xe-133 m	43	63	53	
Xe-135	1.6	2.6	2.1	
I-131	0.012	0.16	0.09	
I-133	< 0.14	< 0.15	< 0.15	MDA[6]
Cs-137	< 0.1	< 0.1	< 0.1	MDA[6]

Others[7]: N_2 79.2%, O_2 13.2%, H_2 11.4%;
Hydrocarbons: CH_4 0.04 mole percent
(ethane, propane and butane detected and all
were $<$ 0.01 mole percent each)

TABLE V TMI-2 Reactor Coolant Sample #2
Sampling Date: April 10, 1979
All radioactivities corrected for decay to 1200 March 30, 1979

	µCi/ml at 1200 March 30, 1979 Note (1)		Note (2)	
I-131	24000		-	
Cs-136	210		180	
Cs-134	75		72	
Cs-137	340		290	
Ba-140	420		390	
La-140 (decay corrected with parent $T_{1/2}$)	270		-	
Xe 133 g	< 300	MDA(6)		
Sr-89, 90	-		730	
Mo-99			2300	
Ru-103			0.030	
Ru-106			< 0.083	MDA(6)
Te-132			19	

Gross α(3)	1.3×10^{-3} µCi/ml (no decay correction)
Gross β(3)	4.0×10^{-3} µCi/ml (no decay correction)
Total Uranium(4)	1.4 µg/ml
Total Boron(5)	2400 ± 100 ppm (duplicate analyses)
pH	7.65

NOTES

(Applies to Tables I thru V)

(1) Direct GeLi spectrum analysis of aliquot

(2) Radiochemical separation

(3) Direct alpha or beta counting

(4) Mass spectrometry

(5) Quinalizarin colorimetric analysis

(6) MDA is minimum detectable activity where

$$\text{MDC} = \text{Minimum Detectable Counts} = 1.645\sqrt{2\left(\frac{\text{Total Count Rate}}{\text{Counting Time}}\right)}$$

$$\text{and} \quad \text{MDA} = \frac{\text{MDC x decay factor x yield factor}}{\text{Counting efficiency x dilution factor x aliquot volume}}$$

(7) Gas chromatography

TABLE VI

Typical Particle Analyses of Filter Paper Residue From Three-Mile Island Samples*

	Si	Fe	Mg	S	Cl	K	Ca	P	Na	Al	Ti	Mn	Cu	Zn	Sn	Cr	Ni	Pb
	Weight Percent																	
(A) WG-1 **																		
(1) Long, Black Fibers	54	.5	--	7	27	11	1	--	--	--	--	--	--	--	--	--	--	--
	26	.1	--	4	41	20	8	--	--	--	.4	--	--	--	--	--	--	--
(2) Long Yellowish-White Fibers as Separated	25	5	--	7	33	3	19	--	--	7	--	.3	--	--	--	--	--	--
	19	--	--	10	42	--	19	--	--	--	--	3	--	--	--	--	6	--
	30	--	--	5	50	--	8	--	--	--	--	--	--	5	--	--	1	--
(3) Yellow-white Fiber Still on Collection Filter	27	30	--	6	11	4	20	1	--	--	.6	--	--	--	--	--	--	--
(4) Y-W Fiber on Tan Spot	9	8	--	3	51	6	19	--	--	--	.7	1	--	--	--	2	--	--
(5) Particles on Y-W Fiber	25	27	--	6	8	4	18	1	--	6	1	.2	--	3	--	--	--	1
(6) Flat Flake	39	6	--	.3	.1	17	--	.5	--	36	2	--	--	.3	--	--	--	--
(7) Small Round Particles	50	28	--	4	1	3	4	2	--	--	1	2	3	1	--	--	--	--
	45	22	1	3	.4	1	14	1	--	9	.6	1	.6	.5	--	.4	.1	1
(8) Small Particle Clump	3	.7	--	5	2	--	4	--	14	1	--	--	58	2	10	--	1	.2
(B) CG-2 ***																		
(1) Fiber	82	1	--	--	--	--	--	4	--	9	--	--	--	--	--	--	4	--
(2) Angular Particles	95	.1	--	--	.2	--	--	1	2	1	--	--	--	.2	1	--	--	--
	96	--	--	--	--	--	--	1	1	1	--	--	--	--	.6	.1	--	--
(3) Particle Clump	36	38	--	2	.2	4	2	.7	.2	15	2	--	.4	.2	--	.5	--	--

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ATTACHMENT 1

TABLE VI

(con't)

	Si	Fe	Mg	S	Cl	K	Ca	P	Na	Al	Ti	Mn	Cu	Zn	Sn	Cr	Ni	Pb
	Weight Percent																	
(C) Asbestos Standards																		
(1) Chrysotile -																		
NBS	39	1-3	32	--	--	--	--	--	< .1	~ .3	--	.5	--	--	--	--	.1	--
Bettis	46-59	.5-3	37-46	1-2	.3	--	--	3	--	--	--	--	--	--	--	--	.6	.2
(2) Amosite -																		
NBS	50	15-28	11	--	--	--	--	--	.1	~ .3	--	1.5	--	--	--	--	--	--
Bettis	44-65	6-44	6-20	1-3	--	--	--	2	--	--	--	1	--	--	--	--	.2	--
(3) Anthophyllite																		
NBS	58	2-4	24	--	--	--	--	--	.1	~ .7	--	1	--	--	--	.1	1	--
Bettis	68	5	23	3	--	--	--	--	--	--	--	--	--	--	--	--	--	--
(4) Crocidolite																		
NBS	49	15-27	4	--	--	--	--	--	5	~ .5	--	.1	--	--	--	--	--	--
Bettis	60	18	16	4	2	--	--	--	--	--	--	--	--	--	--	--	--	--

* Energy Dispersive X-ray Microanalyses May Show Large Ranges for Mixtures of Materials of Distinct, but of Various, Compositions.

** WG-1 means Waste Gas Tank Sample #1.

*** CG-2 means Containment Gas Sample #2.

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KAPL Evaluation of Radiolysis Associated With The Three Mile Island Unit-2 Incident

By

J. C. Conine
D. J. Krommenhoek
D. Emanuel Logan

May 1979

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Enclosure (2)
R#76

Table of Contents

POOR ORIGINAL

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I. INTRODUCTION

Subsequent to an incident on March 28, 1979 at Unit-2 of the Three Mile Island Power Plant, the Nuclear Regulatory Commission (NRC) requested Department of Energy (DOE) assistance to assess the accumulation and explosive potential of a bubble within the reactor vessel.

This KAPL document presents the chronology of KAPL radiolysis participation with NRC in the following a[illegible].

1. Combustible limits for oxygen/hydrogen mixtures
2. Hydrogen permeation from reactor vessel
3. Nature of initial non-condensibles in reactor vessel
4. Oxygen buildup rate in reactor vessel
5. Reactor radiolysis in the event of a main coolant pump failure
6. Dissolved hydrogen reduction through the addition of hydrogen peroxide
7. Radiolysis within the containment building

These KAPL calculations required many assumptions relative to plant parameters and were performed with conservative values of input parameters to maximize hydrogen and oxygen production.

II. CHRONOLOGY

KAPL assistance on radiolysis evaluation associated with the Three Mile Island Unit-2 Incident was initiated on Saturday, March 31, 1979 pursuant to a request from NR (Telecon, W. P. Engel, NR, and D. J. Kro[illegible], KAPL 3/31/79). This assistance was provided at a time when emergency conditions prevailed, and direct communication was established between KAPL and the NRC per NR/KAPL agreement (Telecon, R. [illegible], NR, and L. R. [illegible], KAPL 3/31/79).

A summary of the NRC questions, time of question, NRC contact, and KAPL response is given below.

1. What are the explosion and detonation pressures for high pressure oxygen, hydrogen, water vapor mixtures? (3/31/79, R. Tedesco, NRC)

   Results from KAPL sponsored [illegible] Test Station and Mine Safety Appliance explosion tests were transmitted to NRC. Information relative to the TMI-2 reactor vessel conditions of 1000-1500 [illegible] non-condensible gas and steam bubble at 1000 psia and 300°F was provided by the NRC. The combustion and detonation limits for mixtures of hydrogen, oxygen, and water vapor in closed containers are shown on Figure 1.

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2. What is the composition of gas in the bubble in the reactor vessel? (3/31/79, R. Tedesco, NRC)

At the time this question was received, NRC had one estimate of 96% H_2, and 4% O_2, or very close to the combustion limits. NRC information indicated that the bubble was formed after reactor scram. KAPL provided an upper bound for the oxygen/hydrogen release rate for an assumed 0.08 watts/gm gamma heating rate (~1% power) in the liquid. The upper bound was based on maximum theoretical release rate in a 2000 ft^3 of liquid and gave ~50 ft^3 of hydrogen and oxygen per day at 1000 psia and 300°F. Thus, it was not possible to produce a large non-condensible bubble from radiolysis, and furthermore, the mixture was evaluated as not combustible. The bubble was primarily composed of hydrogen, with less than 1% oxygen.

KAPL experience with hydrogen/oxygen mixtures with strong dissolved gas return to the gamma flux region showed a strong hydrogen/oxygen recombination mechanism. KAPL stated that continued operation of the main coolant pumps is very favorable to promote recombination of any oxygen that may have formed during the incident. Additional calculations on oxygen release were reported to NRC on 4/2/79.

3. What is the rate of non-condensible bubble growth due to radiolysis for a 0.3% power decay heat level? (3/31/79, R. Tedesco, NRC)

KAPL calculated that the rate of reactor vessel bubble growth is less than 10 ft^3/day. Additional best estimate calculations giving less than 1.0 ft^3/day were provided to NRC on 4/2/79. All calculations indicated that radiolysis would represent an insignificant contribution to the bubble growth.

4. What are the consequences of venting the bubble from reactor vessel to the containment building? (3/31/79, R. Tedesco, NRC)

KAPL calculations indicated the following:

a) The bubble in the reactor vessel was not combustible.
b) Oxygen in the bubble will recombine with hydrogen in a gamma flux region.
c) Transfer of gas from reactor vessel to containment will r... containment hydrogen concentration by ~3.0%. Since measured containment hydrogen was given as 1.3 and 1.7%, this would exceed the 4% hydrogen upward flame propagation limit.
d) Equilibrium dissolved hydrogen in primary coolant was calculated as 1700-2000 scc/kg for 1000 psi hydrogen at 300°F.
e) The reactor vessel bubble will not grow due to radiolysis, however, it will expand on depressurization.
f) A slow degas of the primary coolant (as opposed to a more rapid depressurization) will result in significant reduction in reactor vessel bubble size with the high calculated primary coolant dissolved concentrations.

5. What is the extent of radiolysis in containment building? (4/1/79, R. Tedesco, NRC)

Based on the results of one primary coolant sample used to establish energy deposition in the water and an estimated 100,000 gallons of water in the containment the maximum hydrogen and oxygen production was calculated to be 10 std ft^3/day in the 2,000,000 ft^3 containment building.

6. Is reactor vessel hydrogen concentration being reduced due to hydrogen permeation? (4/1/79, R. Tedesco, NRC)

KAPL data at 300°F, and applied to the 300°F assumed temperature for the reactor pressure vessel boudnary showed hydrogen permeation to be insignificant.

7. Results of additional calculations in response to question number 2 and 3 were reported on 4/2/79 to Butler, NRC.

The calculations substantiated initial conclusions that the reactor vessel bubble was composed of very little oxygen from radiolysis. KAPL stated that some hydrogen (greater than 1.0 scc/kg) should be maintained in the coolant to suppress subsequent oxygen production. With NRC estimates of ~10-15% of the core in boiling, it is necessary to maintain hydrogen excess to suppress oxygen production.

8. What are flammability limits for use in hydrogen, oxygen mixtures at 1000 psia and 300°F? (4/2/79, W. A. Riehl, Technical Advisory Support at TMI)

KAPL provided the same oxygen limits as given in question 1, and also limits for hydrogen. A conservative set of flammability limits in the absence of water vapor were identified as follows:

	% By Volume	
	Hydrogen (Remainder Air)	Oxygen (Remainder Hydrogen)
Upward Flame Propagation	4	6
Stable Burning	8	6
Detonation	15	12

9. Did an explosion or detonation occur in containment, and what are recombination mechanisms in containment? NRC provided information on containment pressure increase (28 psi) and containment gas samples as follows: (4/2-3/79, [illegible], NRC)

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Sample Results

		N_2 + Others	O_2	H_2
	Air	79	21	0
2330 4/1/79		78.9	18.9	2.3
0630 4/2/79		79.9	17.6	2.4

POOR ORIGINAL

KAPL supplied the following:

a) The gamma, and beta induced recombination in the containment building at 6000 r/hr reduces an initial hydrogen concentration of 2% by 0.1% per day.
b) The pressure increase of 28 psi in the containment building, the decrease in containment pressure, and the decrease in oxygen concentration are consistent with combustion in containment.

10. What is the effect of paints, and organics in the containment on radiolysis? (4/3/79, Butler, NRC)

KAPL has no experience in this area.

11. What is dependence of radiolysis on temperature? (4/4/79, Butler, NRC)

Lower temperature increases oxygen production, but still yields small release rates for present assumed reactor condition of 0.3% power decay heat. KAPL also presented results of additional calculations for 0.06%/power, no hydrogen in coolant, and 60 scc H_2/kg in coolant.

12. What is the extent of radiolysis at 10 MW for each of the following: (a) no hydrogen in coolant; (b) discharge of coolant into containment; and (c) no primary coolant flow? (4/5/79, F. Witt, NRC)

a) Total maximum hydrogen and oxygen production equals 15 std ft^3/day with 10% of core in boiling.
b) Based on coolant sample maximum hydrogen and oxygen production equals 1 std ft^3/day for 100,000 gallons of water.
c) Total maximum hydrogen and oxygen production equals 10 std ft^3/day.

13. What is maximum rate of recombination of hydrogen in 250,000 gallons of containment liquid? (4/9/79, F. Witt, NRC)

The maximum hydrogen recombination in liquid is calculated as 3? std ft^3/day based on solubility of [illegible] gas. The total hydrogen in containment gas space equalled 40,000 std ft^3.

14. What is a recommended hydrogen peroxide addition to the primary coolant to reduce the dissolved hydrogen concentration and what are dissolved hydrogen recombination rates in the presence of hydrogen peroxide? (4/9/79, F. Witt, NRC)

KAPL did not provide a recommendation or information on this question. NRC provided information on previous experience on the effects of hydrogen peroxide on shutdown chemistry transients in pressurized water reactors (EPRI NP 692, dated 4/78). KAPL and NRC agreed that calculations could not be performed without better knowledge of [illegible] concentrations and coolant activity and impurity levels.

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III. BACKGROUND

Given the reactor conditions, there were two possible sources for the bubble, a zirconium-water reaction or radiolysis. If the bubble formation was a consequence of a zirconium-water reaction then the bubble would be essentially 100% hydrogen, with no immediate potential for explosion. However, the potential would gradually increase if a net decomposition radiolytic condition existed within the core; oxygen would then be added to the bubble. On the other hand, the potential for explosion would be great if the bubble resulted from radiolysis. In this case, the bubble would be 67% hydrogen and 33% oxygen. Additionally, conditions within the containment building were also of concern since primary coolant had been released into the building and hydrogen had effervesced from the liquid.

Whenever water is exposed to a radiation field it dissociates into a number of chemical species, one of which is the hydrogen molecule. The resulting concentration of the various species is a consequence of the nature (e.g., neutron, gamma, etc.) and strength of the radiation flux. These species in turn react with each other in numerous chemical combinations. The particular reactions that are realized in this chemical kinetics depends primarily on the relative concentrations of the species. Some of the possible reactions further produce hydrogen molecules, while others produce molecular oxygen.

If the thermal and hydraulic conditions in the irradiated volume are such that boiling occurs, then the dissolved molecules of hydrogen and oxygen can diffuse into the local voids and be removed from the irradiated volume by mass transfer. These gases are then free for possible accumulation within the system. Accumulations of hydrogen and oxygen gases can be combustible or even detonable. Depending upon the gas concentration, peak pressure generated during a combustion process can achieve a magnitude up to approximately 6 times the initial pressure. By comparison, detonation can produce a peak pressure well in excess of levels produced during combustion with a potential for a peak pressure approximately 50 times the initial pressure.

The accumulation of combustible mixtures of hydrogen and oxygen can be avoided by controlling the chemical kinetics to prevent the formation of molecular oxygen. This control is achieved by maintaining dissolved hydrogen above a critical concentration in the irradiated volume.

IV. ANALYSIS

Resolution of the NRC inquiries was primarily achieved through analyses with a KAPL radiolysis computer program. A basic model was constructed having an irradiated volume, a vapor region, and a primary coolant [illegible] irradiated region as shown on Figure-2 schematic. This model was [illegible] for calculations within the reactor vessel as well as some calculations within the [illegible]. It is [illegible] to note that [illegible] [illegible] no return of the non-condensable (via gas dissolution and mass transfer) to the gamma flux region is assumed.

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Since the actual conditions within the reactor were not accurately known, the calculations were purposely biased to yield worst case results. Accordingly, calculations for the initial bubble formation were performed with the entire core volume (estimated as 2000 cubic feet) in boiling, and no coolant flow. Those calculations performed after flow restoration still assumed some boiling, 10% of the core. Additionally, the stripping coefficients (mechanism for removal of the gases from solution) were evaluated at a factor of 8 times their best estimate values; thereby removing hydrogen and oxygen from solution at a higher than expected rate.

Determination of the recombination within the containment vessel was hand calculated. Likewise, a determination of the hydrogen permeation from the reactor vessel to the containment building was also hand calculated.

V. CONCLUSION

The calculations discussed in this document were performed to assess the potential for an explosion of gaseous mixtures of hydrogen and oxygen within the TMI-2 reactor. Such mixtures would have been the consequence of a radiation induced net decomposition of the reactor coolant (i.e., radiolysis). Since conditions within the reactor were not very well known, the radiolytic calculations were performed with conservative values of the significant input parameters; i.e., values which tended to produce hydrogen and oxygen.

Evaluation of the calculated result indicated no presence of a combustible mixture of hydrogen and oxygen within the reactor, at any time subsequent to the initial formation of the non-condensibles. Additionally, the potential for generating such mixtures decreases as the power decays. Furthermore, the main source of radiation in the shutdown state is gamma, which is favorable for the recombination of hydrogen and oxygen. Gamma radiation was also estimated to recombine the containment building atmospheric hydrogen at an initial rate of ~0.1% (by volume) per day (based on a concentration of 2% initial hydrogen).

VI. REFERENCES

1. General Electric Reports: NEDO-1032, by B. C. Slifer and T. G. Peterson, "Hydrogen Flammability and Burning Characteristics in BWR Containments".

2. Minnesota University, Minneapolis Dept. of Chemical Engineering: COO-1032-1, dated July 1965 by B. M. Benjamin and H. S. Isbin, "Recombination of Hydrogen and Oxygen in the Presence of Water Vapor Under the Influence of Radiation".

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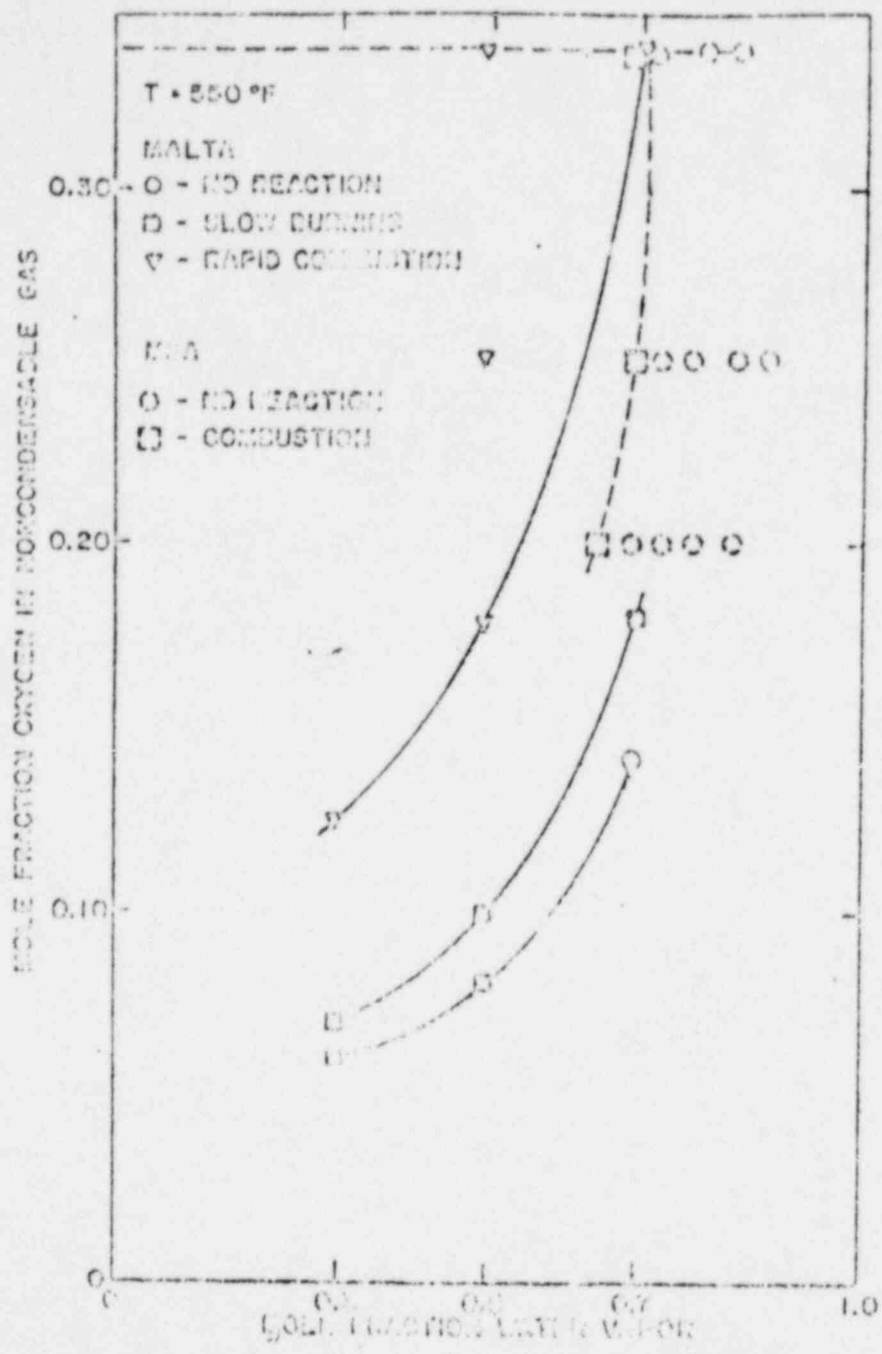

Explosion Test Results for Mixtures of Hydrogen, Oxygen, and Water Vapor in Closed Containers at 550°F.

Figure 1

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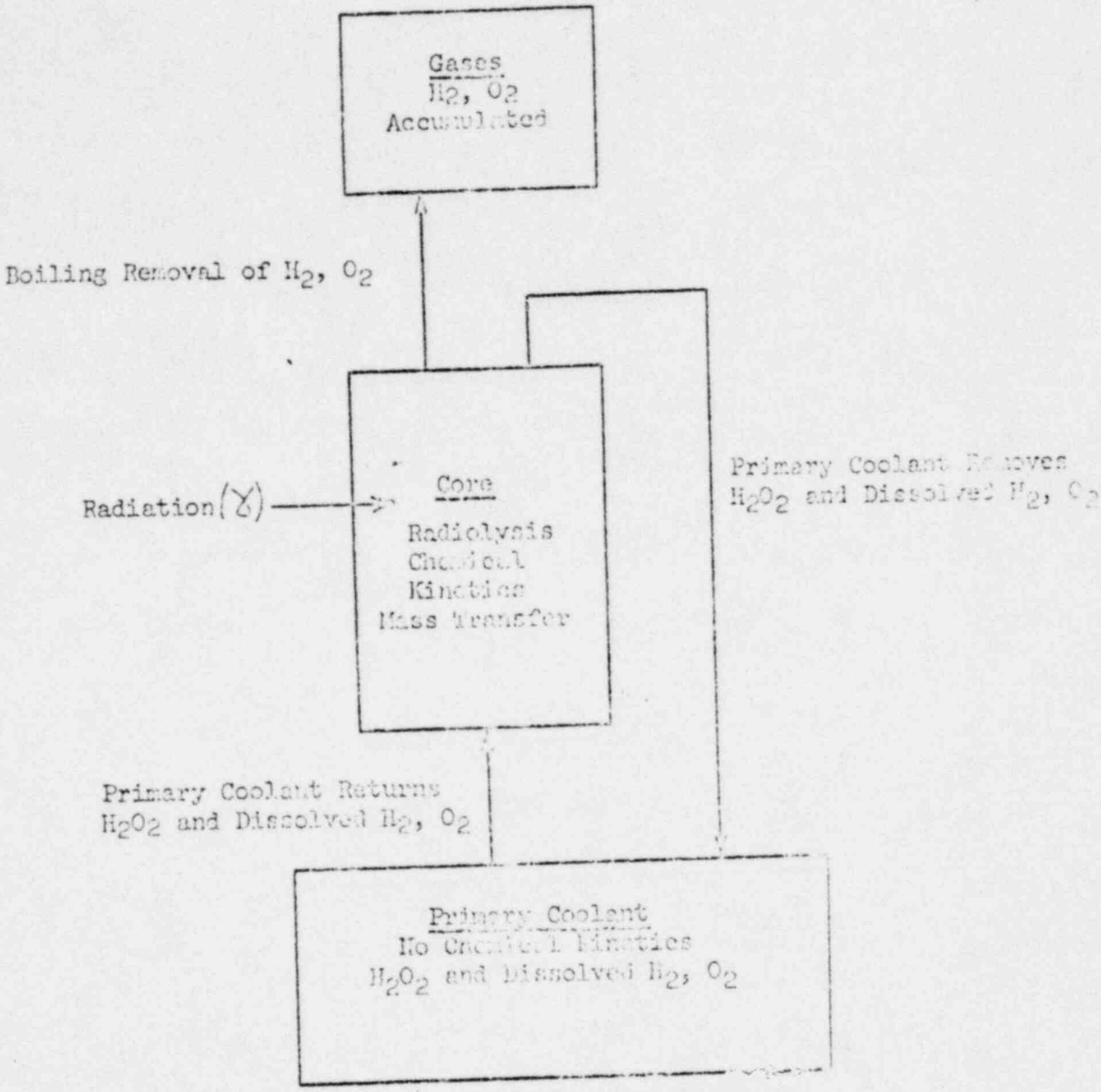

TMI-2 RADIOLYSIS MODEL

Figure 2

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(6)

Department of Energy
Oak Ridge Operations
P.O. Box E
Oak Ridge, Tennessee 37830

May 25, 1979

Robert L. Ferguson, Acting Program Director, Nuclear Energy Programs, Office of the Assistant Secretary for Energy Technology, Mail Station B-107, Germantown, Maryland

OAK RIDGE ASSISTANCE TO THREE MILE ISLAND

This memorandum is in response to your May 15, 1979, TWX, subject as above, asking for information on Oak Ridge involvement in the Three Mile Island accident.

By way of enclosures, we believe we have answered all the questions you raised in your TWX. If there is any additional information you need, please call R. L. Egli of my staff on FTS 626-0725.

ORIGINAL SIGNED BY
RICHARD ...

ER-10:RLE

Joseph A. Lenhard, Assistant Manager
for Energy Research and Development

Enclosures:
1. On-Site Information
2. On-Site Support
3. On-Site Reports
4. Off-Site Reports
5. Cost Estimate

cc w/encls:
H. Feinroth, ET-HQ, MS B-107, GTN

bcc w/encls:
B. J. Davis
R. L. Egli

As of 5/18/79

Attachment I

Chemical Engineers From ORNL Chemical Technology Division

To advise on decontamination and other radioactive material situations.

Requested by Metropolitan Edison.

Name	Organization	Departure	Return	Working For
R. E. Brooksbank	ORNL	4/1/79	4/13/79	TMI
		4/17/79	4/20/79	TMI
		5/15/79	Still on site*	TMI
O. O. Yarbro, Jr.	ORNL	4/1/79	4/14/79	TMI
J. W. Snider	ORNL	4/4/79	4/13/79	TMI
F. E. Harrington	ORNL	4/4/79	4/7/79	TMI
L. J. King	ORNL	4/12/79	4/20/79	TMI
W. A. Shannon	ORNL			
		5/8/79	5/16/79	TMI
E. D. Collins	ORNL	4/15/79	4/20/79	TMI
D. O. Campbell	ORNL	5/16/79	5/18/79	TMI

*This work is still continuing

The present contact at TMI is Benard Rushe who is managing much of the site work for Metropolitan Edison. The initial contact was Herman Diekamp of Metropolitan Edison.

Instrument Engineers From ORNL Instrumentation and Controls Division

To improve confidence level of reactor instrumentation by analyzing signal noise. This was used for determination of boiling detection gas bubble size, flow and level detection. Requested by D. Ross, NRC.

Name	Organization	Departure	Return	Working For
R. C. Kryter	ORNL	4/3/79	4/7/79	NRC
		4/21/79	4/30/79	NRC
D. N. Fry	ORNL	4/3/79	4/13/79	NRC
Jim Robinson	ORNL(Consultant)	4/7/79	4/13/79	I&C
C. M. Smith	ORNL	4/7/79	4/11/79	NRC
S. J. Ball	ORNL	4/27/79	4/30/79	NRC
W. H. Sides	ORNL	4/27/79	4/30/79	NRC
R. M. Carroll	ORNL	4/30/79	5/3/79	TMI
R. L. Shepard	ORNL	4/30/79	5/3/79	NRC

The onsite work was coordinated by the management steering group. D. Ross and V. Stello were the responsible contacts for NRC.

Public Information

Requested by NRC through DOE-PIO HQ to assist in dissemination of information to the media and public.

Name	Organization	Departure	Return
Jim Alexander	ORO	4/3/79	4/8/79

Photographic Services

Requested by NRC through DOE-PIO HQ to produce a pictorial documentation of the Three Mile Island activities.

Name	Organization	Departure	Return
J. E. Westcott	ORO(Consultant)	4/4/79	4/6/79

Primary Coolant Sample Analysis From ORNL

(Next Page)

April 12, 1979

APRIL 12 1979, UPDATE ON ORNL ACTIVITIES AT THREE MILE ISLAND

ORNL has performed some analysis on the primary coolant sample received yesterday. The results are as follows, subject to refinement, and the half-life is included for reference:

Fission Products	μci/ml	$T_{1/2}$
Mo-99	179	67h
I-131	8.3×10^{-3}	8d
I-132	21	2h
C_s-134	82	2y
C_s-136	108	13d
C_s-137	330	30y
B_a-140	290	13d
L_a-140	160	44h
Sr-89	600	52d
Sr-90	50	28y
Tritium	1.2	12y

pH-8

Boron - 3.58 mg/ml

Total uranium - in the part per billion range

Alpha - none detectable

Gross beta - 1.4×10^{-10} d/m/ml

With these results, it appears that there was minimal fuel melting because the uranium content is very low. Most of the fission products are short lived. Therefore, in several months time the primary concern will be with the cesium, stontium, and tritium which is still a bunch. The cesium isotopes will probably be the greatest contribution to the penetrating radiation problem.

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OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX X
OAK RIDGE, TENNESSEE 37830

May 23, 1979

Mr. Ron Williams
Technical Support Group
for Waste Management
Trailer 118
Metropolitan Edison Company
Three Mile Island Station
Middletown, PA 17057

Dear Mr. Williams:

The enclosed tables present data that were obtained on the initial set of three water samples submitted via P. O. 80042 from Metropolitan Edison Company. For comparison purposes results that were obtained on the original water sample (of April 11, 1979) are also listed.

The primary staff members involved in this work were Dr. Joel A. Carter (FTS 624-2447) for mass spectroscopic analysis and Dr. Juel F. Emery (FTS 626-7560) for radiochemical analyses. Please feel free to contact these men if you wish to discuss these results or the details of the analyses.

Sincerely yours,

Wilbur D. Shults

Wilbur D. Shults
Director
Analytical Chemistry Division

WDS:gv

Enclosures

CC (Enclosures):
D. O. Campbell
J. A. Carter
L. T. Corbin
J. F. Emery
L. J. King
J. A. Lenhard (ORO)
F. Mynatt
J. R. Stokely
D. Trauger
A. Zucker

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Y 4715

Table I

Analytical Results: TMI Water Samples

Parameter	Sample				
	U-I	U-IA	U-IA, Residue	U-II	Original Water, No. I
pH	7.0	6.0		7.5	8.0
Uranium, ppb	0.6±0.3	15.6±0.5		1.7±0.3	110.0
Uranium isotopic, atom %					
^{234}U	≤0.01	0.0075		≤0.01	0.021
^{235}U	0.6	0.84		0.6	2.22
^{236}U	≤0.01	0.026		<0.01	0.072
^{238}U	99.4	99.126		99.4	97.69
Plutonium, ppb	0.00021	0.0046		0.00006	0.24
Plutonium isotopic, atom %					
^{239}Pu		82.42			90.3
^{240}Pu		12.10			7.8
^{241}Pu		4.67			1.8
^{242}Pu		0.81			0.1
Gross Alpha Activity, dpm/ml	0.99±0.30	1.19±0.31		0.37±0.24	Background

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Table II

Elemental Composition in PPM
(Semi-quantitative via SSMS)

Element	Sample				
	U-I	U-IA	U-IA, Residue	U-II	Original Water
Ag	<.01	<.02		<0.1	≤0.2
Al	0.5	5	500	1	10
B	830	11,000	Major	70	3,220
Ba	0.02	0.05		0.02	
Cl	1	2	20	0.7	
Ca	10	70	100	10	≤1
Cd	<0.03	<.1		<.07	≤0.3
Co	<.01	0.05		<.01	
Cr	0.2	2	5	0.2	
Cs	<.01	<.01		<.03	8
Cu	0.05	1		0.1	
Fe	2.5	70	300	0.7	≤1
^{129}I	<.02	<.01		<.01	
^{131}I	<.06	<.02		<.1	0.9
In	<.03	<.01		<.01	≤0.05
K	5	50	100	2	0.4
^{7}Li	0.03	<.03	0.5	0.02	4.64
Mg	3	30	50	3	
Mn	0.7	10	20	0.05	0.1
Na	20	50	500	30	960
Ni	0.5	5	7	0.05	0.2
P	0.1	2	3	0.2	0.1
Rb	0.1	0.3		0.01	1
S	10	300	300	20	20
Sr	0.07	0.4	5	0.07	0.5
Te	<.05	<.05		<.05	0.5
Zn	0.5	2		0.2	0.5
Zr	<.01	<.02		<.01	≤3

Table III

Radioelement Analysis in $\mu Ci/ml$

[Effective Time for Original Water Sample: 0800; April 11, 1979.
Effective Time for U-I, U-IA, U-II: 0800; May 9, 1979.]

Isotope	U-I	U-IA	U-IA Residue	U-II	Original Water
^{54}Mn	2.30E-4	4.49E-3	8.9E-3		
^{57}Co	-	3.46E-3	5.4E-3		
^{58}Co	0.0419	1.35	2.05		≤ 3
^{60}Co	6.2E-4	0.0135	0.0316		
^{89}Sr	2.84E-2	1.77E-2		0.757	5.5E2
^{95}Nb			1.4E-3		
^{110}Ag			2.7E-3		
^{124}Sb			6.4E-3		
^{131}I	0.166	0.096	0.743	1.37	8.2E3
^{132}I					$\leq$2E1
^{134}Cs	0.0266	0.086	0.157	0.147	8.2E1
^{136}Cs	7.22E-3	4.49E-3	6.6E-3	0.045	1.1E2
^{137}Cs	0.106	0.158	0.297	0.586	3.3E2
^{140}Ba	4.49E-3	$\leq$2E-3	6.5E-3	0.180	2.4E2
^{140}La	6.57E-3	3.6E-3	5.2E-3	0.230	1.6E2
^{3}H					1.2
^{90}Sr					6.8 ± 0.25
^{99}Mo					1.8E2
^{141}Ce					≤ 1
^{144}Ce					≤ 20

Mobile Manipulator - UCCND/Y-12 ("HERMAN")

1. Assistance requested by D. Stello, NRC. Specific mission was to take primary coolant samples and other functions within its capabilities. Procedures were developed by GPU/MET ED and verified but the manipulator was not used to take a primary coolant sample due to reservations on the part of the GPU/MET ED regarding the manipulator's reliability.

2. Manipulator and crew arrived at TMI on March 31, 1979. (Semi-tractor provided by DOE with DOE drivers.) Crew returned to Oak Ridge on April 7, 1979. Manipulator was returned on May 4, 1979.

	UCCND/Y-12	DOE Drivers
Crew:	R. W. Frazier, Y-12	D. J. Lee, ORO 3/31/79 - 4/1/79
	W. L. Pankratz, Y-12	P. Warren, ORO 3/31/79 - 4/1/79
	R. E. Turner, Y-12	J. Goodman, ORO 5/2/79 - 5/4/79
	T. E. Copeland, Y-12	

3. No work performed.

4. Manipulator still available but no plans for use at TMI as of May 18, 1979.

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ON-SITE SUPPORT

Radiation Monitoring

ORNL's initial assistance at Three Mile Island was in the area of radiation monitoring. Four two-person teams conducted environmental monitoring at the site for three weeks after the accident. The teams took soil, water (standing and rain), and vegetation samples. Using a Thyac GM-Survey Meter, they took direct radiation measurements in the air and located and followed the plume from the reactor area. Samples were gamma-spectrum analyzed at the DOE control point and then sent to Brookhaven Laboratory for further radiochemical analyses. Results of counting room analyses were reported to NRC. Spectrum analysis yielded only two samples with ^{131}I, right at minimum detectable activity for the detector used; which is to say, background levels. Another ORNL team provided specialized equipment for environmental surveillance, and one staff member accompanied the Y-12 robot manipulator crew and provided health physics monitoring services for them.

(R. L. Clark, A. C. Butler, W. D. Carden, M. L. Conner, B. J. Davis, J. S. Eldridge, S. A. Hamby, W. M. Johnson, B. A. Powers, J. E. Smith)

Chemical Engineering - Liquid and Gaseous Effluents

A team of ORNL chemical engineers provided technical support to the Waste Management Group (WMG) of the TMI Recovery Team. This work began on April 1, 1979, and is continuing at various levels of effort. The original request for ORNL assistance was made by F. L. Culler of EPRI, and results of the work have been reported to NRC and GPU. The major problems addressed by the WMG immediately following the accident were: (1) stopping the release of ^{131}I to the environment and (2) providing adequate storage capacity for water that was being generated by necessary operations (pumps seal leakage, sampling system flushes, and floor flushing).

The ORNL personnel had considerable input into the following:

1. The existing stack monitor was in an unsuitable location following the accident. High backgrounds hampered on-line monitoring and sample changing. A new monitoring and sampling station was installed in a more-suitable location and new on-line instrumentation was installed for monitoring rare gases and radioiodine.

2. A new off-gas treatment system (HEPA filters and charcoal adsorbers) was obtained from a reactor construction site and was installed in series with the existing TMI Unit 2 systems. The exhausted adsorbers in the existing systems were replaced. The existing systems, which had been in service more than a year, had very low efficiencies for iodine adsorption.

3. A tank farm was designed that was installed in the TMI Unit 2 fuel handling pool. This tank farm was designed to provide an additional 110,000 gallons of storage capacity for water, thereby relieving the very severe shortage of tankage. The system was designed to permit the water to be transferred to as-yet-undetermined treatment systems.

4. Considerable effort was expended in the design and/or evaluation of a suitable waste evaporator.

5. Many attempts have been made to define the scope and character of the ultimate water treatment methods. ORNL personnel have had extensive discussions among themselves and have made extensive contacts with chemical processing and liquid waste experts throughout the nation to formulate a reasonable scenario for treating (a) the slightly contaminated water in most of the tanks, (b) some highly contaminated water in several of the tanks, (c) the water in the containment building and (d) the primary reactor coolant water.

6. ORNL is providing technical advice to a special group that was established within the Waste Management Group to coordinate the decontamination effort.

Conclusions and Results:

1. Methods for monitoring the release of radionuclides via the air handling systems have been improved.

2. Off-gas treatment systems have been upgraded significantly.

3. Water storage capacity will be increased when the tank farm installation is completed in the fuel handling basin.

4. A definitive action plan should be formalized for water treatment and waste disposal.

5. The decontamination of equipment and building areas may turn out to be the most difficult part of the task of regaining beneficial use of TMI-2 except for removal of the fuel and recertification of the reactor itself. The decontamination effort impacts on all facets of the recovery program. Decontaminating agents must be compatible with liquid waste treatment methods and with gaseous treatment systems.

(R. E. Brooksbank, E. D. Collins, F. A. Harrington, L. J. King, W. A. Shannon, J. W. Snider, O. O. Yarbro)

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Instrumentation and Diagnostics

Teams of ORNL instrumentation experts provided on-site technical assistance and special diagnostic instrumentation from April 3 to April 30, 1979. Direct current, strip chart, and spectral analytic measurements were made of 1) in-core thermocouples (approx. 50); 2) reactor coolant pressure; 3) in-core self-powered neutron detectors; 4) ex-vessel neutron detectors. The two principal concerns which depended heavily on these diagnostic measurements were boiling in the core and entrained or trapped gas in the primary system. Two to five people were at the site during this period to provide immediate assistance to operations as needed.

(R. C. Kryter, D. N. Fry, S. J. Ball, R. M. Carroll, T. E. Mott (TEC), J. C. Robinson (TEC), R. L. Shepard, W. H. Sides, C. M. Smith, G. L. Zigler (SAI))

DIRECT AND DEDICATED SUPPORT

ORNL has provided analytical support and technical assistance to the recovery efforts at Three Mile Island in several areas including:

1. Cooling of Disrupted Core
2. Analysis of Fuel and Cladding Effects
3. Analysis of Primary Coolant Water Sample
4. Radiation Effects and Core Nuclear Analysis
5. Radiation Shielding and Effects
6. Hydrogen Chemistry
7. Instrumentation and Diagnostics

The work involved as many as 50 staff members. The following is a summary of each effort.

1. Cooling of Disrupted Core

Four ORNL staff members made calculations to determine the feasibility of cooling core debris if it were located on the lower vessel head. If all the heat generated (at one-week decay heat levels) were transferred downward some boiling of water external to the vessel would have to occur to maintain cooling. If a free path existed around the vessel so that ste water mixtures were free to flow out around the upper level of insulation, leakage past the instrument thimbles would be adequate to allow ingress of sufficient cooling water. Also, the outer two inches of the steel vessel would be at temperatures below 800°F and therefore would have sufficient structural strength to hold the core debris. The flow distribution plate immediately below the core lower grid plate is not strong enough to hold the entire weight of the debris. Unfortunately, the pressure vessel is supported on a support ring which blocks access of water to a roughly toroidal volume space and would trap steam between it and the vessel. No information was available on whether gaps existed in the support ring-to-vessel welds that could serve as vents. There is a small area of the vessel (that extends below the level of the bottom of the support ring) that touches water. Heat transfer from this zone is sufficient to handle about one ton of fuel debris, compared to a total potential inventory of about 100 tons.

Calculations were also made to estimate limits of coolability of debris in the core, given the existing exit temperature profiles and estimated flow and pressure drop. The behavior of the core was consistent with flow through a debris bed of spheres of 0.020 in. diameter.

(Irregular shapes could be larger than this because of shape factor effects). Velocities in the upper plenum were insufficient to levitate spheres of diameter larger than 0.010 in. Calculations of behavior under natural circulation indicate that no boiling would occur at 1000 psia pressure. Partial boiling would occur at 500 psia, but would be statically stable (no effort was made to evaluate dynamic instability). However, dryout would occur at 50 psia. Cooling by radial conduction of a cylinder of debris with steam in the interstices would be ineffective for diameters exceeding about one foot, if the internal temperature is not to exceed 5000°F. The SABRE calculations for flow around an obstruction covering 30% of the core indicate low velocities downstream of the blocked zone and no recirculating flow; however, temperature effects were quickly homogenized at distances exceeding about six inches from the downstream "face" of the blockage. Therefore, if the thermocouple junctions were somewha above the core debris, slumped around the guide tubes which contain them, they would indicate temperatures somewhat more uniform than actually occurring.

(M. H. Fontana, J. F. Dearing, P. W. Garrison, S. Rose)

2. Analysis of Fuel and Cladding Effects

Three ORNL staff members participated in an EPRI-chaired utility-industry meeting on the condition of the Three Mile Island core. About 30 persons with expertise in areas relating to LWR cores were present at B&W Headquarters in Lynchburg, Va. The ORNL team was part of the working group on Zircaloy water reaction and hydrogen inventory. The discussions centered around the interpretations of the information and data from which the critical temperatures, the degree of clad oxidation, and the spatial distribution of damage in the core might be estimated. While much of this information was peripheral, (in the sense that no direct measurements of temperatures, water levels, etc., during the accident were at hand), clearly at least parts of the core had seen very high temperatues, and a substantial fraction of the Zircaloy comprising the core had been oxidized. In evidence, too, were temperature "maps" from thermocouples persumably at the top of the core that pointed to the existence of nonuniform damage areas. The details of the fission product release information, the calculations of the amount of hydrogen generated, and the characteristics of the self-powered neutron detectors were especially useful in this regard.

(D. O. Hobson, R. A. Lorenz, R. E. Pawel)

3. Analysis of Primary Coolant Water Sample

ORNL's Analytical Chemistry Division performed analyses on a water sample from the TMI primary coolant. Types of analyses were for radioelements, gamma spectrometry, mass spectrometry, Boron titrimetry, pH, and gross activity. The sample was 3 ml clear water in a glass vial and read 2.7 R/hr at contact. It was shipped within a small lead pig (2") within a 55 gal drum. No smearable alpha contamination was detected. Slight smerable beta contamination was observed on the vial exterior. Several analytical methods were used in some cases. Also, because the sample was alkaline and hence subject to absorption effects, tests were performed upon the empty vial itself and upon the sample after acidification.

(W. D. Shults, J. A. Carter, W. H. Christie, L. T. Corbin, J. F. Emery, G. I. Gault, L. R. Hall, L. M. Jenkins, W. R. Laing, L. Landau, E. G. Miller, W. R. Musick, K. J. Northcutt, H. A. Parker, B. Philpot, S. H. Prestwood, J. C. Price, H. C. Smith, J. R. Stokely, R. L. Walker)

4. Radiation Effects and Core Nuclear Analysis

The Nuclear Engineering Applications Department of Computer Sciences Division was requested by NRC to consider three problems associated with radiation effects.

1. Radiation exposure to instrumentation on the pressurizer tank.

   A. Dose rate at surface of tank given the interior water-bourne sources.

   B. Dose rate at instrument bank on lower containment wall given the air and water-bourne sources.

2. Determine Xe-135 source strengths in 1 and 2 inch Schedule 40 pipes for given dose rates at outer surfaces.

3. Inventory of all radioactive isotopes in the reactor at the time of the accident and at given time intervals after shutdown.

(G. E. Whitesides, R. L. Childs, O. W. Hermann, J. R. Knight, J. V. Pace, R. M. Westfall)

5. Radiation Shielding and Effects

Members of the Engineering Physics and Computer Sciences Divisions have provided assistance on several questions involving radiation shielding. This group is still actively involved in continuing analyses. Specific problems which have been investigated include 1) pressurizer transducer dose, 2) core isotopics and decay heat, 3) tank car storage and time-dependent tank car dose, 4) steam generator repair, 5) BF_3 detector readings, 6) ion-exchange resins, 7) pipe transport of primary coolant, and 8) tank storage in spent fuel handling pool.

(D. E. Bartine, T. J. Burns, R. L. Childs, W. W. Engle, D. T. Ingersoll, J. V. Pace, D. L. Selby)

6. Hydrogen Chemistry

The Nuclear Safety Information Center (NSIC) was requested by Floyd Culler of EPRI to provide information that would help in analysis of the hydrogen bubble that had formed in the reactor pressure vessel. NSIC staff estimated radiation induced recombination rates, studied the TMI high pressure system in the SAR, and collected data on hydrogen explosion limits. Working in the same area, Glen Jenks of the Chemical Technology Division provided valuable information about hydrogen chemistry and radiolytic gas generation and recombination.

(J. R. Buchanan, R. B. Gallaher, G. H. Jenks, G. T. Mays, R. L. Scott)

7. Instrumentation and Diagnostics

Members of ORNL's Instrumentation and Controls Division are continuing to provide on-call expert technical assistance to diagnose problems and lend advice to those working at the TMI site. Principal areas of concern are instrumentation and control, thermohydraulics, noise diagnostics, reactor systems, and neutronics. Calculations, investigations, and experiments with subsequent recommendations have been made in the following areas:

1. Assist on-site team performing noise diagnostics and other measurements.

2. Study to predict or explain the pressure pulse response observed on the primary system pressure sensors.

3. Calculations to examine adequacy of natural convection for core cooling.

4. Evaluation of available data to determine if boiling is occuring in the core.

5. Search for substitute methods for measuring pressurizer level, should all existing sensors fail.

6. Evaluate failure modes and predicted radiation life of pressurizer level sensors.

7. Evaluate failure modes of core thermocouples to help establish validity of core temperature measurements.

8. Confirm and evaluate sequence of events to predict probable extent of damage.

9. Evaluate feasibility of using a Resistance-Temperature Device (RTD) as a level probe in the pressurizer.

10. Examine failure modes of Rhodium self-powered neutron detectors to explain anomalous readings.

11. Evaluate dynamic hydraulic conditions at the pressurizer differential pressure cells to find explanation for unexpected behavior.

12. Calculate the potential for cooling the core by flooding the cutside of the reactor vessel if natural convection cannot be established.

13. Evaluate or propose ways in which noise analysis might be used to infer water level in the reactor vessel.

14. Propose methods for deriving a direct readout of subcooling for display to operators.

15. Explore what can be learned about thermal-hydraulics from in-core neutron detectors (eg., water level, boiling, pressure, etc.).

16. Evaluate implications and probable causes of high count rates on startup neutron detectors. Determine if related to reactivity in the core.

17. Participate in boiling tests at the Blowdown Heat Transfer experiment to acquire baseline data for determination of boiling at TMI.

18. Provide on-site assistance in rigging substitute pressurizer level measurements with RTD or pressurizer heaters.

(L. C. Oakes, J. L. Anderson, R. S. Booth, F. H. Clark, R. E. Hedrick (SAI), M. B. Herskovitz, J. T. Mihalczo, P. J. Otaduy, J. R. Penland (SAI), R. S. Stone)

FRM:JS
5-25-79

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Three Mile Island - 2
Technical Support

Boiling Detection

S. J. Ball

Problem:

Perform noise measurements on the core exit thermocouples to determine: (a) if boiling is occurring in the core; (b) if there were substantial changes in the noise signals since the initial measurements were made (Apr. 6-7, during forced circulation mode); and (c) if other useful information could be obtained from the measurements.

Description of Measurements:

1. During the period April 27-30, noise measurements were made on each of the 51 core outlet thermocouples (C/A grounded junction) using Princeton Applied Research (PAR) amplifiers (Gain ~30,000), and an HP 5420A noise analyzer. The signals were analyzed two-at-a-time and recorded simultaneously on a Brush recorder. Power spectral densities (PSD), cross PSD, and coherence were stored on tape for later plotting and analysis. Integrated power from 0.0122 to 1.53 Hz was recorded for each PSD signal.

2. Except for one case (see item 3 below), the signals appeared to be uncorrelated. The amplitudes of the fluctuations ranged from <0.1°F peak-to-peak (p-p) for most cases, to ~1°F p-p (5 cases), to ~6°F p-p (1 case). The signals were not all stationary, as occasionally on repeat analyses the magnitude and character of some PSD's would be quite different.

3. In one case, two thermocouple signals analyzed (9H and 12F, see core map, Figs. 1-2) had oscillatory (rather than random noise) characteristics, a ~50 sec period, were strongly correlated (coherence ~0.9 at 0.02 Hz),

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and were clearly related to similar oscillations in the loop A and B pressure signals recorded elsewhere. These oscillations were ~2-3 psi p-p, lightly damped, of a slightly different frequency from each other, and appeared to be provoked by the actions of the primary water makeup system. Brush recorder traces of 9H (the hottest of all the signals, ~315°F) showed what appeared to be a cancellation effect during periods when the two "opposing" loop pressure signals had a certain phase relationship. (The phase was not readily obtainable since the 3 signals were recorded on 3 separate recorders.)

4. Tests were made of the effects of various thermocouple leakage resistance paths on their readings. Abrupt "shifts" in several thermocouple readings had been observed, so the tests were made to see if the behavior could be attributed to intermittent leakage. A thermocouple that normally read 180°F read ~40°F with the low side shorted directly to ground, and ~140°F with the high side shorted to ground. Shorting the high side through a 10 KΩ resistor reduced the reading ~4°F, and through a 2 KΩ resistor ~16°F.

Observations and Tentative Conclusions Related to the ORNL Measurements:

1. The thermocouple noise measurements did not suggest, nor did they rule out, the possibility that boiling is taking place somewhere in the core. In many cases, thermocouples with the larger PSD's had relatively low temperature readings. The magnitudes of the noise signals were surprisingly low, especially considering the large thermal gradients present in the upper plenum. Saturation temperature for the operating pressure during our tests (~900 psi) is ~530°F. It would be possible for steam bubbles to recondense before reaching the thermocouples, and for boiling to occur in non-monitored areas. Probably the best way to infer

boiling would be to closely monitor a group of the most suspect (noisiest) thermocouples, then lower the system pressure and watch for significant increases in the noise power.

2. The case of the oscillatory and correlated thermocouple signals (9H and 12F) cannot be reasonably explained by a pressure-affecting-temperature boiling argument, since: (a) the start of the thermocouple oscillations coincided with the start of the pressure oscillations; any changes in signal due to a well-submerged boiling region would probably have taken several minutes to become established; (b) the signals were quite "clean" and uncharacteristic of a boiling noise signal; and (c) the 180° phase relation between the two signals would not be readily explainable by boiling. One theory which can account for this behavior is as follows: The upper plenum during the natural convection mode is relatively stagnant and has numerous "plumes" of widely-varying temperatures emanating from the core regions. A hydraulically-induced pressure transient sets up wave-like oscillations in the two loops, causing back-and-forth sloshing in the upper plenum, and thus lateral "waving" of the plumes in the vicinity of the 9H and 12F probes. This could account for the "immediate" response to the pressure change, the cleanliness of the signals and the 180°F phase. This model also requires more than one free surface in the system in order to set up the wave motion, so it infers that there may be noncondensible gas pockets in the reactor and/or steam generator upper plenums. Followup measurements and studies based on this model may also be useful in deriving more detailed information about the condition of the system. For example, the periods of oscillation of the two loops might be related to the size and location of voids. If so, one may infer

changes in void size due to a system pressure reduction. In any case, it would be useful to record the two pressure signals and thermocouple 9H noise on the same instrument.

3. The "waving plume" model may also be used to rationalize the behavior of several thermocouple signals in the three hours following the transition to the natural circulation mode. The textbook behavior of core outlet temperature would be an abrupt rise (corresponding to the reduction in flow) followed by a slow decline as the full natural circulation flow is established (Fig. 3, H8). However, several signals instead showed abrupt drops following the pump trip (fig. 4, H5). Considering the plume model and noting that the upper plenum flow distribution patterns would be quite different for single-pump forced circulation and natural circulation, it appears likely that the thermocouples such as H5 are simply monitoring a different combination of plumes.

4. The abrupt shifts observed in some thermocouple readings (notably 3F) can more readily be attributed to a postulated shifting of the leakage resistance of a damaged probe than to postulated scenes in which hot debris settles temporarily near the sensor.

Work Requested By:

NRC-TMI (Stello, Ackerman).

Results Reported To:

NRC-TMI, ad hoc Industry Advisory Group (IAG), General Public Utilities, and Babcock and Wilcox, April 30, 1979.

Worked Performed By:

S. J. Ball, R. C. Kryter, W. H. Sides, Jr., and G. L. Zigler (SAI).

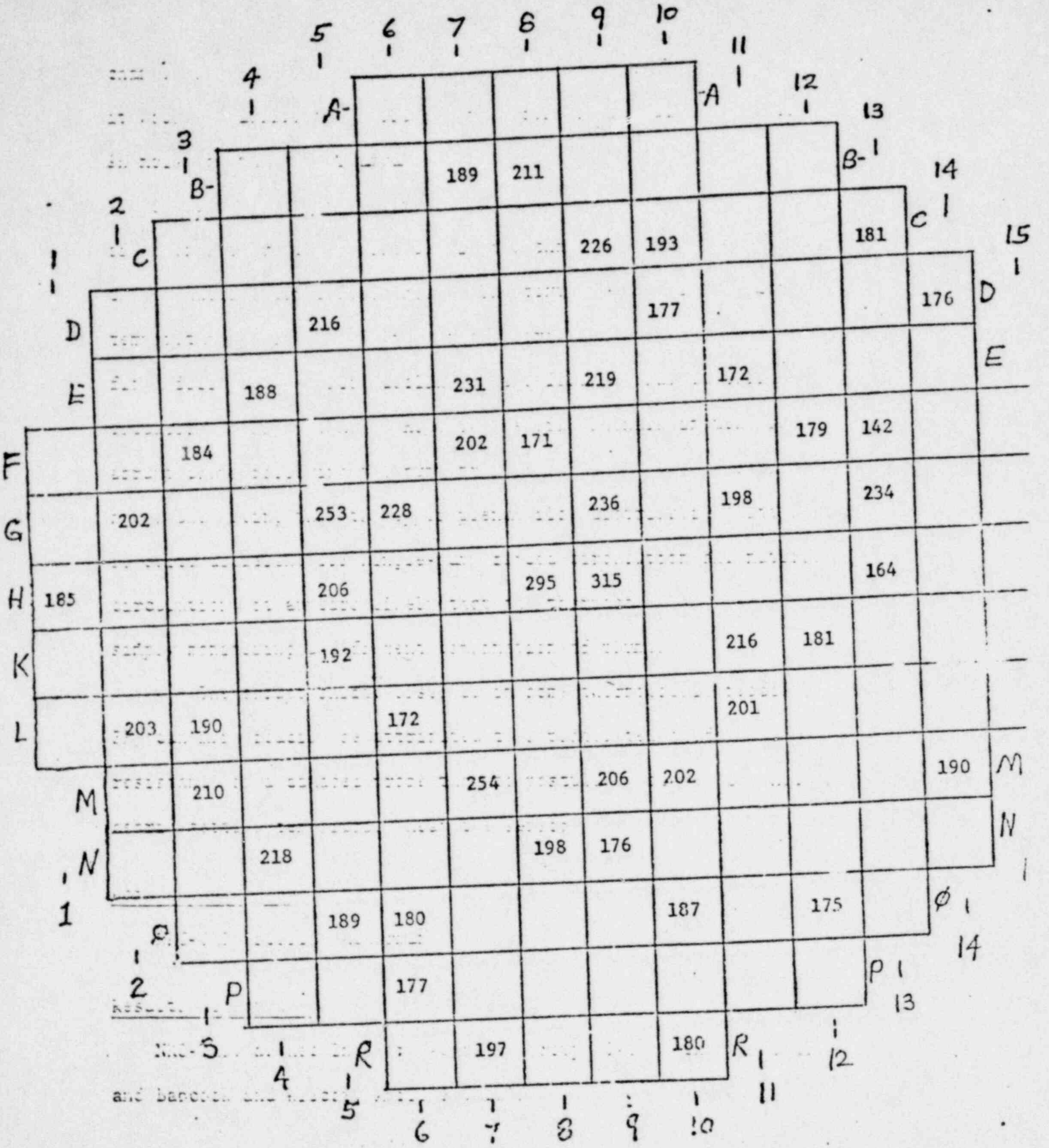

Fig. 1. TMI-2 Core Exit Thermocouple Steady-state Readings (°F), 3/28/79, 2400 hrs. Natural Circulation Cooling

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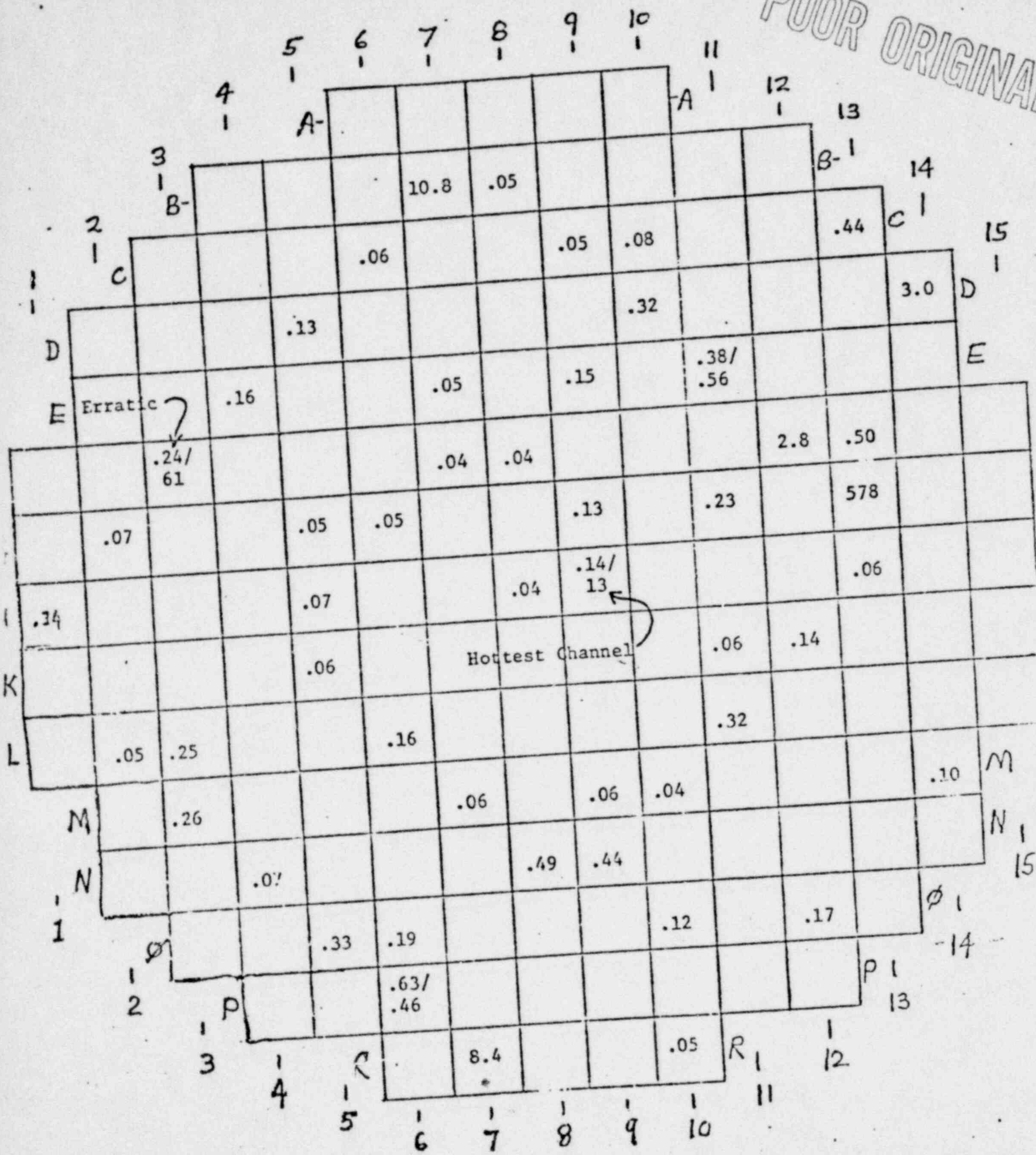

Fig. 2. TMI-2 Core Exit Thermocouple Mean Square Noise (mV^2). Bandwidth 0.0122-1.53 Hz. Data Period: 4/28/79, 1630 hrs. to 4/29/79, 1230 hrs. Natural Circulation Cooling.

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Estimated Cost:

20 Mandays plus transportation; $9,600.

Three Mile Island - 2
Technical Support

Instrumentation

R. L. Shepard

Problems:

1) Six of 52 in-core thermocouples are reading significantly higher than the others. Can these readings be validated and/or can failure mechanisms be postulated to explain the readings?

2) Can the operable Resistance-Temperature Devices (RTD) in the primary loop be calibrated using Johnson noise measurements or other techniques?

3) Can a method be developed for using the RTD in the pressurizer for indicating pressurizer water level?

4) Diagnose the condition of the in-core thermocouples using Time Domain Reflectrometry or otner techniques.

Description of Work Performed:

1) Possible failure modes were considered by several thermocouple experts. The most likely failure mechanisms-melting of the sheath or ingress of moisture into the insulation-could not cause readings higher than the true temperature. The location of the thermocouple junction may have been displaced by melting along the sheath.

2) Methods are not sufficiently developed for using Johnson noise for field calibration of RTD's.

3) Methods have been developed for measuring in-situ time response of RTD's using loop current step response (LCSR) and for using electrical selfheating for detecting the presence or absence of water in contact with

the sheath of an RTD. The methods are of questionable applicability to TMI because of the massive (1.4 inch O.D.) stainless steel well which houses the RTD. Calculations were made to predict the sensitivity. Rosemont Engineering Company cooperated by determining the maximum safe heating current that could be used. Idaho National Engineering Lab (INEL) made a simulation using a tippable autoclave and a well similar to the one at TMI. Temperature differences of only 5°F were obtained between a wetted and unwetted well.

On April 28, Shepard and Carroll went to TMI, then to Davis-Besse to try the selfheating method in place. The method was found to be unsuitable due to long time constants (greater than 3000 seconds) and low sensitivity. Temperature changes due to water spray and heaters in the pressurizer would mask the level change.

A bridged heater method of determining pressurizer level, using 4 of the 40 or more heater elements was devised by B&W with assistance by ORNL.

4) Experiments were performed at ORNL to determine if Time Domain Reflectrometry could be used to verify the integrity of the in-core thermocouples. It was determined that the technique was unsatisfactory because of the large electrical resistance and long extension wires used on the thermocouples.

Conclusions and Results:

It was concluded that:

1) The in-core thermocouples are probably accurate in spite of experiencing temperatures above 200°F.

2) Time Domain Reflectrometry is not a useful diagnostic tool for long, high-resistance thermocouples.

3) The RTD might have been a useful indicator of pressurizer water level if the well had been less massive.

4) There is a general need in PWR's for diverse measurement techniques of process parameters, such as pressurizer water level.

Work Requested By:

1) NRC-TMI; MetEd-TMI; Mar 31.

2) NRC-Bethesda (Buhl); April 2.

3) NRC-IAG-TMI (Ackerman) April 3.

4) NRC-IAG-TMI (Ackerman) April 13.

Results Reported To:

Same as above plus:

Tom Murley, NRC April 6

Billy Jo Sheperd (B&W), TMI April 6

A review of activities was presented to David Cain and Alex Long at EPRI on May 10, 1979.

Work Performed By:

D. Agouridis, ORNL
R. M. Carroll, ORNL
J. L. Horton, ORNL
T. W. Kerlin, UT
Don Matychuk, Rosemont
T. E. Mott, TEC
R. L. Shepard, ORNL
Marlin Stanley, INEL
Wylie Stansell, Tektronix

Three Mile Island - 2
Technical Support

Diagnostic Instrumentation

D. N. Fry, et. al.

Problem:

Provide TMI on-site expert technical assistance with special diagnostic instrumentation as part of a special instrumentation group (SIG). The group charter is:

I. Collect the maximum amount of diagnostic data consistent with not interfering with operations in either the Control Room or cable spreading areas. No safety circuits will be accessed and all hookups to plant systems must have the agreement of the instrumentation technican on duty in the cable spreading room. All efforts will be coordinated with the control room.

II. If and when plant instrumentation signals are lost, the special instrumentation group is to aid operations by determining whether laboratory or special instrumentation might be used to obtain the desired data.

Description of Work Performed:

The group made frequent measurements and diagnostic evaluations over the period from April 3 to April 30, 1979. Direct current, strip chart, and spectral analytic measurements were made of 1) In-core thermocouples (approx. 50); 2) Reactor Coolant Pressure: 3) In-core self-powered neutron detectors; 4) Ex-vessel neutron detectors. The two principal concerns which depended heavily on these diagnostic measurements were boiling in the core and entrained or trapped gas in the primary system. Two to five people were at the site during this period to provide immediate assistance to operations as needed.

Conclusions and Results:

The measurements and recommendations of this group were very helpful in degassing the primary system and in determining that satisfactory natural convection cooling had been established after the coolant pumps were deactivated. Separate detailed reports have been prepared describing the measurements made and their significance.

Work Requested By:

NRC-TMI (Stello, Ackerman).

Results Reported To:

NRC and GPU as timely.

Work Performed by:

D. N. Fry, R. C. Kryter, G. L. Zigler (SAI), C. M. Smith, W. H. Sides, S. J. Ball, J. C. Robinson (TEC).

Estimated Costs:

Approximately 60 Mandays plus a variety of transportation to and from the site with people and equipment; $35,000.

INTRA-LABORATORY CORRESPONDENCE

MAY 14 1979

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

To: F. R. Mynatt

From: R. E. Brooksbank

Subject: ORNL Assistance in Response to the Three Mile Island Accident

Statement of Problem: Provide technical support to the Waste Management Group of the TMI Recovery Team. The primary concern was to "make certain that nothing reaches the environment that would be detrimental to the health and general welfare of the general public."

Work Performed: The technical Group was fully integrated with the Waste Management Group (WMG). Members functioned as peers in WMG meetings and interacted continually with supervisors of the WMG groups concerned with liquid effluents, gaseous effluents, and construction. Technical Group members functioned in areas of their own expertise and provided liaison with experts at ORNL and elsewhere. Thus, the realities of operational capabilities and limitations were continually integrated into theoretical judgements.

The Technical Group was active in evaluating data concerning the release of radionuclides in gaseous and liquid effluents including evaluation of sampling and analytical methods.

The major problems addressed by the WMG immediately following the accident were: (1) stopping

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the release of ^{131}I to the environment and (2) providing adequate storage capacity for water that was being generated by necessary operations (pumps seal leakage, sampling system flushes, and floor flushing). Plans for overcoming these problems were formulated which addressed short-term solutions with as much consideration as possible for the long term.

The ORNL personnel had considerable input into the following:

(1) The existing stack monitor was in an unsuitable location following the accident. High backgrounds hampered on-line monitoring and sample changing. A new monitoring and sampling station was installed in a more-suitable location and new on-line instrumentation was installed for monitoring rare gases and radioiodine.

(2) A new off-gas treatment system (HEPA filters and charcoal adsorbers) was obtained from a reactor construction site and was installed in series with the existing TMI Unit 2 systems. The exhausted adsorbers in the existing systems were replaced. The existing systems, which had been in service more than a year, had very low efficiencies for iodine adsorption.

(3) A tank farm was designed that was installed in the TMI Unit 2 fuel handling pool. This tank farm was designed to provide an additional 110,000 gallons of storage capacity for water, thereby

relieving the very severe shortage of tankage. The system was designed to permit the water to be transferred to as-yet-undetermined treatment systems.

(4) Considerable effort was expended in the design and/or evaluation of a suitable waste evaporator. This effort proceeded rapidly initially, but later became bogged down in deliberation of alternatives when treatment and discharge of some of the water and improved control of water sources relieved the urgency and permitted deeper deliberation.

(5) Considerable effort has been expended in attempts to define the scope and character of the ultimate water treatment methods. It is in this area that the desire for detailed data of sufficient magnitude and quality to permit the formulation of processing flowsheets has conflicted most severely with the realities of plant operations. ORNL personnel have had extensive discussions among themselves and have made extensive contacts with chemical processing and liquid waste experts throughout the nation to formulate a reasonable scenario for treating (a) the slightly contaminated water in most of the tanks, (b) some highly contaminated water in several of the tanks, (c) the water in the containment building and (d) the primary reactor coolant water. ORNL personnel continue to feel that a comprehensive plan based on analyses of the water to be treated is

required for safe and efficient treatment of the liquid wastes. Evaporation, ion exchange, and waste solidification will be involved. The details of the nature and sequences of processing steps remain to be decided.

(6) ORNL suggested that a special group be established within the Waste Management Group to coordinate the decontamination effort. Such a group has been established and ORNL has been deeply involved in providing technical advice to the group.

Conclusions and Results:

1. Methods for monitoring the release of radionuclides via the air handling systems have been improved.
2. Off-gas treatment systems have been upgraded significantly.
3. Water storage capacity will be increased when the tank farm installation is completed in the fuel handlin basin.
4. A definitive action plan should be formalized for water treatment and waste disposal.
5. The decontamination of equipment and building areas may turn out to be the most difficult part of the task of regaining beneficial use of TMI-2 except for remova of the fuel and recertification of the reactor itself. The decontamination effort impacts on all facets of the recovery program. Decontaminating agents must be compatible with liquid waste treatment methods and

with gaseous treatment systems. An example of the problem is the selection of cleaning agents. Foaming agents might cause difficulties in liquid waste treatment and solvents might cause a significant decrease in the absorption efficiency of charcoal adsorbers for removing iodine for building ventilation air. Care must be exercise to prevent recontamination of cleaned areas by personnel movement and air flow.

This is clearly the area in which the utility personnel have the least experience and must rely upon outside technical help.

Work Requested By: The original contact with the ORNL Chemical Technology Division was made by F. L. Culler or EPRI.

Results Reported to: Herman Diekamp (GPU), John Collins (NRC), Bob Arnold (GPU), Ron Williams (GPU), Ben Rusche (Consultant: letters dated April 20, 1979), countless daily contact:

Work Performed by: R. E. Brooksbank, O. O. Yarbro, F. A. Harrington, J. W. Snider, E. D. Collins, W. A. Shannon, L. J. King

Estimated Costs:

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BUSINESS CONFIDENTIAL

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 18, 1979

To: D. E. Ferguson

Subject: Status of Effluent Treatment at TMI

From May 15-18, the writer visited the TMI site to review the status of effluent treatment and to inspect the facilities, procedures and the construction projects recommended by ORNL personnel on earlier assignments. On May 18, D. O. Campbell was requested to discuss with Metropolitan Edison the problems associated with fission product removal by ion-exchange methods.

1. Status of Gaseous Effluent Treatment

Currently, the level of iodine releases from TMI are below the technical specification limit (0.025 Ci/day) as measured in HPR-219. Of the 4-trains of charcoal traps, 3 have been completely changed (240 trays) and the fuel handling building Train B traps are currently awaiting a shipment of charcoal to permit the changeout. Charcoal trap efficiencies have not yet been measured by DOP testing nor have iodine DF's been measured. These tests are scheduled within the next week. Following the tests an air-flow balancing exercise will take place of the Auxiliary and Fuel Handling buildings. A hinged blank has been placed in the suction plenum of the stack and the stack has been isolated. All gaseous effluent is now directed through the new 100,000 cfm emergency off-gas systems and discharge is at roof level.

2. Status of Liquid Effluent Treatment

Three categories of water resulting from the TMI incident are of concern. These include (1) water contained in Auxiliary-1 Building containing some incident water, (2) water contained in the Auxiliary-2 Building containing incident related water, and (3) water in the existing primary loop circuit and in the reactor containment building generated from the incident. The status of the treatment of each of these classes of water are more fully described below.

2.1 Auxiliary-1 Building Water Treatment

A total of 83,500 gallons have been released to the river (below technical release limits, 10 CFR-20) after treatment by evaporation followed by disposable mixed bed ion-exchange processing (EPICOR-1 or Cap-Gun Process). The processing scheme consists of evaporating the water with the condensate being collected, analyzed and discharged to the river; the resultant evaporator bottoms are then transferred through a series of 185 ft^3 mixed-bed ion exchange columns. These disposable

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beds are loaded in a carrier for ultimate disposal in an off-site burial ground. Data collected on all of the beds is given in Table 1.

Table 1. Ion Exchange Cannister Data

Resin Container No.	Treated Gallons	Total Curies	I Curies	Bed Radiation Level (R/hr)
D-1	20,000	32.6	5	3.5
D-2	10,000	18.4	7.7	7.0
D-3	24,700	63.1	5.7	5.0
D-4	33,400	45.7	26.5	2.5
D-5	15,400	39.2	17.9	3.8

At the present time there is a "hold" on shipment of solid materials leaving TMI because of political implications; therefore none of the loaded ion-exchange cannisters have been transported to the commercial burial ground sites (Barnwell, SC; Richland, Wash). Attempts are currently in progress to allow shipments to move from the site and the appropriate individuals have been contacted.

2.2 Auxiliary-2 Building Water Treatment

Currently, a total of 150-190,000 gallons of water is contained in this building which has been generated as the result of the incident. The leakage rate into this building continues at the 0.32 gpm rate with activity levels in the range of 1-100 µCi/cc (400 R/hr level at tank surfaces in some cases). The processing scheme proposed for this system employs a modification of the EPICOR or Cap-Gun process. The basic system will be composed of three absorber beds hooked in series. The first bed (24 ft^3), will be loaded with silver-impregnated charcoal for I_2 removal, the second bed will be cation resin (nuclear grade-strong acid) for Cs removal followed by a third mixed bed of resin for polishing. Finally, a particulate filter will be utilized to catch resin should the system malfunction. The equipment will be semi-remotely operated in the equipment decontamination building conceptually designed by ORNL.* This building has been sealed, contained, equipped with an 8,000 cfm off-gas system, a remote control room and the necessary crane network to permit remote handling of loaded resin beds. The objective will be to sorb sufficient fission products to a level of 2500 R/hr on the surface of the vessel or (30 R/hr) at the surface of the integral biological shield. Operational checkout of this system is expected to take place within the next month and construction of the facility has essentially been done. A sufficient quantity of approved licensed containers has been located to permit the orderly shipment of these units from the site.

*J. W. Snider.

2.3 Primary and Containment Water Treatment

A total of approximately 490,000 gal of water is now contained in the reactor building floor (up from 225,000 gal on April 1). The primary loop contains 85,000 gal which also has the same radioactive characteristics. Currently, a proposed contract is under consideration by GPU offered by Chem. Nuclear to treat this water in a series of 3-submergible ion exchange beds employing commercial technology. Beds will consist of a stage of filtration followed by Dowex ion-exchange systems. Beds (10 ft^3) will be loaded to resin degradation limits (10^8 Rads, 30 liter off-gas release/ft^3 resin degraded). Serious concerns were expressed about the number of shipments that would be required (+300/100,000 gal treated), and the performance of the resin in the radiation and heat load environment and the problems inherent in bed changeout under water. Dave Campbell reviewed the zeolite sorption of ^{137}Cs and the very major advantages for the use of this material for application in treatment systems. Because of the continuing inleakage of water into the reactor containment system there is a sense of urgency to obtain data as soon as possible. Currently, the liquid level in the reactor building is 6 1/2 ft; at 8 ft the reactor primary loop components (currently operating in the natural convection mode) becomes covered and natural convection systems are jeopardized. Because of this situation. ORNL will be formally requested to perform a series of scouting experiments to establish resin and zeolite performance. Metropolitan Edison is currently drafting a letter to D. B. Trauger urging our assistance in this critical technical area. A total of 130 ml of primary loop water is expected to be shipped to ORNL to initiate these studies pending approval by DOE and ORNL's management. An attached letter was generated outlining a proposed process development scheme for water treatment.

In addition to Chem Nuclear involvement, AGNS is expected to submit a proposal to supervise the operation of this system for Metropolitan Edison. Because of the experience base of AGNS in the handling of radioactive materials in reprocessing plants, this involvement lends considerable creditability to the safety of operations at TMI.

Currently, the construction of the tank farm system* has not been completed and two additional weeks are envisioned before this system can receive water. This facility was constructed to hold 110,000 gal of high activity water in a safely-contained manner prior to subsequent processing.

3. Miscellaneous Information

Metropolitan Edison is desirous to obtain DOE's help in the formation of senior technical advisory group for the review of proposed recovery schemes from a technical and operational standpoint. Currently, C. Ice, former Director of the Savannah River Laboratory, is employed as a consultant to TMI and will coordinate the efforts of this 4-man team. Other

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*Designed by ORNL.

members being requested include myself and two individuals from SRL including one engineer and a chemist. It is hoped that in this manner, the expertise of both ORNL and SRL can be utilized as required as the recovery program proceeds.

Because of the changing nature of the operations at TMI (from the reactor control phase to the waste management phase), a management review team is currently attempting to more firmly establish the organization of the entire recovery operation. This effort should considerably minimize the on-site confusion that has been experienced to date.

On May 16, a public hearing was held in Lancaster, Pa., to discuss the treatment of effluents at TMI. Following a presentation by NRC, GPU and Metropolitan Edison, lengthy question and answer sessions were conducted. Approximately 50 individuals attended (200 were invited). Participants jointly felt that much had been said to alleviate the concerns of the public about the release of contaminated water to the environment. Prior to the hearing, an injunction was drawn by a local community to hold the water at TMI without further treatment. On May 17, the Mayor of Lancaster said the city would file an injunction on May 18.

On May 17, the Presidential Commission on TMI toured TMI-Unit 1, spending a great deal of time inside the Reactor Containment Building. The hearings by this Commission which were scheduled for this week were not held because Congress did not give the Commission subpoena powers.

R. E. Brooksbank

REB:bjh

cc: D. O. Campbell
L. T. Corbin
L. J. King
J. A. Lenhard (ORO)
A. P. Malinauskas
F. R. Mynatt
W. D. Shults
J. W. Snider
D. B. Trauger
R. G. Wymer

Date MAY 17, 1979

GPU Service

Subject PROPOSED ASSISTANCE TO TMI

To B. RUSCHE   Location TMI

Certain experimental measurements are important to the selection of a suitable process for cleaning up the contaminated water at TMI with minimal risk to safety and the public. The goals of the clean-up process are removal from the water of (1) cesium, (2) strontium, (3) iodine, and (4) possibly other radioisotopes so that the product water meets the specifications (of 10 CFR-part 20) for release to the environment.

A sample of water as representative as possible of the larger quantity of highly contaminated water will be required. Because of the high radiation level the experiments will have to be carried out in a hot cell facility. The following experiments are suggested:

1. Measure distribution coefficients (K_d) between the water and selected absorbents (such as organic ion exchange resins, zeolites, and charcoal) for cesium, strontium, iodine, and other isotopes as appropriate. The purpose is to verify the behavior of the particular chemical system as it exists on site.

2. Make similar measurements after adjusting the water composition to represent better the range of compositions expected, or to cause favorable changes in chemical properties, if necessary.

3. Carry out small scale column tests with promising absorbent systems to verify multi-stage performance.

4. If practical, extend column tests to two or more columns in series to determine maximum DF attainable. This could be a miniature version of an actual clean-up system.

The results of these experiments will permit the selection of processes particularly suited to the goals of the clean-up operation, and will provide more confidence in their ultimate success.

R. E. Brooksbank

R. E. BROOKSBANK

D O Campbell

D. O. CAMPBELL

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cc: J. Collins
C. Ice
R. Williams

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

To: F. R. Mynatt

From: M. H. Fontana MH Fontana

Subject: Assistance in Response to Three-Mile Island Accident

This is in response to your memo of May 2, 1979, requesting a brief summary of work that we did in response to the Three-Mile Island Accident. This entailed: 1) cooling of debris in the lower head and 2) coolability of core debris in place.

I. COOLING OF DEBRIS IN LOWER HEAD

I1. Statement of Problem

Compute feasibility of cooling core debris, if it should fall onto bottom head, by external flooding of vessel.

I2. Description of Work Performed

Calculations were made of the volume of the lower head that would be filled by core debris, the heat transfer through the head, and the free convection cooling by water external to the bottom head, between the vessel and the stand-off, multi-layered stainless steel insulation.

I3. Results

If all the heat generated (at one-week decay heat levels) were transferred downward some boiling of external water would have to occur to maintain cooling. (The upward/downward heat flux split would really be about 3/1.) If a free path existed around the vessel so that steam water mixtures were free to flow out around the upper level of insulation, leakage past the instrument thimbles would be adequate to allow ingress of sufficient cooling water. Also, the outer two inches of the steel vessel would be at temperatures below 800°F and therefore would have sufficient structural strength to hold the core debris. The flow distribution plate immediately below the core lower grid plate is not strong enough to hold the entire weight of the debris.

Unfortunately, the pressure vessel is supported on a support ring which blocks access of water to a roughly toroidal volume space and would trap steam between it and the vessel. No information was available on whether gaps existed in the support ring-to-vessel welds that could serve as vents. Additional calculations were made of steam refluxing (convection on the vessel and condensation on the inner surface of the support ring) and of direct radiation heat transfer from the vessel to the water directly below; neither of these phenomena can transfer sufficient heat.

There is a small area of the vessel (that extends below the level of the bottom of the support ring) that touches water. Heat transfer from this zone is sufficient to handle about one ton of fuel debris, compared to a total potential inventory of about 100 tons.

I4. Work Requested

This work was requested by Joe Murphy (NRC) at Saul Levine's request, on Friday April 6, 1979.

I5. Results Reporting

Results were reported to Mark Cunningham (NRC), Gary Holahan (NRC), and Rich Denning (BMI-Columbus) by telephone and telefax, of April 10, and by memo of April 11, 1979.

I6. Estimated Costs

Three-man days, about $750.00.

II. COOLABILITY OF CORE DEBRIS-IN PLACE

II1. Statement of Problem

Estimate limits of coolability of debris in the core, given the existing exit temperature profiles and estimated flow and pressure drop.

II2. Description of Work Performed

Calculations were made of the flow-pressure drop characteristics of a region of the core characterized as a column of debris. Estimates were made of 1) coolability by natural circulation as a function of pressure, using hand calculations and also by using a computer program originally written to compute natural circulation boiling in the Sodium Boiling Test Facility; 2) limit to cooling by radial conduction, using hand calculations; and 3) flow around and behind an obstructed region of the core, using SABRE, a UKAEA code made operable at ORNL by the THORS program.

II3. Results

The behavior of the core was consistent with flow through a debris bed of spheres of 0.020 in. diameter. (Irregular shapes could be larger than this because of shape factor effects). Velocities in the upper plenum were insufficient to levitate spheres of diameter larger than 0.010 in. Calculations of behavior under natural circulation indicate that no boiling would occur at 1000 psia pressure. Partial boiling would occur at 500 psia, but would be statically stable (no effort was made to evaluate dynamic instability). However, dryout would occur at 50 psia.

Cooling by radial conduction of a cylinder of debris with steam in the interstices would be ineffective for diameters exceeding about one foot, if the internal temperature is not to exceed 5000°F.

The SABRE calculations for flow around an obstruction covering 30% of the core indicate low velocities downstream of the blocked zone and no recirculating flow; however, temperature effects were quickly homogenized at distances exceeding about six inches from the downstream "face" of the blockage. Therefore, if the thermocouple junctions were somewhat above the core debris, slumped around the guide tubes which contain them, they would indicate temperatures somewhat more uniform than actually occurring.

II4. Work Requested

This work was requested by Joe Murphy (NRC), as in Item I4, and by Rich Denning, BMI-Columbus, who was conducting this effort at NRC's request, about April 6, 1979.

II5. Results Reporting

Results were reported to Gary Holahan (NRC) and Rich Denning (BMI-Columbus) by telephone and telefax of April 10 and memo of April 11, 1979.

II6. Estimated Costs

Two man-weeks plus computer cost equivalent to one man-week (~$3,000).

MHF:ls

cc: S. Ball
J. Cleveland
J. Dearing
P. L. Garrison
W. O. Harms
K. Kibbe
T. S. Kress (Planning Committee)
L. C. Oakes
P. Patriarca
S. Rose
H. E. Trammell

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MAY 15 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

TO: F. R. Mynatt

FROM: M. H. Fontana, MHF

SUBJECT: Addendum to May 14 memo on Assistance for Three-Mile Island

In my memo responding to your request of May 2, I inadvertently left out Item 6 "Work Performed By".

I did the work on Part I. Part II was done by P. W. Garrison, Simon Rose, J. F. Dearing, and me.

MHF:ldj

cc: File

1915 212

MAY 1 [illegible]

NUCLEAR DIVISION

INTERNAL CORRESPONDENCE

May 14, 1979.

To: F. R. Mynatt

From: R. E. Pawel

Subject: Summary Report of Work Performed Relating to Three Mile Island Accident

My involvement in TMI-2 problems has been limited to two areas:

1. I was an appointed member (by J. R. Weir) along with D. O. Hobson and R. A. Lorenz of the TMI-2 Utility-Industry Committee on Reactor Core Condition. This committee met at Babcock & Wilcox Headquarters, Lynchburg, Va. on April 5, 1979 to discuss the then-available information that might be used to infer the extent and distribution of core damage. In particular, after a general session, I was a member of the Working Group on Zircaloy-Water Reaction and Hydrogen Inventory.

The discussions at the general session and work session centered around the interpretations of the information and data from which the critical temperatures, the degree of clad oxidation, and the spatial distribution of damage in the core might be estimated. While much of this information was peripheral, (in the sense that no direct measurements of temperatures, water levels, etc., during the accident were at hand), clearly at least parts of the core had seen very high temperatures, and a substantial fraction of the Zircaloy comprising the core had been oxidized. In evidence, too, were temperature "maps" from thermocouples persumably at the top of the core that pointed to the existence of nonuniform damage areas. It is my opinion that the committee, and the B&W/EPRI people prior to the meeting, made logical inferences from data on hand. The details of the fission product release information, the calculations of the amount of hydrogen generated, and the characteristics of the self-powered neutron detectors were especially useful in this regard.

It is my understanding that this committee was organized by EPRI at the request of G.P.U. Ed Zebrowski of EPRI is chairman; our B&W contact is Jim Tulenko. Other than with W. V. Johnson/M. L. Picklesimer, NRC, I have had no further direct contact with committee members outside ORNL. I am unaware of the present status of the committee.

2. I have been involved with M. L. Picklesimer in making estimates of the heat generation by the oxidation process and its relative contribution to the temperature excursions underwent by the cladding in the TMI-2 core. W. V. Johnson suggested at the Lynchburg meeting that this might be a useful calculation to make, and I assume that Pic is leading the way on this.

I have constructed a computer program that uses recent kinetic data for the steam oxidation of Zircaloy-4 to estimate the instantaneous rate of oxidation and heat generation for a given time-temperature excursion. This program was used to obtain the reaction energy contributions for a series of linear temperature ramps suggested by Picklesimer. The results were submitted to him by telephone and by mail in an informal manner.

While not specifically requested to do so, I have constructed an additional computer program that calculates the time-temperature excursions experienced by a section of cladding subject to a given decay heat, a computed oxidation reaction energy, and a simplified empirical heat loss scheme. I have used this program to examine the effect of several of the system variables. Because the model utilized in the code is idealized in many respects, the quantitative predictions of the time-temperature ramps are recognized as approximate. Nevertheless, the characteristics of these ramps, and their sensitivity to variations in the extent of the heat losses, are interesting and might conceivably be useful in conjunction with other information in suggesting a possible configuration of the TMI-2 core. A letter describing this program and some of the preliminary results and interpretations has been written and will be sent in a few days to M. L. Picklesimer for his comments and possible distribution.

I have received considerable support from several colleagues in the above efforts. Discussions with J. V. Cathcart, D. O. Hobson, D. L. McElroy, and R. K. Williams have been particularly helpful and are much appreciated. Aside from my own time, the costs of this work is limited to computer expenses, perhaps $200.00.

R. E. Pawel
Metals and Ceramics Division

REP:spb

cc: J. V. Cathcart
D. O. Hobson
C. J. McHargue
J. O. Stiegler
J. R. Weir

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LTR NO: 0514-07-79

MAY 15 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

To: F. R. Mynatt

From: D. O. Hobson D.O.H.

Subject: Contribution to ORNL Assistance in Response to the Three Mile Island Accident

Statement of Problem: To infer the probable condition of the Three Mile Island core from information available as of April 5, 1979.

Description of Work Performed: I was one of a group of three ORNL personnel (R. E. Pawel and R. A. Lorenz) asked to go to Babcock and Wilcox, Lynchburg, Virginia, to participate in an EPRI-chaired utility-industry committee meeting on the condition of the Three Mile Island core. Approximately 25 to 30 persons with expertise in areas relating to LWR cores were present. I was included because of my knowledge of zircaloy fuel cladding behavior under accident conditions, particularly oxidation, embrittlement, and cladding rupture characteristics.

The meeting lasted one day, the first part of which consisted of presentations by Babcock and Wilcox and other personnel of what was known about the accident. For the last part of the meeting, we broke up into three working groups, including one on the effects of zircaloy oxidation which I attended. Finally, the whole committee reconvened to review the day's findings and to make recommendations for future investigations of the core condition. A full description of the meeting was given in my trip report – D. O. Hobson to G. M. Slaughter, April 30, 1979, of which you have a copy. EPRI has my name as a contact for any later meeting of this nature.

Conclusions or Results: See trip report referenced above

Work Requested by: NRC and Warren Chernock (CE) 4/3/79

Results Reported to: EPRI (Ed Zebroski) 4/5/79
plus trip report 4/30/79

Work Performed by: See enclosed list of participants

Estimated Cost: Approximately $500 to NRC for my participation

DOH:lc

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xc: G. M. Slaughter
J. R. Weir, Jr.

MAY 3 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY Letter No.: 0430-06-79

April 30, 1979

To: G. M. Slaughter

From: D. O. Hobson D.O.H.

Subject: Trip to Babcock and Wilcox, Lynchburg, Virginia, April 5, 1979

On April 4, Dick Lorenz, Dick Pawel, and I flew by chartered plane to Babcock and Wilcox (B&W) at the request of NRC to attend a utility-industry committee meeting on the condition of the Three-Mile Island core. This meeting was chaired by Ed Zebroski of EPRI and was attended by those listed in Enclosure One.

Zebroski opened the meeting on April 5th by stating that the committee would review the thermal history of the incident which would be based on about 60–70% solid evidence and on conjecture for the remainder. He outlined the day's activities on fuel damage assessment in five parts: (1) Introduction, (2) Event sequence, (3) Summary of measurements and observations, (4) Core condition during transient and, (5) Engineering evaluation of core condition.

Chuck Baroch (B&W) presented the sequence of events which is shown in the second enclosure and which was emphasized as being very preliminary. Enclosure Two contains various notes by me on additional information presented during the entire meeting. Enclosure Three is a schematic of the reactor system.

George Meyer presented the thermocouple (TC) information that was acquired during the incident. One of the main problems in understanding the core behavior during the incident was caused by the TC recorders - none recorded above 700°F! Therefore, once the TC's started to heat up - from the approximately 650°F operating temperature of the fuel cladding - there were no longer any temperature records. Much of the analysis of the core behavior was inferred from the thermal characteristics of the self-powered neutron detectors in the core and from fission product analyses. Meyer estimated that at approximately two hours into the incident the core was uncovered down to about the one foot level.

Zebrowski stated that approximately 25 to 30% of the core saw temperatures as high as 1600°C, but that the remainder saw temperatures in the 1100 to 1200°C range. More than just the pellet-to-cladding gap inventory of I_2 was released.

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Dick Demar (B&W) gave a summary of observations and measurements made with radiochemical analyses, hydrogen analyses, noise analyses (for loose parts in the primary system), TC data and in-core neutron detectors. Failure estimates based on fission product data cannot discriminate between a small core volume that had lost almost 100% of its inventory, and a large core volume that released only about 5% of its inventory. I personally believe the latter to be the case.

It was discussed that the 52,000 lbs of Zircaloy in the reactor could release 408,000 ft^3 of H_2 gas at STP. It was estimated that a maximum range for the amount of Zircaloy actually reacted with steam was from 29 to 34 wt. percent.

Demar also stated that sensor arrays positioned around the system detected noises in the upper part of the "A" steam generator equivalent to a few ounces of material impacting the structure.

Ralph Frederickson (Bettis) reported on their analyses of coolant samples. Among his findings was the release of 22 to 28% of the total xenon inventory of the core. He said you could argue for 3000 to >4000°F fuel temperature, but little fuel melting. At this point Bill Johnston (NRC) raised the question of the solubility of Ba, Sn, Mo, and Ce oxides in the coolant water. Lack of those elements in the water does not necessarily preclude fuel melting. Frederickson stated that finding only small amounts of Ag, Cd, and In in the coolant indicates little loss of control rod intergrity.

Lew Walton (B&W) discussed TC data trends some more, but most was inferential or conjectural.

Art Lowe (B&W) presented an interesting calculation of components temperatures relative to assumed cladding temperatures as shown in the following table.

Fuel Rod Components	Peak Temperatures at Top of Fuel		
Cladding (Zircaloy-4)	2000°F		2800°F
Fuel (UO_2)	< M.P.		< M.P.
Spacer (Zircaloy-4)	<2000°F		<2800°F
Retainer Spring (304 SS)	<2000°F		<2800°F
Structural Components			
Guide Tubes (Zircaloy-4)	>1500°F		>1950°F
Spacer Grids (Inc-718)	>1800°F	upper	>2200°F
	~1100°F	intermediate	~1500°F
End Fittings (304 SS)	<1500°F		<2200°F
Holddown Spring (Inc-718)	<1500°F		<2200°F
Control Components	~1500°F		~1600°F

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It should be emphasized that the cladding temperatures are assumed.

Another interesting bit of information was the rate at which the decay heat decreased with time from the initial full-power level of 2772 MW (thermal). This is shown below.

Time after scram (hr)	Core Decay Heat MW_{th}
0	168
0.5	37
1	30
2	24
4	19
8	14.5

The core had had 80 full-power days of running time at the time of the incident.

In the afternoon the group broke up into three working groups to discuss (1) the effects of Zircaloy oxidation, (2) fuel pellet behavior, and (3) available diagnostic information. The summaries of these meetings are appended as Enclosures Four, Five, and Six. I was a member of the working group on Zircaloy oxidation. The conclusions and recommendations are as stated in Enclosure Four. One final interesting calculation was brought out in the working group meeting. An NRC staff member had calculated that a total of approximately 380 lb-moles (760 lbs) of hydrogen gas was released by the metal water reaction according to the equation:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2\uparrow + \text{Heat}$$

Included in the 380 lb-mole figure were approximately 226 lb-moles that must have burned during the 28 psig pressure rise in the $2 \times 10^6 ft^3$ containment building. The oxygen content in that building went from 21.2 to 18.9 vol % during the fire.

Lorenz, Pawel, and I voted among ourselves and I lost. Therefore, EPRI has my name as their contact in case a later meeting of this type is to be convened.

DOH:sej

Enclosures

cc: C. R. Brinkman
F. R. Mynatt
DOH-FILE

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MAY 1 1 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 7, 1979

TO: F. R. Mynatt

FROM: R. A. Lorenz

SUBJECT: Summary of Work Performed for Three Mile Island (TMI-2)

1. Make preparations to evaluate fission gas and fission product release measured at TMI-2.
   Requested by Mike Tokar, NRC, via A. P. Malinauskas and T. M. Besmann (4-2-79(?)).
   Also requested by Joel Buchanan (ORNL) for Ed Zebroski (EPRI) (4-2-79(?)).
   Request from Brent Beuscher (B&W) for references to our fission product release work (4-2-79(?)).
   Estimated costs: 5 work days, 2 or 3 phone calls.

2. Attend industry group meeting at B&W, Lynchburg, (4-5-79).
   Requested by E. Zebroski (EPRI) via Chernock (CE), (4-4-79).
   Working groups presented recommendations while at Lynchburg (4-5-79).
   (Copy of recommendations available from R. A. Lorenz).
   Estimated costs: travel expenses ~\$300, working time - 4 days
   2 phone calls

3. Estimate fission gas release that could occur for TMI-2 environment (3000 MWd/MT, H_2-rich steam and water) at high temperatures (1600° to 2000°C).
   Requested by and results reported to R. O. Meyer, NRC Fuels Analysis, (4-10-79).
   Estimated costs: 3 work day, 1 Telecon (copy attached).

RAF:clf

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ERDA 8210
(1-75)
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6. FROM

R. A. Lorenz
ORNL
P. O. Box X, Oak Ridge, TN 37830

7. OFFICIAL BUSINESS (TIME)

(Signature of certifying official) A.M. X P.M.

8. DATE April 10, 1979

9. TO

R. O. Meyer (NRR Phone 2-7603)
Nuclear Regulatory Commission
Washington, D. C. 20545

COMMUNICATION CENTER ROUTING

69 Elite 69 Pica

Data from 53 post-irradiation annealing experiments, conducted by G. W. Parker et al., were analyzed in order to obtain estimates of xenon, iodine, and cesium released from UO_2. These tests were conducted with UO_2 irradiated to burnups of trace to 4000 MWd/MT heated after irradiation in flowing inert atmospheres for 5.5 hr. The tests are summarized on p. 80 of report ORNL-3981. The following numbers are the percentage releases of xenon corresponding to "minimum probable," "most probable," and "maximum probable" release.

At 1600°C the estimated release percentages are 2.2, 5.3, and 11.5. At 1800°C they are 7, 15 and 30. At 2000°C they are 19, 37, and 61. Release percentages for iodine and cesium averaged approximately twice the above values.

R. O. Meyer Telecon numbers: 492-7371
492-8110
492-7617 (24-hr remote)

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MAY 7 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 2, 1979

Cost: ~$6,000

TO: Distribution

FROM: W. D. Shults (4-4881) W. D. Shults

SUBJECT: Analysis of Three-Mile Island Water Sample

This memo summarizes the actions taken and results obtained with the first sample of water sent to ORNL/ACD.

1. We were first contacted by phone on Sunday, April 8, 1979 regarding analysis of a primary coolant water sample (by R. E. Brooksbank and O. O. Yarbro). Many phone calls transpired and many people (see attached list) were involved after that. The sample was ultimately received on April 11, 1979, at 3:30 a.m. at McGhee-Tyson Air Force Base. An ORNL team of health physicists, drivers and riggers met the plane and delivered the sample to ORNL, Bldg. 2026.

2. The sample was 3 ml clear water in a glass vial and read 2.7 R/hr at contact. It was shipped within a small lead pig (2") within a 55 gal drum. No smearable alpha contamination was detected. Slight smearable beta contamination was observed on the vial exterior.

3. Portions of the sample were analyzed by the laboratories within ACD listed below. Several methods were used in some cases. Also, because the sample was alkaline and hence subject to adsorption effects, tests were performed upon the empty vial itself and upon the sample after acidification.

Type of Analyses	Section	Contact(s)
Radioelements, gamma spectrometry	Nuclear and Radiochemical Analysis	J. F. Emery (6-7560) H. H. Parker J. R. Stokely
Mass spectrometry	Mass and Emission Spectrometry	J. A. Carter (4-2448) R. L. Walker
Boron titrimetry, pH, gross activity	Technical Support and Services	W. R. Laing (4-4885) L. T. Corbin

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4. Results and Comments

pH

≈8	Obtained with pH paper; ±0.2 pH unit.

Boron

(a) 3440 ppm via ID/SSMS	Isotope dilution with ^{10}B concentration established by micro-titration of NBS H_3BO_3; range = 3314 to 3565 ppm.
(b) 3220 ppm via micro-titration	Micro-titration via the mannitol procedure; range = 3211 to 3235 ppm.
	NOTE: isotopic distribution of sample boron was "normal" 80:20 = ^{11}B:^{10}B.

Uranium

(a) 110 ppb	Isotope dilution with ^{233}U and analysis by thermal emission mass spectrometry; range = 107 to 114 ppb.
(b) 150 ppb	Via delayed neutron counting of ^{235}U; corrected for Pu contribution using mass spectral isotopic values.
(c) Uranium isotopic analysis	
^{235}U = 2.22 atom %	Range ≐ 2.21 to 2.23%
^{234}U = 0.021 ^{236}U = 0.072 ^{238}U = 97.69	NOTE: these uranium results were obtained on an acidified sample. Basic sample gave lower results by ID/SSMS, DLN, and fluorometry.

Plutonium

(a) 0.24 ppb

Isotope dilution with ^{242}Pu and analysis via thermal-emission mass spectrometry; ORNL's "resin bead" sampling technique was used.

NOTE: these results obtained on acidified sample.

(b) Pu isotopic analysis

^{238}Pu	< 0.1
^{239}Pu	90.3
^{240}Pu	7.8
^{241}Pu	1.8
^{242}Pu	0.1

Sodium

960 ppm

Neutron activation analysis gave 970 ppm; flame emission spectrometry gave 960 ppm.

Lithium

(a) 4.64 ppm

Via isotope dilution and ion microprobe mass spectrometry. Flame emission spectrometry gave $\leq$ 5 ppm.

(b) Lithium isotopic analysis

^{6}Li	$\leq$ 0.02
^{7}Li	99.98

Gross Radioactivity

Alpha = background only

Beta = 1.4×10^{10} dpm/ml

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Radioelements

NOTE: results are μCi/ml, computed to April 11, 1979 @ 0800.

^{131}I	=	8.2×10^3
^{132}I	=	$\leq 2 \times 10$
^{134}Cs	=	8.2×10
^{136}Cs	=	1.1×10^2
^{137}Cs	=	3.3×10^2
^{140}Ba	=	2.4×10^2
^{140}La	=	1.6×10^2
^{99}Mo	=	1.8×10^2
^{89}Sr	=	5.5×10^2
^{90}Sr	=	6.8 ± 0.25
^{58}Co	=	≤ 3
^{141}Ce		None detected (≤ 1)
^{144}Ce		None detected (≤ 20)
^{95}Zr		None detected
^{97}Zr		None detected
^{106}Ru		None detected
^{103}Ru		0.5 μCi/ml detected on vial wall; 0.13 μCi/ml found in 3 ml x 5 M HNO_3 wash.
^{132}Te		None detected in solution; 2.3 μCi/ml detected on vial wall.
^{95}Nb		0.5 μCi/ml detected on vial wall; 0.13 μCi/ml found in 3 ml x 5 M HNO_3 wash solution.
^{3}H		1.2 μCi/ml

Semiquantitative Analysis of Elements Not Listed Above

Via spark source mass spectrometry with enriched ^{57}Fe isotope internal standard.

Element	ppm	Element	ppm	Element	ppm
Ag	≤ 0.2	Mn	0.1	Zn	0.5
Al	10	Mo	1	Zr	≤ 3
Ca	≤ 1	Ni	0.2	Tc	≤ 0.1
Cd	≤ 0.3	P	0.1	S	20
*Cs	8	Rb	1	Se	0.6
Fe	≤ 1	Si	5	^{89}Sr,Y	1
In	≤ 0.05	Sr	0.5	^{131}I	0.9
K	0.4	Te	0.5		

*Isotopic Cs atom %

^{133}Cs	45
^{135}Cs	13
^{137}Cs	42

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Adsorption Study

Enhanced activity on the glass sample vial was observed for ^{132}Te, ^{140}La, ^{140}Ba, ^{95}Nb and ^{103}Ru; adsorption is indicated for these elements and should be circumvented in subsequent analyses.

5. Initial results were reported to Tom Telford of DOE and Trojamowski of NRC by phone on April 12, 1979. Subsequent boron results were reported to Telford on April 13, 1979.

6. Contacts (FTS Harrisburg, PA 590-2200)

NRC

Bob Trojamowski and
Mike Slobedian
(717) 944-0101
(717) 944-2483

Joe Colitz
(717) 944-4646
(717) 944-4649

Richard Emch
FTS 492-8066

DOE

Tom Telford
(717) 944-3583

Joe Hegner and Mr. Weiss
(301) 492-8111

Fritz Wolfe
FTS 233-5555

DOE/ORO

Dick Smith

WDS:sjw

Distribution: R. E. Brooksbank
D. O. Campbell
J. A. Carter
L. T. Corbin
B. Duggins
J. F. Emery
D. E. Ferguson
~~L. J. King~~
W. R. Laing
F. R. Mynatt ✓
J. R. Stokely
D. B. Trauger
A. Zucker

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MAY 15 1979

NUCLEAR DIVISION

INTERNAL CORRESPONDENCE

May 14, 1979

F. R. Mynatt, 9201-3, MS-124, Y-12 (4-0422)

Summary of Three Mile Island Work

Attached is a brief summary of work performed by the Nuclear Engineering Applications Department. Due to the extensive and diverse nature of the results, I have not included them in the summary. However, for each problem we were able to supply all of the requested information. Our records do not include the NRC contracts on problem number 2. Since you made the initial request for these analyses, perhaps you may recall who the NRC contacts were. I have summed all of the charges to account number L-61096-02 for this work. If you need the charges associated with the individual tasks, I will be glad to break them out for you.

G. E. Whitesides

G. E. Whitesides, 6025, MS-1M, X-10 (4-5267)
Computer Sciences Division

GEW:bn

File - NoRC

1915 226

Summary of Analyses Performed for the Nuclear Regulatory Commission
on the Three Mile Island Plant

Nuclear Engineering Applications Department
Computer Sciences Division

Problem 1 - Radiation Exposure to Instrumentation on the Pressurizer Tank

A. Dose Rate At Surface of Tank Given the Interior Water-Bourne Sources.
Requested by: Rod Satterfield, NRC
Incident Control Center

Reported to: Rod Satterfield, Carl Neal

Source spectrum calculated by O. W. Hermann using ORIGEN-S.
Shielding analysis performed by G. E. Whitesides, R. M. Westfall using ANISN.

B. Dose Rate At Instrument Bank on Lower Containment Wall Given the Air and Water-Bourne Sources.
Requested by: Rod Satterfield

Reported to: Carl Neal

Source spectra calculated by O. W. Hermann using ORIGEN-S.
Shielding analysis performed by J. V. Pace and R. L. Childs using DOT-IV.

Problem 2 - Determine Xe-135 Source Strengths in 1 and 2 Inch Schedule 40 Pipes for Given Dose Rates at Outer Surfaces.

Requested by:

Reported to:

Hand calculations by O. W. Hermann.
ANISN analyses by J. R. Knight.

Problem 3 - Inventory of All Radioactive Isotopes in the Reactor at the Time of the Accident and at Given Time Intervals after Shutdown.

Requested by: Dennis Rathbun, Office of Policy Evaluation, NRC

Reported to: Dennis Rathbun

ORIGEN-S calculations by O. W. Hermann

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ESTIMATED COSTS

63 Man Hours	$1884
Computer Charges	183
	$2067

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 17, 1979

TO: F. R. Mynatt

FROM: D. E. Bartine DEB

SUBJECT: Three Mile Island Work

Reporting Past Efforts

Sorry these are a few days late, they were quite an organizational effort. Hope the delay causes no problems. Would you object to our presenting some of this work at the Atlanta ANS meeting? (RP&S division meeting)

Current Work Efforts

As a result of a meeting with Jim Snider at 1:00-3:30, 5/15/79, the following items are still active:

1. General problem of verifying sources in the individual storage tanks in the auxiliary building. (Consists of calculations of projected dose levels based on hypothesized coolant content and comparison with measurements and/or analyses.)

2. Determine the curie content of fission products allowed in the ion-exchange resin barrel in order to meet 0.5 mrem/hr unlimited access level at the overlook from a "bare-pick" of the resin barrel. Also wants dose as function of dis tance from barrel for above sequence (relates to courtyard areas and to close-in barrel decoupling requirements).

3. Determine concrete roof thickness for evaporator in spent fuel storage pool, to contain full core inventory at 90 days from 3/28.

4. Investigate radiation dose leakage from the spent fuel handling pool through the gate leading to the spent fuel storage pool.

In addition, Robbie called Doug Selby today and indicated that EPRI has designated him the task of explaining the BF_3 detector readings. He is apparently planning to request our assistance and will be contacting you.

Radiation Hardening

In a related issue, my contact at DNA (Major Mike Kemp) would like to propose that they develop a process to harden the components for the

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pressurizer transducer. Estimated cost $200-400K, unclear whether they would insist on "outside" (DOE and/or NRC) funding or whether they could dig up DNA internal funding if the work were requested by NRC or DOE. Any suggestions as to how to proceed from here?

DEB:nc

Attachments (11)

cc: T. J. Burns
R. L. Childs
W. W. Engle
D. T. Ingersoll
F. C. Maienschein
J. V. Pace
D. L. Selby
A. Zucker

1915 229

ASSISTANCE PROVIDED BY THE REACTOR PHYSICS AND SHIELDING GROUP FOR THE THREE MILE ISLAND PROBLEM

I. Pressurizer Transducer Dose

A. Statement of Problem

To determine the dose received by the pressurizer transducers so that an estimate of their functional lifetime could be obtained. The transducers are located in a basement hallway in the primary containment building and are necessary for the control of the system pressure since they indicate the level of water in the pressurizer. The transducers were tested for a 10^5 rads exposure level.

The basement hallway (13' wide, 22' high) was filled with 2' of water. Radiation sources assumed were primary coolant at 52 hours (diluted 2/1) for the water and an air sample for the primary containment building at 84 hours.

B. Work Performed

A 2-D (x-y) DOT discrete ordinates calculation was performed to determine the dose at the detector positions for the sources given. A time-dependent estimate of the air and water sources was obtained to use in determining the time-integrated dose received. The dose received from the air and water sources was plotted separately as a function of time. The integral dose for air, water, and air and water sources was also plotted as a function of time (see attachments).

C. Results

The dose rate calculated from the sample sources was about 1 rad/sec from the water at 52 hours and 1.75 rads/sec from the air at 84 hours for a combined total of $\sim$10,000 rads/hr. The integral dose at the time of the calculation ($\sim$6 days) was $\sim 2\times10^6$ rads, indicating that either the transducers were operating far beyond their specified lifetimes or that the sources used for the calculation were large. However, the calculations also indicated that in going from 6 days to 20 days after the accident, the integral dose would only increase from 2×10^6 rads to 3×10^6 rads, i.e., that the transducers had already received most of the integral dose which they would receive from the current source).

D. Work Requested By

George Knighton, NRC, 4/3/79

E. Results Reported To

George Knighton, NRC, 4/4/79

1915 230

F. Work Performed By

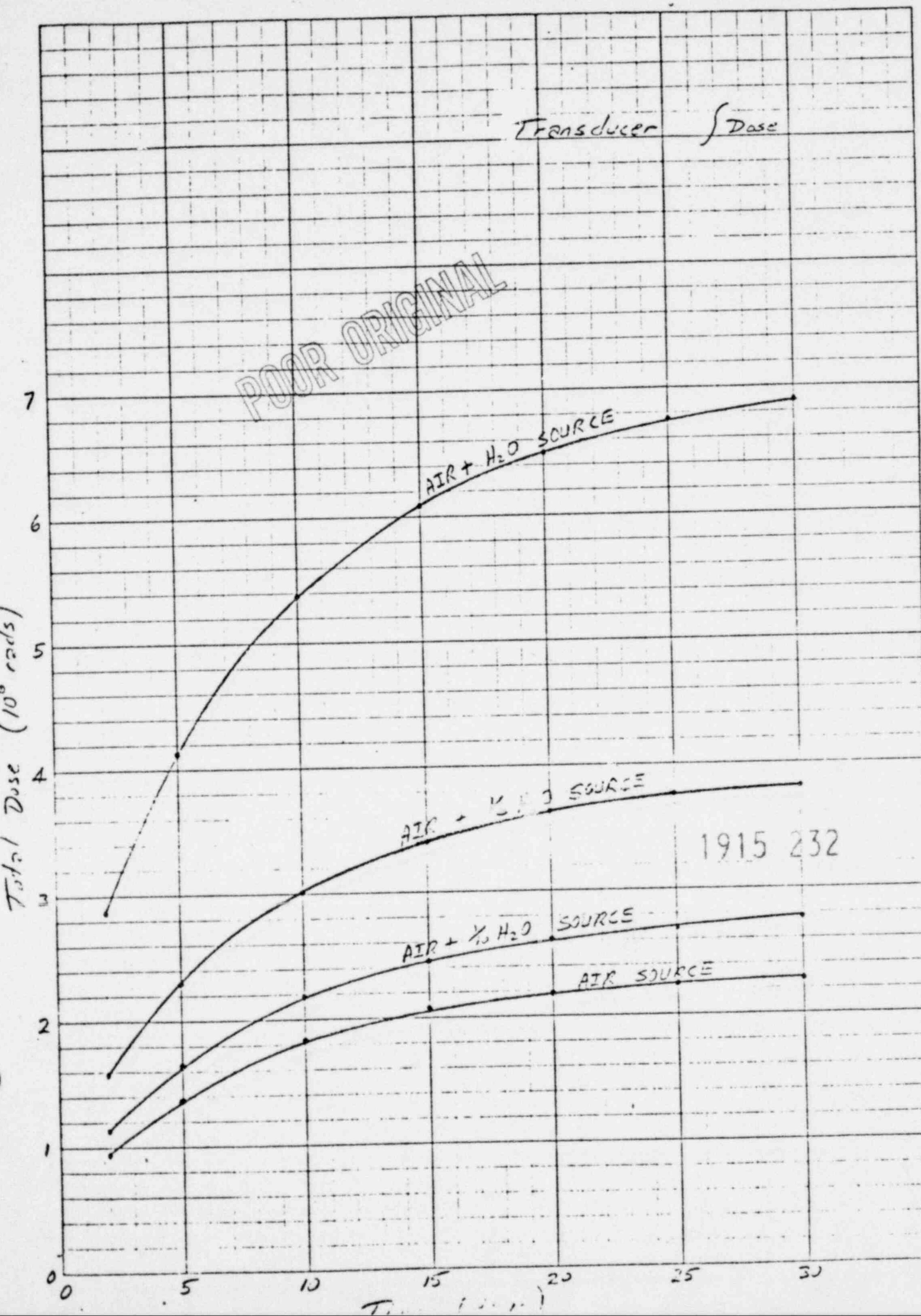
D. E. Bartine EPD
T. J. Burns EPD
R. L. Childs CSD
W. W. Engle EPD
J. V. Pace CSD

G. Estimated Cost

$2,000

1915 231

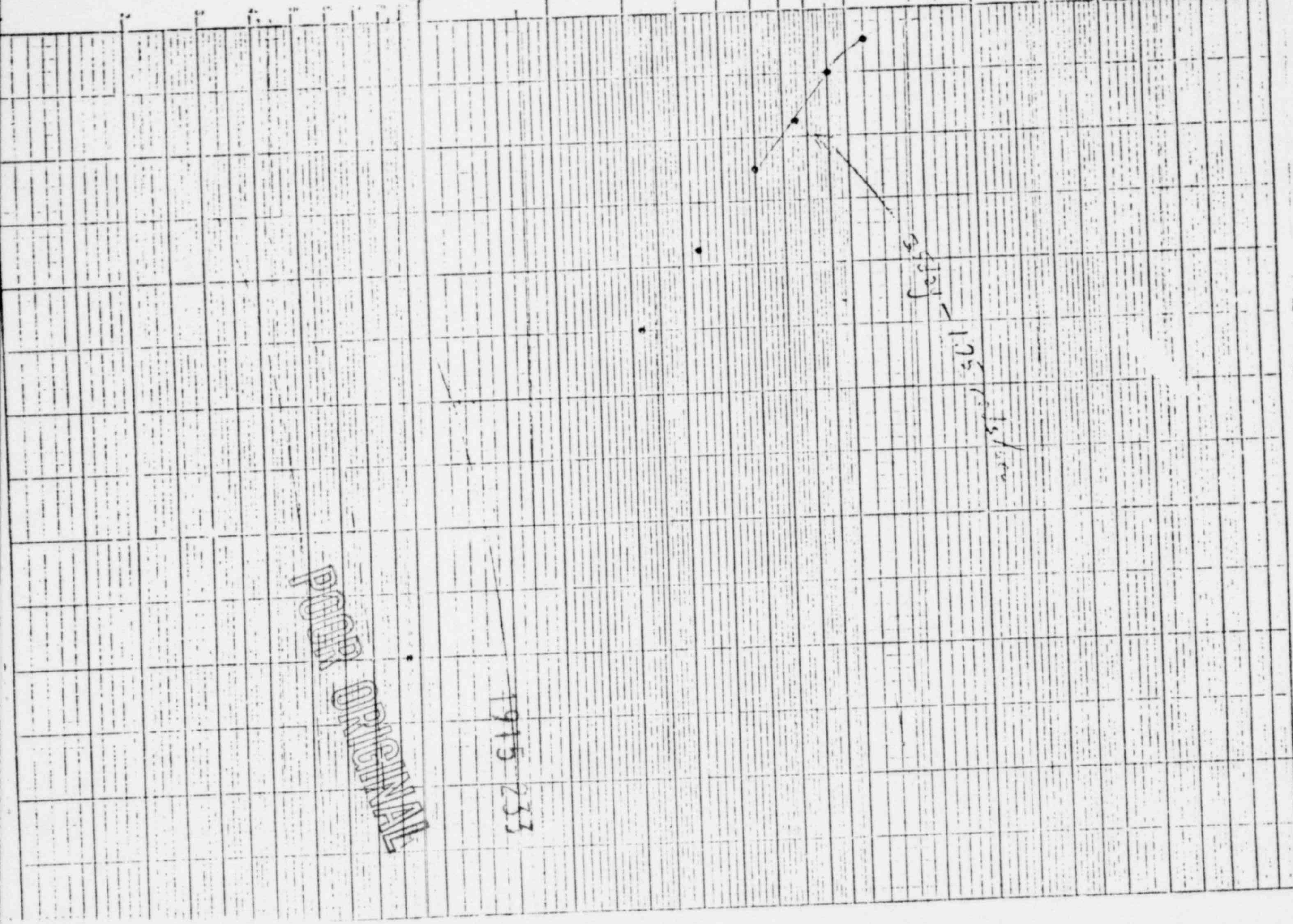
5/1/79



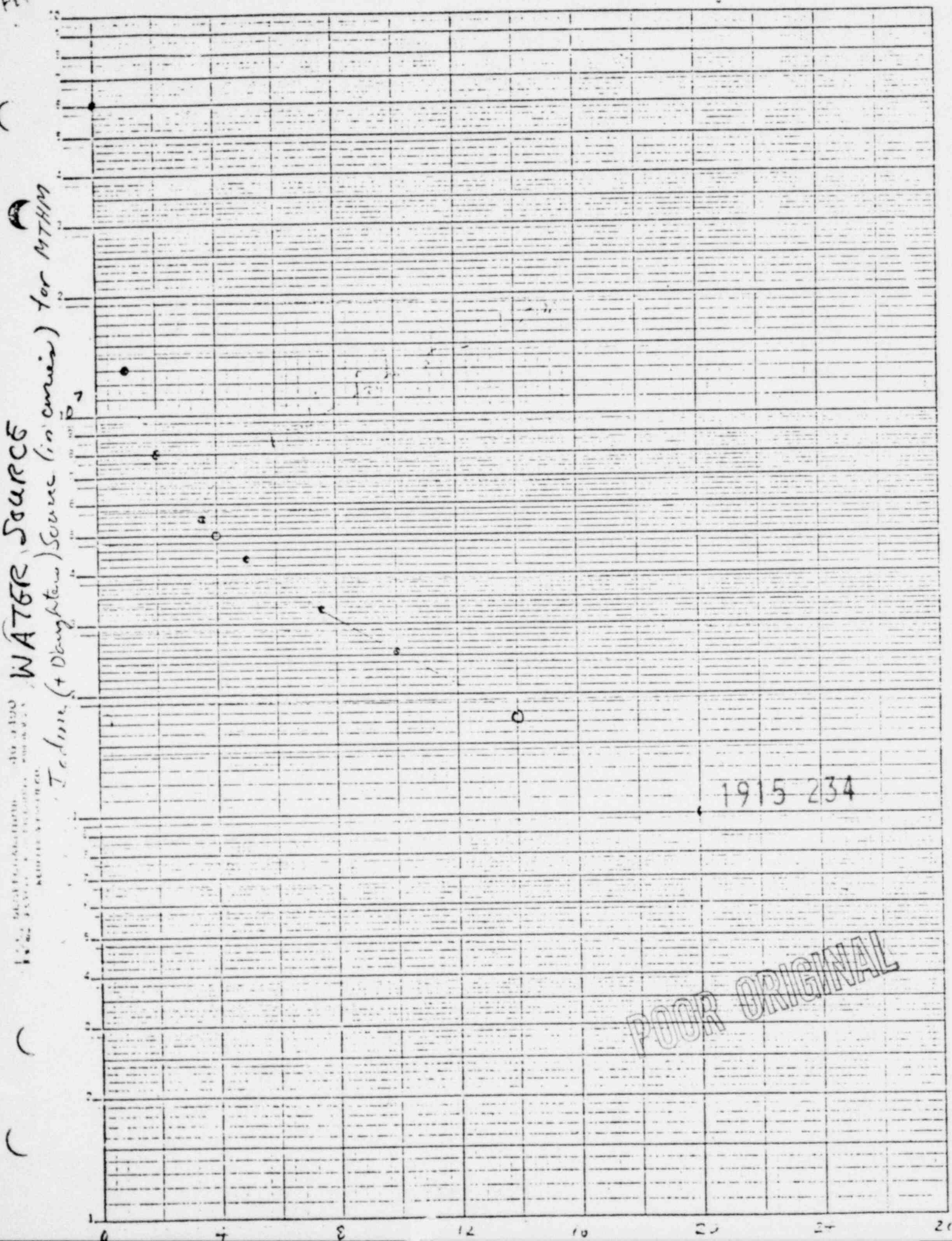
1915 232

Air Source

Xenon + Krypton (+ Daughters) Source (in curies) for 5 MTHM



1915 233


WATER SOURCE
Iodine (+ Daughters) Source (in curies) for MTHM
0
4
8
12
16
20
24

1915 234

II. Core Isotopics and Decay Heat

A. Statement of Problem

Determination of isotopic concentrations and decay heat rate in the core at the time of the accident and for the following 60 days. The isotopic concentrations were of interest for determining potential levels of contamination, and the decay heat rate indicated cooling requirements.

B. Work Performed

ORIGEN calculations were performed for both a short term (0-84 hours) and a long term (0-60 days) case. The case assumed 80 full power days with a burnup of approximately 3000 MW_{th} days/M.T. heavy metal. The total core inventory was 82.06 M.T., and the specific power was 33.8 MW/M.T. Since the initial runs were made with 37.5 MW/MT, a correction factor was supplied with the case.

C. Results

The ORIGEN runs were disseminated, with the isotopics, thermal decay heat ($\beta+\alpha$), α power, and source terms (photons/sec) for use in later transport runs. The total decay heat production (MW_{th}) varied from 168 MW at the time of the accident to 5 MW at the time of the calculation (7 days after the accident) to 2 MW at 30 days, and then stabilized at about 1 MW for 60 and 90 days.

D. Work Requested By

Floyd Culler and Ed Zebrowski, EPRI; Les Oakes, M&C; 4/4/79

E. Results Reported To

Floyd Culler, Ed Zebrowski, EPRI; Les Oakes, M&C; 4/5/79

F. Work Performed By

T. J. Burns, EPD

G. Estimated Costs

$500

III. Tank Car Storage

A. Statement of Problem

Determine the dose rate in the vicinity of a railroad tank car containing water from the auxiliary building to determine the feasibility of temporary storage in this mode.

B. Work Performed

ANISN calculations were performed to determine the dose from a cylindrical tank. A source term was provided for the auxiliary building water at 14 days after the accident.

C. Results

The results indicated a contact dose of 40 rem/hr, 100 mrem/hr at 200', 2.5 mrem/hr at 900', and 0.1 mrem/hr at 2000'. The contact dose was too high for the specified auxiliary building water to be put into the tanks at the time.

D. Work Requested By

Frank Ackstulewicz, Accident Analysis Branch, NRC, 4/11/79

E. Work Reported To

Frank Ackstulewicz, NRC, 4/11/79

F. Work Performed By

D. E. Bartine	EPD
W. W. Engle	EPD
D. T. Ingersoll	EPD
J. V. Pace	CSD
R. L. Childs	CSD

G. Estimated Costs

$1,500

IV. Steam Generator Repair

A. Statement of Problem

Determination of the dose in the area of pipes carrying water from the secondary side of the "B" loop steam generator. The intent was to use the steam generators, connected to the condenser, as coolant mechanisms for a natural convection no-pressurizer "cold shut-down" operational mode.

B. Work Performed

An acitvity source for the secondary loop was obtained (activity resulting from tube ruptures in the primary side), and a dilution factor of 3 was assumed.

ANISN calculations were performed to obtain dose rates around infinite cylinders representing steel pipes with 12", 16", and 20" internal diameters.

C. Results

Doses were reported in mrem/hr at contact and 2 1/2', 4 1/2', 6', and 10' from the pipes. The dose rate was about 50 mr/hr for the 12" pipe carrying the dilute solution and roughly fell off as 1/r from the pipe centerline for all three cases.

D. Work Requested By

Frank Ackstulewicz, NRC, 4/16/79

E. Work Reported To

Frank Ackstulewicz, NRC, 4/17/79
Richard Emsch, NRC, 4/18/79

F. Work Performed By

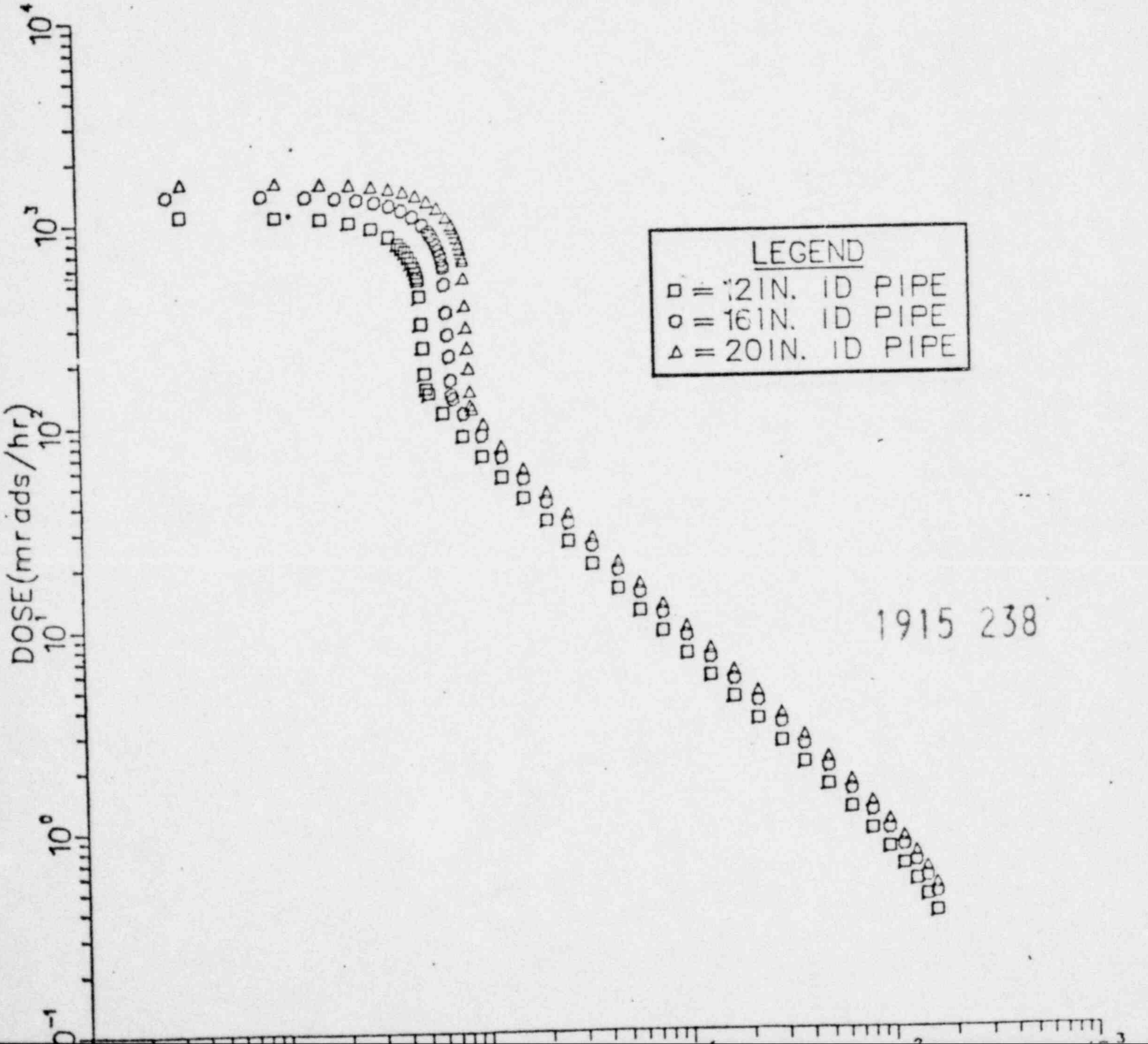
D. E. Bartine	EPD
D. T. Ingersoll	EPD
J. V. Pace	CSD

G. Estimated Costs

$750

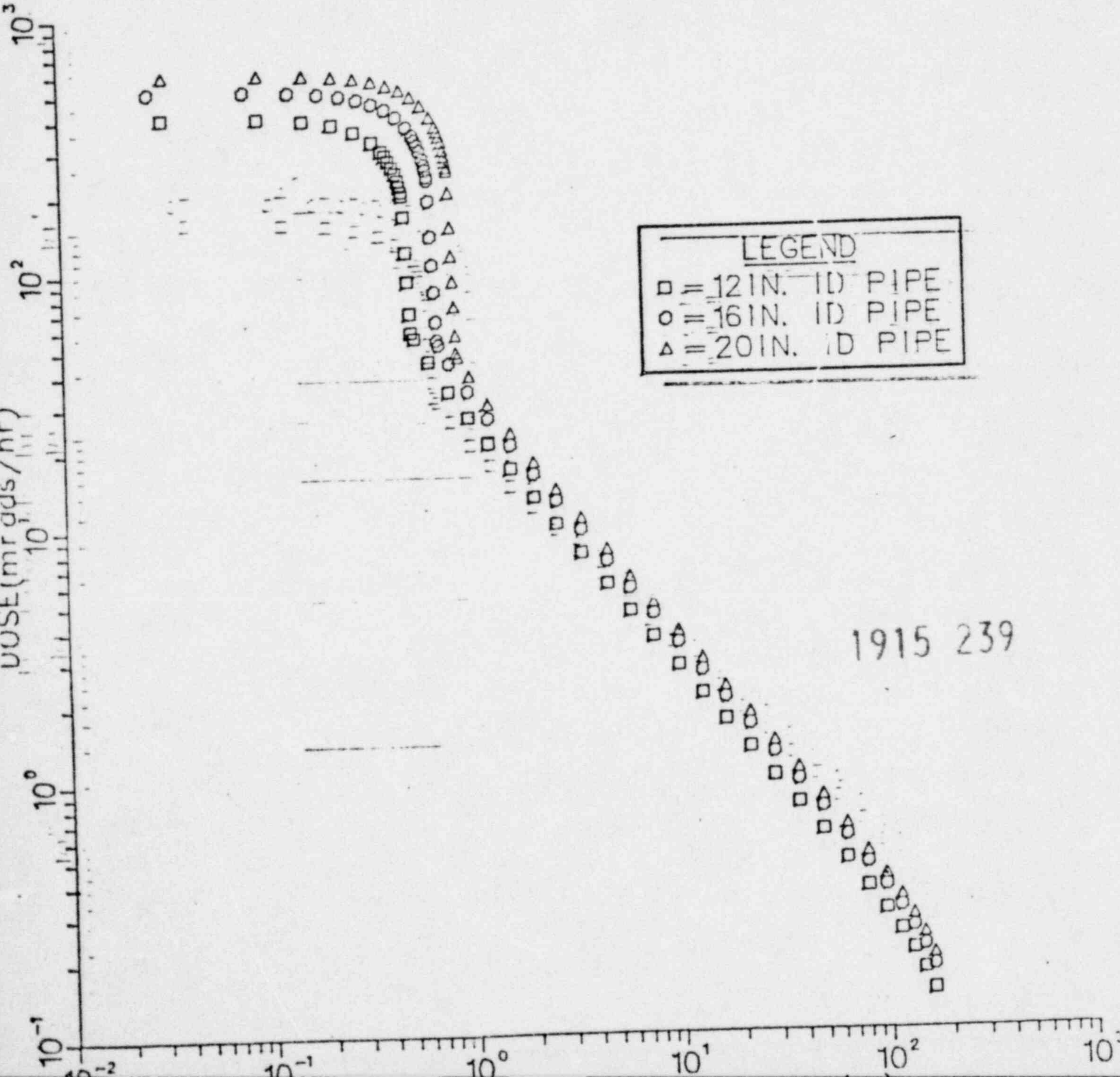
1915 237

C/T GAMMA DOSE VS DISTANCE FROM PIPE CL



C/T GAMMA DOSE VS DISTANCE FROM PIPE CL

(Source multiplied by 65/170)



1915 239

V. Tank Car Recalculation

A. Statement of Problem

Determination of the dose rate in the vicinity of railroad tank cars if they were to contain water from the auxiliary building. This is a repeat of problem III with an additional source term, and an attempt to obtain a more exact answer, since the first set of results was obtained from a quick, conservative approach.

B. Work Performed

A cylindrical ANISN calculation was performed for the new source, followed by a spherical ANISN case conserving the surface area. A 10,000 gallon tank car was used instead of a 20,000 gallon car.

C. Results

The new results indicated about 20 rem/hr contact, 100 mrem/hr at 85', 2.5 mr/hr at 550', 0.3 mr/hr at 1000' from a 10,000 gallon car.

D. Work Requested By

Frank Ackstulewicz, NRC, 4/16/79

E. Work Reported To

Frank Achstulewicz, NRC, 4/17/79

F. Work Performed By

D. E. Bartine	EPD
T. J. Burns	EPD
R. L. Childs	CSD

G. Estimated Cost

$750

1915 240

VI. Time-Dependent Tank Car Dose

A. Statement of Problem

Determination of the dose rate in the vicinity of railroad tank cars if containing auxiliary building water as a function of time.

B. Work Performed

Calculations were performed to obtain source terms for the auxiliary building water as a function of time. ANISN calculations, cylindrical and spherical, were then performed to determine the dose at the time of the analysis (14 days after the accident) and 30, 60, and 90 days after the analyses. Calculations were run for both the 10,000 and 20,000 gallon cars.

C. Results

Doses were reported at contact, 1,000', 2,000', and the distances were given for 100 mrem/hr and 2.5 mrem/hr dose rates. The contact level was 19 rem/hr at T=0 (14 days) and dropped off to ∿ 750 mrem/hr at T=60 days, but was still ∿600 mrem/hr at T=90 days. The implication is that the material would require shielding before shipping, either in shielded tank cars, or the hot isotopes could be concentrated and shipped in heavily shielded containers such as shipping casks or spent fuel storage casks. To obtain the 200 mrem/hr surface dose (and <100 mrem/hr at 3') for shipping regulations would require a Pb liner about 1" thick for the tank car.

D. Work Requested By

Richard Emsch, Div. Operating Reactors, NRC, 4/18/79

E. Work Reported To

Richard Emsch, NRC, 4/19/79

F. Work Performed By

D. E. Bartine	EPD
W. W. Engle	EPD
T. J. Burns	EPD
R. L. Childs	CSD

1915 241

G. Estimated Cost

$1,500

VII. Tank-Car Algorithm

A. Statement of Problem

Define a simplified method for determining the dose rate level in the vicinity of a 10,000 gallon railroad tank car containing fission product-contaminated solutions, such as the water in the auxiliary building or other dilutions of the primary coolant.

B. Description of Work Performed

Gamma sources were obtained for several individual nuclides, namely, ^{131}I, ^{133}I, ^{134}Cs, ^{136}Cs, ^{140}Ba, and ^{140}La. ANISN calculations (cylindrical and spherical) were performed for 10,000 gallon tank car configurations for each gamma group (using a source strength of 1 photon/cm^2/sec) in which any of these isotopes has a source. The sources for each individual isotope were then folded with the individual group calculations to obtain the doses.

C. Results

Results were reported in mrem/hr at contact and distances of 20' to 1,000' from the tank car for 1 microcurie/cc solutions of each of the above isotopes (see attached tables). The tables will be used by multiplying the values given by the measured concentration of a given isotope and summing over the isotopes. A table giving the same information for an infinite cylinder (simulating a long train of such tank cars) was also included.

D. Work Requested By

Richard Emsch, NRC, 4/19/79

E. Work Reported To

Richard Emsch, NRC, 4/26/79

F. Work Performed By

D. E. Bartine	EPD
R. L. Childs	CSD
W. W. Engle	EPD
D. T. Ingersoll	EPD

G. Estimated Cost

$1,200

1915 242

DOSE IN mR/HOUR AS A FUNCTION OF DISTANCE FROM A
10,000 GALLON CAR*PER MICROCURIE/CC OF:

	I^{131}	I^{133}	Cs^{134}	Cs^{136}	Cs^{137}	Ba^{140}	La^{140}
contact	121.0	274.0	728.0	1080.0	199.0	91.6	1250.0
20 feet	9.76	21.7	57.4	84.5	15.9	7.33	96.2
40 feet	2.49	5.42	14.3	20.9	3.98	1.85	23.6
60 feet	1.16	2.48	6.52	9.46	1.83	0.852	10.8
80 feet	0.664	1.40	3.68	5.31	1.04	0.486	5.93
100 feet	0.443	0.926	2.42	3.47	0.684	0.322	3.86
150 feet	0.210	0.429	1.12	1.58	0.319	0.151	1.74
200 feet	0.122	0.245	0.634	0.892	0.182	8.69E-2	0.972
400 feet	3.02E-2	5.92E-2	0.153	0.212	4.44E-2	2.13E-2	0.227
600 feet	1.14E-2	2.27E-2	5.88E-2	8.19E-2	1.70E-2	8.10E-3	8.87E-2
1000 feet	2.26E-3	4.86E-3	1.28E-2	1.85E-2	3.58E-3	1.67E-3	2.10E-2

*(Sphere approximating 102" i.d., ~24' long cylinder)

1915 243

DOSE IN mR/HOUR AS A FUNCTION OF DISTANCE FROM
AN INFINITE CYLINDER* PER MICROCURIE OF:

	I^{131}	I^{133}	Cs^{134}	Cs^{136}	Cs^{137}	Ba^{140}	La^{140}
contact	121.0	274.0	728.0	1080.0	199.0	91.6	1250.0
20 feet	23.9	51.7	136.0	198.0	38.0	17.7	224.0
40 feet	12.9	27.2	71.3	103.0	20.1	9.43	114.0
60 feet	9.08	18.8	49.1	70.1	14.0	6.58	77.4
80 feet	7.02	14.4	37.4	53.0	10.7	5.05	58.2
100 feet	5.76	11.7	30.4	42.8	8.71	4.13	46.8
150 feet	3.94	7.88	20.4	28.6	5.88	2.80	31.0
200 feet	2.92	5.81	15.0	20.9	4.35	2.07	22.6
400 feet	1.21	2.41	6.24	8.72	1.80	0.857	9.46
600 feet	0.583	1.20	3.14	4.46	0.894	0.421	4.94
1000 feet	0.153	0.347	0.922	1.39	0.252	0.116	1.65

*Infinite cylinder is a conservative approximation of the effect of multiple tank cars in train (as opposed to single car source).

VIII. BF_3 Detector Readings

A. Statement of Problem

The BF_3 detectors, located in the cavity outside the reactor pressure vessel but inside the primary containment, are giving high readings which are dropping with time. (Normal readings ~3 cts/sec, current reading 35 cts/sec, was 50 cts/sec at 52 hrs.) The problem is to attempt to account for the readings.

Two basic possibilities were investigated:

1. high levels of neutron production from (γ,n), spontaneous fission, and (γ,n) reactions
2. high gamma dose levels giving false neutron readings

B. Work Performed

A search was made for cross sections resulting in neutron production representing reactions which might be occurring in the reactor core and primary coolant. The resulting neutron flux level at the detector position was then calculated.

An estimate of the gamma flux was obtained along with an indication of the possible effect of these gammas on the detector reading (normally subtracted as background). The dose to the gamma detectors present further out in the primary containment was also estimated.

C. Results

The neutron flux levels at the BF_3 detectors was estimated to be 0.1 neutrons/cc/sec, much too low to account for the observed readings. The gamma dose at the detector readings was estimated to be ~10,000 rem/hr, much higher than the 27 rem/hr level which was reportedly being assumed in interpreting the BF_3 detector readings. It therefore appeared that the high gamma field might be responsible for the high detector readings, although there may be other, equally valid explanations.

D. Work Requested By

R. Ball, B&W through J. Robinson, TEC, 4/17/79

E. Work Reported To

J. Robinson, TEC, 4/20/79

F. Work Performed By

D. E. Bartine	EPD
T. J. Burns	EPD
D. T. Ingersoll	EPD
D. L. Selby	EPD
J. V. Pace	CSD

1915 245

G. Estimated Cost

IX. Ion-Exchange Resins

A. Statement of Problem

Determine the allowable concentration of fission products in ion exchange barrels. A dose limit of 40 rem/hr at 1', which is the current operating limit, was assumed.

B. Work Performed

Calculations were made for two different ion trap models, one for iodine and the other for the metal ions such as Cs. The barrel was assumed to be filled with H_2O, and were 6' i.d. by 6' long of 1/4" iron. A constant distribution of source was assumed, and a total number of curies was calculated.

C. Results

The cation (metal ions) barrel was estimated to contain 790 curies, which corresponds to the content of 350 gallons of primary coolant at April 30. By comparison, the barrel volume would hold 1270 gallons of primary coolant, so that running to the above concentrations would multiply the volume of active material by ~4. The iodine barrel would only accommodate the content of 250 gallons of primary coolant (~2,000 curies) on April 30, but should be able to accommodate 1,000 gallons by June 1. Since the volumes of contaminated resins resulting from the separation investigated here would be overwhelming, the use of shielded containers or of evaporators will be considered.

D. Work Requested By

J. Levendusky, EPICORE; J. Snider, ORNL; 4/27/79

E. Work Reported To

J. Snider, ORNL; 5/1/79

F. Work Performed By

D. E. Bartine EPD
T. J. Burns EPD

G. Estimated Cost

$375

X. Pipe Transport of Primary Coolant

A. Statement of Problem

Determination of the dose in the vicinity of pipes carrying primary coolant. In order to move contaminated coolant into longer-term storage facilities, for example in the spent fuel handling pool, it would be necessary to pipe the coolant through existing passageways. Many of these, especially stairwells, would need to be available for personnel use, so that the dose rate levels anticipated in these passageways is important for planning considerations.

B. Work Performed

One-dimensional ANISN discrete ordinates calculations were performed to obtain the dose rate as a function of distance from contact out to 10'. These calculations were performed for bare pipes and for pipes with 1/2", 1", and 2" lead coverings. In a separate set of calculations, those doses were also obtained as a function of time, extending from T = 0 (5/30/79) to T = 90 days.

C. Results

The contact dose calculated without a lead covering was 100 rem/hr. The contact doses for 1/2", 1", and 2" lead coverings were 50 r/hr, 11r/hr, and 1.3 r/hr, respectively. In all cases, the dose outside the pipe fell off by a 1/r factor from the pipe centerline, for example, the uncovered pipe gave a dose of 9 r/hr at 1', 2 r/hr at 5', and 1r/hr at 10'. A plot of the results is included.

D. Work Requested By

J. Snider, ORNL: A. Cyzelman, Burns & Roe; 4/27/79

E. Work Reported To

J, Snider, ORNL; A Cyzelman, Burns & Roe; 5/7/79

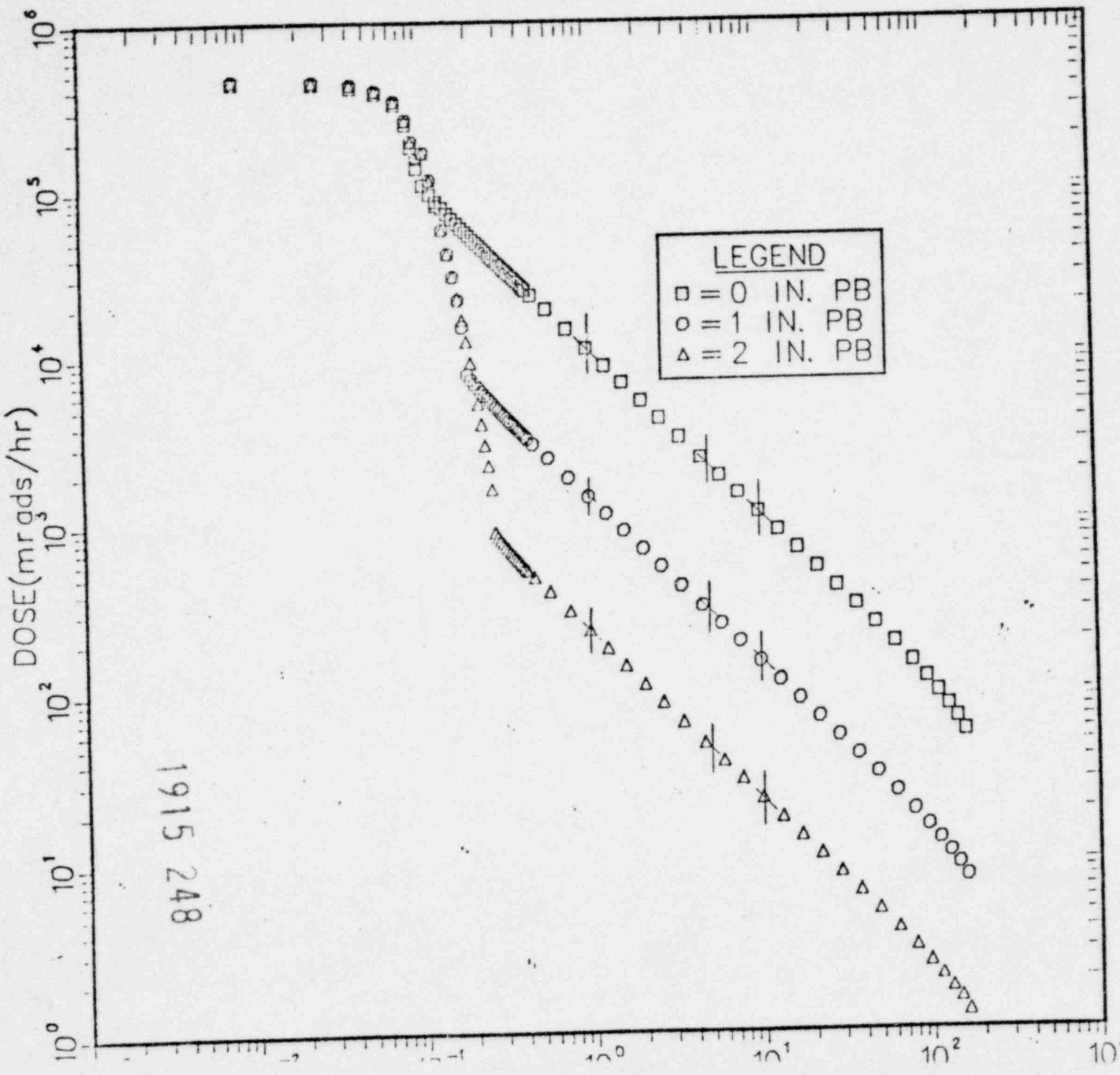
F. Work Performed By

D. E. Bartine	EPD
W. W. Engle	EPD
J. V. Pace	CSD

G. Estimated Cost

$1,000

C/T GAMMA DOSE VS DISTANCE FROM 2IN. PIPE ℄



XI. Tank Storage in Spent Fuel Handling Pool

A. Statement of Problem

Determine the roof (concrete) thickness required in order to use the spent fuel handling pool for tank storage of primary coolant. Two 25,000 gallon tanks are placed in the bottom of the 40' deep concrete pool, with four 15,000 gallon tanks suspended from an I beam near the top of the pool. The concrete roof will be placed on top of the I beams.

B. Work Performed

ANISN one-dimensional discrete ordinates calculations were performed to determine the dose resulting from filling the bottom tanks with primary coolant and from filling the top tanks with primary coolant and with diluted primary coolant. Estimates of the dose as a function of time (4/30, 5/31, 6/30) were made for concrete thicknesses of 0, 1', 2', 3', and 4'.

C. Results

Utilizing the primary coolant source from the 4/14 analysis and decaying it to 4/30 showed a dose rate of 1280 rem/hr at the pool top without any concrete shield. Of this dose, 1200 rem/hr came from the top tank layer and 80 rem/hr from the bottom layer. The total doses through 1', 2', and 3' of concrete were 15 rem/hr, 170 mrem/hr, and 1 mrem/hr. These results indicate that 2' or 3' of concrete should be sufficient (for limited access areas) by the time the tanks are likely to be filled.

D. Work Requested By

J. Snider, ORNL; 4/27/79

E. Results Reported To

J. Snider, ORNL; 5/7/79

F. Work Performed By

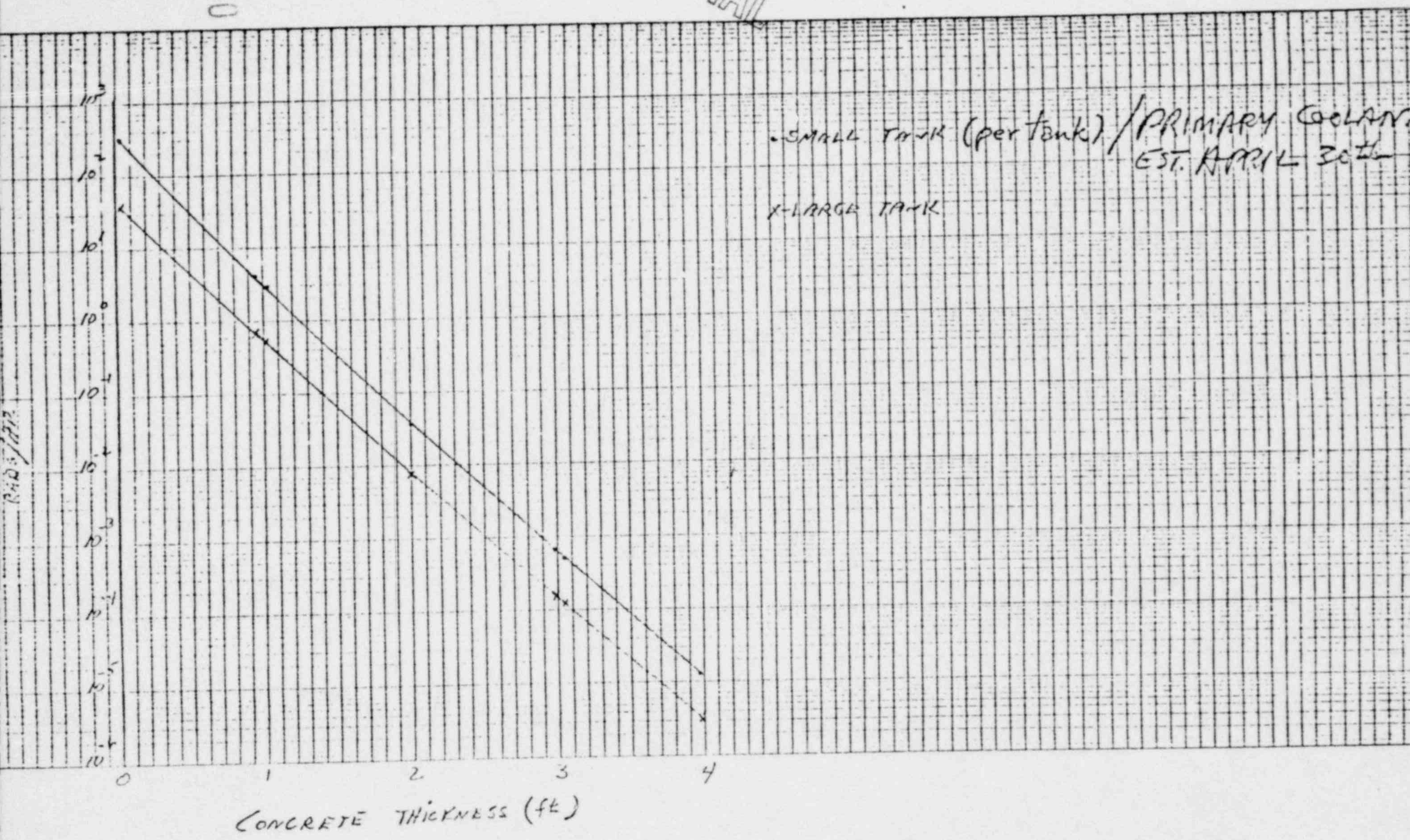
D. E. Bartine	EPD
D. T. Ingersoll	EPD
J. V. Pace	CSD
R. L. Childs	CSD

G. Estimated Costs

$1,500

POOR ORIGINAL

1915 250



Distance in Cone	Large tank (R/hr)	Small Tank (R/hr)
0'	3.9+1	3.0+2
1'	6.0-1	3.5+0
2'	7.7-3	3.8-2
3'	1.4-4	6.1-4
4'	3.1-6	1.2-5

MAY 14 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 10, 1979

To: F. R. Mynatt

From: G. H. Jenks (6-2576) GJ

Subject: Report Summarizing My Assistance in Response to the Three Mile Island Accident

The information which you requested in your May 2, 1979, letter is enclosed.

GHJ:dd

Enclosures

1. Problem Summary, "NRC Request Through D. B. Trauger on April 2, 1979, for Information on H_2 Solubility in Water at TMI and on Transfer and Stripping of H_2 Via Side Stream Flow of Reactor Water Through Pressurizing"
2. Problem Summary, "Consultation on Radiolytic Gas Generation and Recombination Within Reactor at TMI"

cc/enc: D. E. Ferguson
W. C. McClain

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Problem - NRC Request Through D. B. Trauger on April 2, 1979, for Information on H_2 Solubility in Water at TMI and on Transfer and Stripping of H_2 Via Side Stream Flow of Reactor Water Through Pressurizing

Work and Conclusions - The following information was assembled and reported.

1. Total volume of H_2 dissolved in reactor main stream water under following assumed conditions:

   a. Uniform concentration of H_2 throughout main stream water

   b. Total volume of main stream water = 12,000 ft^3

   c. Equilibrium pressure of dissolved H_2 = 1000 psi

   d. Temperature of main stream water = 138°C

   The solubility of gases can be expressed as,

   α = psi per cm^3 of gas(STP) per kg of water

   at 138°C, α for H_2 = 0.62.

   Then H_2 concentration in water = 1.6×10^3 cm^3(STP)/kg

   and total amount of H_2 = $(1.6 \times 10^3)(3.15 \times 10^5) = 5.1 \times 10^8$ cm^3(STP)
   = 18,000 ft^3(STP)

2. Temperature of minimum solubility of H_2 in water

   this is at ~60°C where α = 0.995.

   (at 100°C, α = 0.80)

3. Rate of transfer of H_2 out of main stream with gas-stripped side stream of 15 gpm or 50 gpm and times to effect given reduction of (H_2) in main stream.

$$\frac{d(H_2)}{dt} = -k(H_2) \qquad (1)$$

$$\frac{(H_2)}{(H_2)_o} = e^{-kt} \qquad (2)$$

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or

$$t = -\frac{1}{k}\,\ell n\frac{(H_2)}{(H_2)_o} \qquad (3)$$

where

$k = 1.67 \times 10^{-4}$/min at 15 gpm

$k = 5.57 \times 10^{-4}$/min at 50 gpm

Time to Effect Given Reduction in (H_2)

Item	Required Ratio $(H_2)/H_2)_o$	Time (days) 15 gpm	Time (days) 50 gpm
Pressure of dissolved H_2 reduced from 1000 psi to 300 psi at 138°C	0.30	5.0	1.5
Pressure of dissolved H_2 reduced from 1000 psi to 300 psi at 60°C	0.19	7.0	2.1
Pressure of dissolved H_2 reduced from 1000 psi to 300 psi at 100°C	0.23	6.1	1.8

4. Gas transfer into gas space of pressurizer at Three Mile Island by 15 gpm side stream

Assumed conditions in pressurizer:

a.	Temperature	285°C
b.	Temperature of entering side stream	138°C
c.	Mass of boiling liquid	1.4×10^4 kg
d.	Volume of steam dome	1.75×10^7 cm^3
e.	Heat loss from walls of steam dome	$\leq$25 kcal/sec
f.	By-pass liquid enters pressurizer near top of steam dome	
g.	The steam dome is free of permanent gas at the start of by-pass flow	
h.	Initial concentration of H_2 in main stream	1.6×10^3 cm^3(STP)/kg

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I analyzed the gas distribution conditions in the pressurizer using the results of a theoretical analysis which I reported about 10 years ago (G. H. Jenks, "Gas Distribution in PWR Steam Pressurizers," in Reference State-of-the-Art Design and Technological Assessment of a 1500-kW(e) Pressurized-Water Reactor, G. Samuels, Program Director, ORNL-TM-2604, August 1969, Appendix H, p. 365).

I concluded that essentially all of the gas which enters the pressurizer within the side stream during a period of about one day would collect in the gas-vapor phase, i.e., in the steam dome. The fraction collecting in the steam dome from additional by-pass flow beyond one day without venting would be <1.

Conceivably, the H_2 (or other permanent gas) could be vented from the pressurizer into the surroundings at intervals as needed or desired by the operators.

Other pressurizer and by-pass flow conditions could be analyzed by the methods used here and described in the above reference.

Requestor - Tom Murely (NRC) through D. B. Trauger on April 2, 1979.

Results Reported To - Tom Murely (NRC) by phone on April 3, 1979. Also, I reported this work to D. B. Trauger in a letter dated April 4, 1979.

Workers - G. H. Jenks

Estimated Costs - About 1 day of my time

Problem: Consultation on Radiolytic Gas Generation and Recombination Within Reactor at TMI

Descriptions of Work Performed - I worked with Joel Buchanan and others at NSIC on Sunday, April 1, 1979, as reported in a letter from Buchanan to Cottrell, dated April 30, 1979, "Technical Assistance on Three Mile Island Accident." (A copy of this letter is attached).

Also, Ed Zebrowsky called me early on Monday, April 2, 1979, and we further discussed these radiation chemistry questions, and I described several published papers which he could consult. Following our discussions, I suggested he talk with C. J. Hochanadel (Chemistry Division), and I transferred Zebrowsky's call to Hochanadel.

I also discussed radiolytic decomposition and recombination of water and references with Tom Murely (NRC) on April 3 (phone), with Bill Johnston (NRC) on April 20 (phone) and with Paul Bryant and Klotz of Combuston Engineering Windsor, Connecticut, on April 10 and 11, 1979 (phone).

These discussions covered my understanding of basic and germane aspects of the radiation chemistry of water and aqueous solutions, as outlined below.

Water is decomposed by nuclear radiations to form radicals (e^-_{aq}, H, OH, and possible H_3O) in heterogeneous distribution, as initial products. These interact as they diffuse into homogeneous distribution,* and the products which appear in homogeneous distribution are, primarily,

$$e^-_{aq}, H, OH, H_2O_2, H_2, H_3O^+.$$

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These are called "primary products." The number of a species which appears as a primary product per 100 ev of absorbed radiation energy (yield) is commonly symbolized by $g(H_2)$, $g(OH)$, etc.

---

*Homogeneous distribution from the standpoint of reaction kinetics.

The following oxidation-reduction relationship prevails among these products.

$$2g(H_2) + g(e^-_{aq}) + g(H) = g(OH) + 2g(H_2O_2).$$

The value for $g(H_2)$ for beta-gamma in pure H_2O is 0.44. That for the radicals is much greater, e.g., $g(e^-_{aq} + H) = 2.86$.

The primary products interact further to form products which include O_2 and H_2O. They may also react with other solutes present in solution. The overall results of the primary decomposition, with respect to the net decomposition of water (expressed as G-values) depend strongly upon these various reactions. This can be illustrated by noting that a low net decomposition occurs in pure water during exposure to β-γ or reactor radiations when the radiolysis products remain in solution (as in the ORR and HFIR). Under these conditions, the molecular products accumulate and reach concentration levels ($\sim10^{-4}$ m in the ORR and HFIR) at which the rates of formation and of removal by reaction with radicals are equal. The addition of H_2 promotes the reactions which lead to recombination of water, and the addition of both H_2 and O_2 in excess of the steady state concentrations leads to combination of the dissolved gases. The G-values for this water formation reaction (as studied and reported by Hochanadel) depend on temperature and the indicated values for $G(H_2O)$ were 3.0 at 100°C and 5.5 at 150°C.

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If the radicals are removed from solution by reaction with radical-scavanger solutes* (e.g., Cu^{+2} or Br^-), the molecular products remain in solution and they accumulate until they reach concentrations at which they can successfully compete with the scavanger ions for the radicals. Depending upon the properties of the particular scavanger

*Radical scavanger ions are those which, in effect, catalyze the removal of the radicals by undergoing oxidation with the OH radical followed by reduction with the reducing radicals, or vice versa.

ions under consideration the molecular product concentration may reach several hundred times that of the scavanger before recombination of water becomes significant.

The reactor coolant at TMI was not pure water, and I mentioned and discussed the following reservations regarding the prediction of radiolytic characteristics of the TMI coolant.

1. Effects of the borates in the coolant - Work of Hart et al., reported in the 1st Geneva Conference indicated that the borates have little effect on the radiolysis of water. Also, H. E. Zittel of ORNL has studied radiolysis in borated solutions, and I asked (by phone) his opinion. He indicated that the borates would probably have negligible effect on the radiolysis.

2. Effect of pH of the coolant - Some uncertainty exists regarding the effects of high pH on radiolysis of water. However, Zabrowsky stated that the borated solutions in TMI would have a pH near 7 at the prevailing temperature (138°C). Assuming this was correct, it seemed likely that no significant pH effects would occur.

3. Effects of dissolved U and/or fission products - Zabrowsky said that available information indicated only a few ppm of these. Assuming total concentrations of $\sim 2 \times 10^{-5}$ $\underline{m}$ of ions with high radical scavanger action, the steady state concentration of radiolytic H_2 in solution would correspond to a gas pressure of ≤ 100 psi at 138°C. This was well below the prevailing H_2-bubble pressure of ~ 1000 psi at 138°C and it then seemed unlikely that any net radiolytic decomposition would take place.

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I also mentioned that radiolytic recombination of H_2 and O_2 in the gas phase would probably take place with $G(H_2O) \gtrsim 3$, based on total radiation energy absorption in the gas phase.

Estimated costs - About one day of my time not counting my time on Sunday, April 1, 1979.

Howard McLain of the Energy Division and Paul Haas of Chem Tech assisted us on Monday in our continued assessment of hydrogen explosion limits. We concurred with a NASA specialist who was advising Zebrowsky at the site.

NSIC has continued to assist various clients on safety questions raised by TMI, however, the crisis atmosphere ended with the disappearance of the hydrogen bubble on April 2.

JRB:pc

cc: M. M. Cardwell
R. B. Gallaher
P. A. Haas
G. Jenks
G. T. Mays
H. A. McLain
R. L. Scott

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

April 30, 1979

To: Wm. B. Cottrell

From: J. R. Buchanan JRB

Subject: Technical Assistance on Three Mile Island Accident

On Saturday March 31, I received a phone call from Floyd Culler of the Electric Power Research Institute (EPRI) requesting that NSIC provide technical support information to Milt Levenson and Ed Zebrowsky of the EPRI staff who were on their way to Harrisburg, Pa. The Three Mile Island (TMI) Unit 2 accident that occurred on March 28 was by then recognized to be extremely serious with most of the attention being focused on a hydrogen bubble that had formed in the reactor pressure vessel.

Herman Postma phoned later on Saturday to confirm Culler's request and asked that we assist EPRI. I arranged for technical staff coverage for the Center for Sunday and for RECON to be available at that time. Ray Scott and Richard Gallaher assisted me on the day shift and Gary Mays worked with me on the evening shift. Glen Jenks of the Chem Tech Division came in for several hours to help us calculate "G" factors for hydrogen recombination under assumed TMI conditions. Marie Cardwell opened the Document Room of the Technical Library since we needed access to a number of technical reports. She kept it open for about 10 hours.

Most of our attention on Sunday was devoted to estimating radiation induced recombination rates, studying the TMI high pressure system in the SAR, and collecting data on hydrogen explosion limits. We conferred with Ed Zebrowsky at the site several times by phone. We recommended against adding any internal recombination agent since it appeared that the "G" factor at TMI could be as high as 3 or better and such an agent could upset the favorable hydrogen removal rate. We also recommended that dry surfaces in the pressure vessel be avoided and that the pressurized spray be continued. Zebrowsky found the recombination data particularly helpful in the NRC planning meetings Sunday evening. Jenks' counsel on water chemistry also added an ingredient that was sorely needed.

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UCN-430
(3 3-61)

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

MAY 7 1979

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NSIC has continued to assist various clients on safety questions raised by TMI, however, the crisis atmosphere ended with the disappearance of the hydrogen bubble on April 2.

JRB:pc

cc: M. M. Cardwell
R. B. Gallaher
P. A. Haas
G. Jenks
G. T. Mays
H. A. McLain
R. L. Scott

POOR ORIGINAL

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Three Mile Island - 2
Technical Support

On-Call Technical Assistance

L. C. Oakes, et. al.

Problem:

Provide on-call expert technical assistance to diagnose problems and lend advice as information becomes available from the TMI site. Principal areas of concern are instrumentation and control, thermo-hydraulics, noise diagnostics, reactor systems and neutronics.

Description of Work Performed:

Calculations, investigations, and experiments with subsequent recommendations were made in the following areas:

1. On-site team performing noise diagnostics and other measurements.
2. Study to predict or explain the pressure pulse response observed on the primary system pressure sensors.
3. Calculations to examine adequacy of natural convection for core cooling.
4. Evaluation of available data to determine if boiling is occuring in the core.
5. Search for substitute methods for measuring pressurizer level, should all existing sensors fail.
6. Evaluate failure modes and predicted radiation life of pressurizer level sensors.
7. Evaluate failure modes of core thermocouples to help establish validity of core temperature measurements.
8. Confirm and evaluate sequence of events to predict probable extent of damage.

9. Evaluate feasability of using a Resistance-Temperature Device (RTD) as a level probe in the pressurizer.
10. Examine failure modes of Rhodium self-powered neutron detectors to explain anomalous readings.
11. Evaluate dynamic hydraulic conditions at the pressurizer differential pressure cells to find explanation for unexpected behavior.
12. Calculate the potential for cooling the core by flooding the outside of the reactor vessel if natural convection cannot be established.
13. Evaluate or propose ways in which noise analysis might be used to infer water level in the reactor vessel.
14. Propose methods for deriving a direct readout of subcooling for display to operators.
15. Explore what can be learned about thermal-hydraulics from in-core neutron detectors. Eg. water level, boiling, pressure, etc.
16. Evaluate implications and probable causes of high count rates on startup neutron detectors. Determine if related to reactivity in the core.
17. Participate in boiling tests at the Blowdown Heat Transfer experiment to acquire baseline data for determination of boiling at TMI.
18. Provide on-site assistance in rigging substitute pressurizer level measurements with RTD or pressurizer heaters.

Work Requested By:

NRC.

Results Reported To:

NRC at TMI and Bethesda; summary reports have been prepared for individual tasks.

Work Performed By:

J. L. Anderson
S. J. Ball
R. S. Booth
R. M. Carroll
F. H. Clark
M. H. Fontana
D. N. Fry
M. B. Herskovitz
R. C. Kryter
J. T. Mihalczo
J. E. Mott (TEC)
L. C. Oakes
P. J. Otaduy
J. R. Penland (SAI)
J. C. Robinson (TEC)
R. L. Shepard
W. H. Sides, Jr.
C. M. Smith
R. S. Stone
G. L. Zigler (SAI)
R. E. Hedrick (SAI)

Estimated Cost:

?

Three Mile Island - 2
Technical Support

High Count Rate on BF_3 Startup Detectors

J. L. Anderson, et. al.

Problem:

Ex-vessel startup detectors (BF_3 Proportional Counters) are indicating about 6 times the normal shutdown counting rate. There is concern that if this counting rate is real, the core may be considerably more reactive than it should be in the present highly borated, fully shutdown condition. We need to determine the reasons for the high readings.

Description of Work Performed:

A considerable effort was required to establish the validity of the information received from TMI; to examine the geometry and characteristics of the installation as designed and as it pertains to reactivity; to postulate and evaluate mechanism which could explain the anomalous indications. We became convinced that 1) The detectors were undamaged, functional, and probably accurate; 2) If the high readings represented core multiplication, then the reactor was dangerously near critical and further cooldown would result in criticality and higher core power. The detector indications were followed carefully to determine trends and to attempt to identify the source of the high readings. This turned out to be difficult, because there was considerable confusion about how the data from these detectors was recorded by the reactor operators. This confusion persists, because of an apparent lack of understanding of the detector and amplifier characteristics by the operating crew, and their procedures are of questionable validity. In spite of this uncertainty, it was determined that the counting rate displayed by the startup channels was following anticipated gamma decay curves. This implies that the neutrons detected were generated by gamma-neutron reactions in water and other materials and were not necessarily related to core reactivity. Further, changes which should have affected core reactivity such as boron concentration and temperature did not have a corresponding effect on the counting rate. This adds further weight to the idea that the neutrons seen by the detectors are not core-coupled.

Conclusions and Results:

One candidate to explain higher core reactivity is the redistribution of burnable poisons from the reactor core. These burnable poisons are made of Boron Carbide clad in Zircalloy. Some of the Zircalloy is undoubtedly oxidized and the Boron Carbide may be redistributed. However, there is no positive indication that the core reactivity is higher than normal. Two more likely explanations are; 1) The neutron source containers are oxidized and the neutron sources are distributed throughout the reactor vessel, perhaps in closer proximity to the startup detectors; 2) Because of fuel cladding failure, the distributed gamma activity level is considerably higher than normal and may be located much closer to the detectors yielding more localized gamma-neutron reactions and higher counting rates. Significant gamma or neutron sources external to the reactor vessel in the vicinity of the detectors are unlikely. It is concluded that a significant core reactivity problem does not exist.

Work Requested By:

NRC-Bethesda, NRC-TMI (Beltracchi, Stello, Ackerman).

Results Reported To:

NRC-Bethesda (Buhl); NRC-TMI (Ackerman). April 20, 1979.

Work Performed By:

J. T. Mihalczo, J. L. Anderson, M. A. Schultz (TMI-IAG).

Estimated Costs:

3 Mandays; $1,000.

Three Mile Island - 2
Technical Support

In-Core Neutron Detectors

R. S. Booth

Problem:

High readings are observed on a few of the in-core self-powered Rhodium neutron detectors. Examine failure modes to determine if the high readings are indicative of detector failure or if the readings could be true indications.

Description of Work Performed:

Several experts in the field of in-core self-powered neutron detectors were contacted to survey their experience with failures which might give readings similar to those experienced at TMI. TMI detectors were manufactured by a subsidiary of B&W so that experience of others such as Reuter-Stokes is not directly applicable. Most helpful were Ray Watson and Colin Allen of Atomic Energy of Canada at Chalk River. In their opinion, the high readings are most likely a result of moisture in the detectors or in the cables. Significant differential temperatures on the order of 100 degrees F across portions of moisture filled Magnesium Oxide insulated cable can generate electrical signals of the magnitudes measured at TMI. Chalk River has developed a testing procedure for detecting excessive cable leakage. The procedure involves applying various voltages to the normally self-powered detectors and cables and measuring the resultant currents. In good detectors with dry insulation the resistance will be very high and linear with voltage. Moisture tends to generate spurious emf's and shows up as non-linear current-voltage relationships.

Conclusions and Results:

High reading detectors in the TMI core probably have moisture in their cables as a result of being exposed to the high temperature oxidizing atmosphere which was known to exist. A recommendation was made to test the detectors and cables at TMI using the Chalk River technique.

Work Requested By:

NRC-TMI (Stello, Ackerman)

Results Reported To:

NRC - Ackerman April 6, 1979.

POOR ORIGINAL

Work Performed By:

R. S. Booth, J. L. Anderson, R. K. Abele.

Estimated Cost:

1 Manday; $330.

Three Mile Island - 2
Technical Support

Natural Circulation

R. S. Stone

Problem:

Will the core be cooled adequately by natural circulation should all coolant circulation pumps become inoperable?

Description of Work Performed:

The objective was to extrapolate from fuel temperatures then being measured and so predict the temperatures to be expected if the core cooling was limited to natural convection. Working from dimensions in the TMI PSAR, the net buoyant driving force available as a result of heating in the core and cooling in the heat exchanger were calculated. This was then equated to the frictional resistance to derive the flow rate and to determine the differential temperature necessary to close the loop. At TMI conditions a day and a half after shutdown, assuming the reactor core power then to be 10 Mw, the anticipated differential temperature across the core was calculated to be 54.3°F.

This is a manageable temperature rise. With an intact core and an operational heat exchanger there should be no problem running on natural convection cooling at the 10 Mw level. The concern is that the core may not be intact. At the time of the analysis, thermocouples in one corner were indicating temperatures well above those of the rest of the core. The fear of course was that the high temperature region had blockage and restricted flow.

Additional calculations were then made based on an assumption of quarter-core blockage. These calculations indicated the core temperature rise with natural circulation at 10 Mw could be as high as 331°F. This would produce boiling in the damaged quadrant for any inlet temperature.

Conclusions and Results:

Since the region showing elevated outlet temperature seemed to shift with changes in the choice of coolant pump, the assumption that part of the core is obstructed has been less defensible. If the cause is simply inertial channeling, there should be no problem under natural convection. Even if part of the core were obstructed to the extent calculated, the differential temperature will be proportional to power level. At a month after shutdown, if we assume 2 Mw afterheat, the temperature rise would be 66.2°F, a level which could be tolerated. There should be some concern over accuracy of temperature rise in the hot spot region. As better measurements of temperature and flow become available the above numbers would have to be modified; the methodology should be OK.

Work Requested By:

NRC Bethesda (Hanauer, Buhl).

Results Reported To:

Same. Approximately April 8.

Work Performed By:

R. S. Stone, Bob Hedrick (SAI), Ted Mott (TEC).

Estimated Cost:

8 Mandays, $2,640.

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 11, 1979

To: F. R. Mynatt

From: R. L. Clark RLC

Subject: Three Mile Island Report

The following information is submitted in response to the directive memo from your office dated May 2, 1979.

Statement of Problems:

The initial group (including myself) from ORO-ORNL arrived at Harrisburg's Capital City Airport by charter flight from Knoxville shortly after 9:00 PM on March 30, 1979.

My work assignment(s) while at Three Mile Island could be divided into three distinct phases:

1. On March 31, 1979, I was assigned to provide the health physics monitoring services necessary for proper protection of the robot maniuplator crew during their operations on the island (neither H.P. personnel nor equipment had arrived with the crew). This assignment continued for the duration of stay of the manipulator operating team and consumed by far the major fraction of my time.

   Once the manipulator team had established that all system components were completely operable, we were engaged in many hours of operating practice and demonstrations for various levels of Met Ed and NRC supervisors and engineers so that they could best decide how to apply this capability to their immediate needs. This decision indicated that use of the manipulator for obtaining sample(s) of the primary coolant system would be most beneficial.

   Met Ed engineers and the crew then went to work on development and procurement of special equipment items and procedures necessary to perform that function. Following this, more practices and demonstrations to gain and show competence ensued.

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By this time, the process instrumentation had begun to fail and it was decided that the water line to be used for sampling might become too critical in the need for liquid level information to permit the use of it for sampling purposes (because the risk of some equipment malfunction in the sampling operation had to be acknowledged).

Equipment development, procedural changes, and operational practice continued to the end of the week with the entire crew involved.

No high radiation levels were encountered and exposures to the crew were of no consequence. Some low levels of contamination were discovered on the shoes of one crew member and the hands of two of them while the manipulator was deployed in the area of the Chemistry Laboratory where we expected to collect the water sample. The contamination was readily removed by simple techniques and was not of any great importance.

2. For most of one day (April 3, I think), I was assigned to participate in the early discussions of a planning group constituted by (or at the request of) Mr. Herman Dieckamp and Mr. Ron Williams. This function was taken over on the following day by the chemical engineering support group from ORNL (Brooksbank, Yarbro, et.al.).

3. Because of my on-island assignment (with the robot crew) and my now established contacts with Met Ed engineering and operating personnel, another assignment was to serve as liaison to Brooksbank's group to obtain and transmit information from the island to them. It appeared that some benefits did accrue from this more direct communication channel.

Conclusions or Results:

See last paragraphs in 1, 2, 3 above.

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Work Requested By:

DOE-ORO (B. J. Davis was on-scene leader for all H.P. personnel from ORNL.

Results Reported To:

B. J. Davis, DOE-ORO. Shared motel room with Mr. Davis and kept him informed through contacts there (and by phone) on day-to-day basis.

Work Performed By:

R. W. Frazier, Robot Crew Supervisor
T. E. Copeland, Crew Member
W. L. Pankratz, Crew Member
R. Turner, Crew Member
All above are UCC-Y-12 employees
R. L. Clark

Costs

Labor and travel expenses only ($ unknown).

It is realized that the foregoing is more descriptive narrative than numerically specific, but hoped that this will be suitable to your needs. If not, please contact me.

RLC:ac

1915 275

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

TO: F. R. Mynatt

FROM: M. L. Conner, 4-6703 MLC

SUBJECT: A Brief Description of Work I Performed at Three Mile Island

I was sent to Three Mile Island with four other radiation surveyors and one DOE representative to perform radiation surveys of the environment. Members of our team included R. L. Clark, A. C. Butler, W. M. Johnson, W. D. Carden, and B. J. Davis.

The actual work performed was a 24-hr, continuous survey of the environment at Three Mile Island. This task was accomplished by taking direct radiation readings from portable instruments and air samples from a portable air sampler. The results of our survey were reported to the NRC through DOE. I observed no high radiation readings or abnormal air samples.

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UCN-430
(3 3-81)

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

TO: F. R. Mynatt

FROM: J. S. Eldridge (4-4924) JSE

SUBJECT: Assistance to Three Mile Island (TMI)

Problems: I was requested to provide technical assistance and specialized equipment for environmental surveillance. This equipment consisted of a portable gamma-ray spectrometer and a portable dual-channel spectrometer. In addition, I was asked to transport a portable hydrogen analyzer for possible use.

Work Performed: Work performed included taking the hydrogen analyzer into the control room at 12:30 AM April 1, 1979. At that time, I turned the instrument over to Babcock and Wilcox personnel after a brief operational discussion. Other work involved assessing the need for my instrumentation at the TMI observation control station and at the environmental surveillance operations at the Capital City airport. This latter effort involved the afternoons of March 31 and April 1.

Conclusions: By the end of the second day at TMI, it was obvious that releases of fallout fission products were minimal and additional equipment was not needed.

Work Requested By: J. A. Auxier, 30 March 1979.
(In response to request by B. J. Davis, DOE).

Results Reported To: J. A. Auxier and D. B. Trauger, April 3, 1979.

Work Performed By: J. S. Eldridge and S. A. Hamley.

Estimated Costs: JSE only: Travel $410.11 and 3 days' labor.

JSE:gal

cc: J. A. Auxier
W. D. Chults
J. R. Stokely

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UCN-430

MAY 16 19

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 14, 1979

To: F. R. Mynatt

From: B. A. Powers BAP

Subject: ORNL Assistance at Three Mile Island

As a member of the Industrial Safety and Applied Health Physics survey section, I was asked to perform environmental surveys, arranged by the Department of Energy, to determine conditions of the environment surrounding Three Mile Island following the March 28 incident.

J. E. Smith and I arrived at Harrisburg on April 7, 1979, to replace the first ORNL survey team working for DOE. We reported to Ted Schoenburg at DOE headquarters on April 8. Our work included taking direct radiation measurements in the air, taking soil, water (standing and rain) and vegetation samples, and chasing the plume for determination of location and measurement. We also plotted our sampling locations on a map of the area, so if results indicated, further sampling could be done in the same area.

Our samples were gamma-spectrum analyzed at the DOE control point and then sent to Brookhaven Laboratory for further radiochemical analysis. The results of counting room analyses were reported to NRC by our sampling and counting room coordinator, Boyd Schultz. Spectrum analysis yielded only two samples with ^{131}I, right at minimum detectable activity ($\sim 2 \times 10^{-12}$ µCi/ml) for the detector used, which is to say, background levels.

Costs for room and meals for the six-day stay totaled about $270 for each of us. A daily log of activities and data sheets were kept and is available upon request.

BAP:ac

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MAY 15 1979

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 10, 1979

To: F. R. Mynatt

From: W. D. Carden W.D.C.

Subject: Three Mile Island Report

The following report is submitted in accord with your request of May 2, 1979.

Statement of Problem:

Participate in environmental monitoring program in area near Three Mile Island Unit II.

Description of Work Performed:

Observed and reported background radiation levels and collected samples of air, water, soil, and vegetation for laboratory analysis by others.

Conclusions or Results:

From March 31 to April 7, 1979, I served as part of a two-man survey team assigned to collect information on radiation levels and samples of environmental materials at a number ($\sim$ 15) of designated locations along Pennsylvania State Route 441 which lies along the east shore of the Susquehanna River.

The background radiation levels were measured with Victoreen Thyac Survey Meters (G-M), with notations made in each instance of the observed gamma reading and the beta-gamma reading. Several notations at each location were made during each 12-hour work shift.

In addition to these time-programmed surveys, we were frequently directed to various locations in efforts to observe changes in background levels due to passage of the plume from the reactor site. Air samples were collected (on charcoal cartridges) as directed by the program supervisors.

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F. R. Mynatt

May 10, 1979

From April 13 to 17, 1979, I served again as part of a two-man team engaged in environmental monitoring and sampling. The headquarters for this phase of the operation was located at the Capital City Airport and the program was under direction of DOE personnel. Background measurements at the designated survey points were continued with the Thyac Survey Meter and augmented by readings from a mica-window survey meter (having somewhat greater sensitivity to beta radiation levels).

In addition to the direct radiation measurements, our efforts included collection of samples of air, rainwater, standing water, soil, and vegetative materials. Each sample was identified as to location and time of collection and submitted for analysis by others.

Work Requested By:

The Department of Energy - Oak Ridge Operations Office.

Results Reported To:

Program Supervisors

March 31 - April 7
B. J. Davis, DOE - Oak Ridge
E. J. Jascewsky, DOE - Chicago

April 13 - 17
J. D. Sage, DOE - Pittsburgh Naval Reactor

Work Performed By:

W. D. Carden
W. M. Johnson

Estimated Costs:

Salaries and travel expenses ($ unknown).

WDC:ac

1915 280

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 10, 1979

To: F. R. Mynatt

From: A. C. Butler ACB

Subject: Assistance in Response to the Three Mile Island Accident

Statement of Problem

Radioactivity leaking from the reactor area.

Description of Work Performed

Using a THYAC GM-Survey Meter, located and followed the plume from the reactor area on the east side of the river from the reactor site. Recorded open-window and closed-window radiation readings at predetermined points along a six mile long survey route. This route extended approximately three miles north and approximately three miles south of the reactor area. Air samples were taken at the highest radiation reading locations using a portable gasoline powered AC generator to power the charcoal filter air sampler.

Conclusions or Results

Readings were obtained up to approximately 20 mR/hr beta-gamma (open-window) and 2.5 mR/hr gamma (closed-window). The radiation readings from the radioactive gases varied with time due to changes in wind direction and velocity. The results on the air samples turned in to NRC were not made available to us.

Work Requested by

The work was requested by the Department of Energy (DOE), Oak Ridge Operations Office on March 30, 1979.

Results Reported to

At the reactor work area our team was assigned and reported all results to the Nuclear Regulatory Commission (NRC) during the period from March 31, 1979 until the afternoon of April 7, 1979.

Work Performed by

Team: A. C. Butler
M. L. Conner

Estimated Cost

1915 281

Unknown

ACB:ac

THREE MILE ISLAND RESPONSE
MARCH 28 - MAY 18, 1979
COST ESTIMAGE FOR UCCND

1	2	3	4	5	6	7	8	9
Category	Person Days of Effort	Travel	Salary	Materials, Supplies, and Misc.	Overhead	Total	Incremental Salary and Overhead	Total Incremental Costs, Sum of 3, 5, and 8
A. Radiation Monitoring								
1. On-Site Participation	78	$ 8,926	$ 14,458	$ 0	$ 7,016	$ 30,400	$7,994	$ 16,920
2. Direct and Dedicated Support	10	0	1,846	0	554	2,400	0	0
3. Incidental Support				0				
SUBTOTAL A	88	$ 8,926	$ 16,304	$ 0	$ 7,570	$ 32,800	$7,994	$ 16,920
B. Technical Support								
1. On-Site Participation	127	$15,771	$ 28,031	$ 900	$15,237	$ 59,939	0	$ 34,897
2. Direct and Dedicated Support	267	0	75,042	59,487*	22,512	157,041	0	59,487
3. Incidental Support**	16	1,437	2,806	0	1,272	5,515	0	1,437
SUBTOTAL B	410	$17,208	$105,879	$ 60,387	$39,021	$222,495	0	$ 95,821
TOTAL A & B	498	$26,134	$122,183	$ 60,387	$46,591	[illegible],295	$7,994	$112,741

*$54,500 is computer costs
**Participation in emergency hearing, Lynchburg, Virginia

1915 282

THREE MILE ISLAND RESPONSE
MARCH 28 - MAY 18, 1979
COST ESTIMATE FOR DOE/ORO

1 Category	2 Person Days of Effort	3 Travel	4 Salary	5 Materials, Supplies, and Misc.	6 Overhead	7 Total	8 Incremental Salary and Overhead	9 Total Incremental Costs, Sum of 3, 5, and 8
A. Radiation Monitoring								
1. On-Site Participation	8	$ 400	$ 580	$ 0	$116	$1,096	$ 0	$ 400
2. Direct and Dedicated Support	7	0	580	0	116	696	0	0
SUBTOTAL A	15	$ 400	$1,160	$ 0	$232	$1,792	$ 0	$ 400
B. Technical Support*								
1. On-Site Participation	7	$ 261	$ 563	$ 780	$113	$1,717	$ 321	$1,362
2. Direct and Dedicated Support	7	0	580	0	116	696	0	0
SUBTOTAL B	14	$ 261	$1,143	$ 780	$229	$2,413	$ 321	$1,362
C. Public Information								
On-Site Participation	5	$ 461	$ 711	$ 0	$142	$1,314	$ 0	$ 461
D. Photograph								
On-Site Participation**	3	-	-	$ 806	-	806	-	806
TOTAL	37	$1,122	$3,014	$ 1,586	$603	$6,325	$ 321	$ 3,029

*Drivers for transporting mobile manipulator

**Consultant

1915 283

9

Department of Energy
Savannah River Operations Office
P.O. Box A
Aiken, South Carolina 29801

MAY 25 1979

Robert L. Ferguson, Program Director, Nuclear Energy Programs,
Assistant Secretary Energy Technology, HQ

INFORMATION REQUESTED BY THE PRESIDENT'S COMMISSION ON THREE MILE ISLAND (YOUR TELETYPE 5/15/79)

SR input to the subject teletype is enclosed for your information and use. Questions regarding this information may be directed to C. T. Marsh, Director, Organization and Personnel Division, FTS 239-2855.

for N. Stetson
Manager

PT:RCW:npb

Enclosure

1915 284

2

DOE-SR Emergency Assistance to Three Mile Island

DOE-SR emergency assistance to Three Mile Island was provided by the Savannah River Laboratory and the Savannah River Plant of E. I. du Pont de Nemours and Company, the SR prime operating contractor. The scope of this assistance is outlined as follows:

- Meteorological Assistance

  1. Meteorological assistance was provided at the request of Lawrence Livermore Laboratory at the DOE Command Post, Capital City Airport, Pennsylvania.

  2. M. M. Pendergast, Environmental Transport Division, Savannah River Laboratory, provided assistance during the period 4/10/79 through 4/15/79.

     J. F. Schubert, Environmental Transport Division, Savannah River Laboratory, provided assistance during the period 4/15/79 through 4/18/79.

  3. Pendergast and Schubert provided meteorological expertise at the DOE Command Post, determining wind speed, direction, and plume dispersion trajectories. They also interpreted data calculated at Lawrence Livermore Laboratory using the Atmospheric Response Advisory Capability (ARAC) System.

  4. Results were communicated to personnel at the Lawrence Livermore Laboratory and at the DOE Command Post, Capital City Airport.

- Analytical Assistance

  1. A sample of primary coolant water from Three Mile Island, Unit No. 2 Reactor, was analyzed.

  2. Analytical work was performed on 4/11/79 by the following:

Analytical Chemistry Division Savannah River Laboratory	Laboratories Department Savannah River Plant
E. J. Lukosius D. R. Johnson	L. V. Ruczko A. Gibbs

1915 285

(3)

3. The analytical results are summarized in the attached letter to Mr. Brian Grimes.

4. Information was transmitted to B. Grimes of NRC via the attached letter. The information, in whole or in part, was transmitted by telephone to the following individuals in NRC:

   Tom Teflord
   Joe Hegman
   Bernie Weiss
   Mike Slobedian
   John Collins

- Health Protection Assistance

1. Supplementary supplied air respiratory equipment was provided to Three Mile Island, including:

   - Air compressor (SME 52-77)
   - Two CO monitor carts (L-11848 & L-13241)
   - 4000' of 3/8" breathing air hose
   - 140 plastic suit jackets with air distribution systems

   Two individuals at TMI were trained in use of the equipment. Procedures were provided on operation of the compressor, CO monitor carts, and on use of the plastic suit jackets.

2. D. J. Ratchford, D. B. Zippler, and L. Jolly, Health Protection Department, Savannah River Plant, procured the equipment and arranged for its transport to Three Mile Island on April 17 and 18, 1979.

   L. Jolly, Health Protection Department, Savannah River Plant, visited Three Mile Island on April 19, 20, and 21, to instruct personnel in use of respiratory equipment.

3. Services provided are described above.

4. Assistance was provided at the request of and coordinated through B. C. Rusche, Waste Management Group Leader, Three Mile Island.

5. The compressor and two CO monitor carts will be used at Three Mile Island for a period of about 6 weeks and will then be returned to SRP. Other items will be consumed in use.

(4)

o <u>General Assistance</u>

1. Various information was requested by Three Mile Island personnel in telephone conversations with SRP personnel. No actual work was performed.

2. On April 6 and 7, 1979, M. Siano and T. Johnson, TMI, telephoned H. H. McGuire, Works Engineering, SRP, to request information regarding equipment available to fabricate a backup residual heat removal system.

3. On April 4, 1979, L. Conway, Westinghouse, telephoned E. C. Bertsche, Works Technical, SRP, to obtain information on portable deionizers and filters available at SRP.

4. On April 6, 1979, R. Vollmer, NRC, telephoned E. O. Kiger, Production, SRP, to ask about SRP filter compartments with activated carbon.

5. On April 14, 1979, L. Conway, Westinghouse, called D. A. Ward, Works Technical, SRP, and D. H. Thomas, Production, SRP, to consult about use of eductors to collect leakage from backup residual heat removal system.

o <u>Continuing Assistance</u>

Please note Item 5. under <u>Health Protection Assistance</u> above. DOE-SR assistance to Three Mile <u>Island is presently limited</u> to equipment.

E. I. du Pont de Nemours & Company
ATOMIC ENERGY DIVISION
Savannah River Laboratory
Aiken, South Carolina 29801
(TWX 810-771-2670, TEL 803-725-6211 WU AUGUSTA GA)

DPST-79-340

(5)

May 7, 1979

Mr. Brian Grimes
Division of Operating Reactors
Mail Stop 340
Nuclear Regulatory Commission
1717 "H" Street, N.W.
Washington, D.C. 20555

Dear Mr. Grimes:

Results of analyses requested by NRC on a sample of primary coolant water from the Three Mile Island, Unit No. 2, reactor are summarized in Table I. The sample was received at the Savannah River Laboratory at 5:30 A.M. on April 11. The requested analyses were performed during that day. Results were transmitted by telephone to the NRC Operations Center in Washington, D.C., at 8:30 P.M. that night. Since the initial results were submitted, some of the analyses have been re-evaluated and corrections made, where appropriate. The results reported in the table include all corrected values. Corrected results were discussed by telephone with the Operations Center on April 23.

The presence of particles in the solution, as evidenced by turbidity in the as-received sample, as well as a high solution pH (~ 8) must be considered in interpreting the analytical results. At the high pH, the U, Pu and Zr dissolved by high temperature water reactions would be expected to form colloids as the water cooled. The colloidal particulates could be absorbed on the surfaces of the vessels, pipes, etc., thus selectively removing some or all of those elements from the water sample.

Uranium was determined on the as-received sample and on an acidified aliquot. The particles that caused the turbidity were dissolved in the acidified sample. Uranium concentration was equivalent in the two samples, indicating that insoluble uranium was not present on any particulates contained in the sample. The boron results are suspect for two reasons: (1) some boron would be expected to be absorbed on the particles at pH-8 (causing a low boron concentration in the solution), and (2) the high level of radioactivity required large dilutions before the analysis could be made (causing a decrease in the accuracy and precision of the analysis). We recommend that the higher boron values reported by the other laboratories be used in any calculations.

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The gamma pulse height analysis was done at 10:00 A.M. on April 11. The reported results have not been corrected for decay.

Savannah River Laboratory will be pleased to cooperate with the NRC to analyze any additional samples from the Three Mile Island reactor which are necessary to complete shutdown and decontamination of the reactor facility.

Very truly yours,

Robert F. Overman

R. F. Overman, Senior Chemist
Analytical Chemistry Division

RFO:rvd
Attach.

7

TABLE I

Analysis	Results	Method
Total Boron	3100	Carbon Rod Atomic Absorption
Zirconium	< 50 ppm	Atomic Absorption
pH	7.7	pH Meter
Gross Alpha	$< 2 \times 10^3$ d/m/ml	P-10 Gas Flow Detector
Uranium	< 0.001 ppm	Extraction of Uranium and Fluorophotometric Determination
Transuranics	None detected	Alpha Counting
Gross Beta	9×10^9 counts/m/ml	Uncalibrated Gas Flow, Windowless Detector
Strontium-89, -90	1×10^9 d/m/ml	Chemical Separation (Fuming Nitric Precipitation) Liquid Scintillation Counter
I-131	1.60×10^{10} d/m/ml	Gamma analysis with Ge(Li) Detector and Pulse Height Analyzer. Counted at 10:00 A.M. 4/11/79. No decay corrections were made.
Rb-86, -88	Not detectable	
Ce-141	2.28×10^8	
Cs-137	7.11×10^8	
Cs-134	1.68×10^8	
Cs-136	2.74×10^8	
Ba-136m	2.01×10^8	
Ba-140	5.7×10^8	
La-140	3.0×10^8	
Mo-99	2.77×10^8	
Tc-99m	1.5×10^8	
Ru-Rh-106	Not detectable	
Zr-Nb-95	Not detectable	

1915 290

(9)

Department of Energy
Savannah River Operations Office
P.O. Box A
Aiken, South Carolina 29801

RECEIVED

1979 MAY 1 AM 9 09

APR 25 1979

1. ~~BJS~~ RF
2. ~~Crawford~~ JWT
3. Feinroth

R. L. Ferguson, Program Director for Nuclear Energy, HQ
F. C. Gilbert, Acting Director, Office of Nuclear Materials, Defense Program, HQ
H. Feinroth, Chief, Nuclear Reactor Evaluation Branch, Division of Nuclear Power Development, HQ

ASSISTANCE FOR THREE MILE ISLAND

On April 17, 1979, my staff received a call from E. S. Beckjord requesting that we have Du Pont get in touch with B. C. Rusche at Three Mile Island because he was in urgent need of help. Rusche advised Du Pont that Metropolitan Edison needed breathing air supplies and radiation survey equipment in connection with the accident. The equipment (list attached) was shipped on April 18, and arrived at Three Mile Island on April 19. In addition, Du Pont is sending a consultant to show them how to set up the equipment. Metropolitan Edison will be billed for all associated costs.

N. Stetson
N. Stetson
Manager

D:LL:dm

Enclosure

1915 291

EQUIPMENT FOR THREE MILE ISLAND

ITEM	QUANTITY	BILLING BASIS
1. Sullare Mobile Air Compressor, #SME 5277	1	Lease
2. Carbon Monoxide Monitoring Cart	2	Sale
3. Plastic Suits Tops, Modified	150	Sale
4. Plastic Suit, air distribution system	150	Sale
5. Plastic Suit Plenum	150	Sale
6. 100' lengths of 3/8" breathing air hose	40	Sale
7. Carbon Monoxide Drager tube	50	Sale
8. Radiation Survey Suitcase	1	Sale
1 RO-2 Survey Meter		
1 H.P. 210 Survey Meter		
1 Thyac Survey Meter		
1 Self Reader Pencil		
1 Box Smears		

*Includes packing, handling, transportation and set-up expenses.

1915 292

Department of Energy
Idaho Operations Office
550 Second Street
Idaho Falls, Idaho 83401

MAY 29 1979

(4)

TO: Robert L. Ferguson, Director
Nuclear Energy Programs
U.S. Department of Energy, HQ

FROM: Charles E. Williams
Manager

C E Williams

SUBJECT: INFORMATION REQUESTED BY THE PRESIDENT'S COMMISSION ON THREE MILE ISLAND

References:

1. TWX R 151741Z, Ferguson to C.E. Williams, et al, dated May 1979.

2. EG&G Letter Kiehn-179-79, Kiehn to Col. R.E. Lounsbury, EOC Germantown, dated May 9, 1979, subject: DOE Participants in Operation Ivory Purpose.

The attached information is provided in response to your TWX, Reference 1. EG&G Idaho, Inc. has provided and is continuing to provide assistance to the NRC on the Three Mile Island accident, as reported in Reference 2. Those activities which were funded through NRC programs are noted in the attachment. The amount of DOE involvement in ongoing activities is presently limited to routine field office responsibilities of contract administration and are therefore not reported.

We will be pleased to provide any further information you might require.

Attachment
As stated

cc: T.E. Murley, RSR, NRC
R.W. Kiehn, EG&G
R.E. Tiller, ID

1915 293

I 5-36
Pressely
Ferguson
Crawford

Organization or Employee Name	Type of Activity, Assistance	Dates of Assistance	Results, Data:	Results Communicated To:	Continuing Assistance
Willis W. Bixby (DOE-ID)	On-site	April 6 to 13			None
John James (DOE-ID) Robert A. Long (DOE-ID) EG&G Procurement	Locating emergency pumps	April 6,7,8,9	Located pumps for TMI	TMI NRC site office W.Bixby,DOE-ID	None
Paul E. Litteneker (DOE-ID)	Assisted in communicating INEL test results, defining new RELAP runs of use to TMI.	April 15	Improved technical communication.	Tom Telford at TMI NRC site office.	None
Charles Gilmore (DOE-ID) J. Henscheid (EG&G)	Arranged to have TMI water sample brought to INEL and analyzed.	April 1 and April 13	Dissolved hydrogen content analyzed and reported.	J.Collins at TMI NRC site office.	None
George L. Vivian (DOE-ID)	Arranged for use of a mobile radiological survey lab from INEL at TMI.	April 6	Mobile unit being set up at TMI for effluent monitoring	Don Solberg, NRR NRC	None
Code Assessment* Personnel (EG&G) J. Dearien, et al	Vessel and core structural response calculations to possible H_2 detonation.	March 30 - April 1	Vessel head would remain intact but damaged; core would not withstand forces	Results verbally communicated to T. Murley at NRC Command Center.	None
L.J. Ybarrondo &* N. C. Kaufman (EG&G)	On-site technical assistance. Evaluate sequence of events during TMI Evaluate various natural circulation conditions.	April 1 - April 6			
J. Liebenthal* et al (EG&G) & Allied Chemical Personnel	H_2 and O_2 concentration in TMI primary coolant loop	March 31 and April 1	Calculations indicated small H_2 concentration	Results verbally communicated to T. Murley at NRC Command Center	None

* NRC Funded Work

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Organization or Employee Name	Type of Activity, Assistance	Dates of Assistance	Results, Data:	Results Communicated To:	Continuing Assistance
Code Development * Personnel (EG&G) P. North, et al	-Natural circulation calculations with RELAP for various cold shutdown conditions -Simulation (analytical) of TMI transient -Small break calculations	March 30 to present	-Natural circulation could exist dispite core damage -RELAP calc. the initial portion of accident 0-10 min. and from 100 min on. Results agree with TMI data -Calculations ongoing	-TMI site & NRC Licensing in Bethesda -G. Hollahan at NRC licensing -NRC licensing	None None Ongoing
Instrumentation * Personnel (EG&G) M. Stanley, et al	Experimentally evaluate capability of resistance temperature detector-RTD to measure pressurizer level.	April 4	Experiments proved positive	L.J. Ybarrondo (EG&G) at TMI site	None
Semiscale * Personnel (EG&G) D.J. Hanson, et al	-Experimentally evaluate means to vent hydrogen bubble -Experimental simulation of TMI sequence of events	March 30 - April 3 April 3 to present	-2 reports prepared. Info. indicated bubble could not be vented by depressurization -Test conducted, report in preparation	T. Murley at NRC Command Center & L.J. Ybarrondo at TMI site. T. Murley at NRC Research	None Yes
Billings Energy Co.* (Subcontract to EG&G) R. Billings, et al	-Evaluate means for degassing H_2 -H_2 solubility as a function of depressurization -Examine various catalysts for H_2 scavenging	April 1 to April 17	-Report prepared -Develop H_2 bubble growth relationships as a function of H_2 conc. & depressurization procedure. -Identified catalytic reduction of H_2 with O_2 on Platinum and nickel boride system as catalyst for scavenging H_2	-Final report to DOE-ID & NRC Research -Interim results sent to L. J. Ybarrondo (EG&G) & W.Bixby (DOE-ID) at TMI site for Industry Advisory Group-IAG as available.	None

* NRC Funded Work

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Organization or Employee Name	Type of Activity, Assistance	Dates of Assistance	Results, Data:	Results Communicated To:	Continuir Assistanc
Code Assessment* Personnel C.F. Obenchain et al	Performed calculations of core component temperatures during the accident	April 9 - 12	Temperatures were the same as clad temperatures	W. Johnston, NRC, followed by a letter report	None

*NRC Funded Work

1915 296

Department of Energy
Albuquerque Operations
Sandia Area Office
P.O. Box 5400
Albuquerque, New Mexico 87115

MAY 22 1979

Robert L. Ferguson, DOE Headquarters, Germantown, Md.
Attn: H. Feinroth

THREE MILE ISLAND ACCIDENT

Sandia Laboratories, Albuquerque, has been involved in assisting NRC in the Three Mile Island Accident.

Attached is correspondence from Sandia Laboratories to NRC that describes the type of assistance, names of Sandia and NRC personnel involved, and the results and data of work done.

Minimum assistance is being continued by Sandia Laboratories.

Gil E. Cordova
Area Manager

SBA:DBF

Enclosure:
As Stated

cc: W. B. Johnston, OSD, ALO
Attn: D. Foster, w/encl.

1915 297

Attachment 1

Sandia Laboratories

Albuquerque, New Mexico 87115

April 4, 1979

Mr. Joseph Murphy
Probabilistic Analysis Staff
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Joe:

I would like to summarize the assumptions made in the SANDIA-ORIGEN(1) calculations delivered today via Dave Ericson. The operating history of Three Mile Island-Unit 2 was based on data supplied by Dick Muranka and Jack Crooks of NRC (phone: 492-7735).

The new fuel composition is as follows:

Uranium - 2.66 w/o nominal enrichment:

U^{234}	0.03 w/o
U^{235}	2.66
U^{236}	0.29
U^{238}	97.02

Cladding:

	kg/MTU
Zircaloy 4	281.1
SS-309	21.67
Inconel	11.92

Core Size: 73.0 MTU*

Licensed Power: 2772 MW thermal

The reactor operating history is summarized in Table 1. The column labeled "Cumulative days" corresponds to the time steps used in the first and second burnup phases in the code runs.

*Core size has been revised to 81.893 MTU and the revised computer output was sent by Federal Express to Joe Murphy on April 9, 1979.

1915 298

The decay phases start with the radionuclide inventory at the time of reactor shutdown (corresponds to cumulative days = 186.1) and calculates the inventory at 12 hour intervals.

To provide long term data, I have made another run, two copies of which are enclosed. The two burnup subcases are not printed, simply to save paper; the data would be the same as you already have. Three decay subcases provide the inventory out to 10 days in 1 day steps, to 100 days in 10 day steps and to 1000 days in 100 day steps.

I hope this will assist in your analyses. Please do not hesitate to call (day or night) if there is anything further you need (work: FTS-475-3119, home: 505-298-1142).

Very truly yours,

Dave

Dr. David E. Bennett, III
Nuclear Facility Analysis
Division 4414

Copy to:
4231 S. A. Dupree
4400 A. W. Snyder
4410 D. J. McCloskey
Attn: G. B. Varnado, 4414

P. S. Also enclosed are: 2 micro-fiche copies of the runs delivered by Ericson.

2 micro-fiche copies of the enclosed runs

1 copy of the IEC article on the soluability of H_2 in water.

References:

1. D. E. Bennett, SANDIA-ORIGEN Users Manual, SAND79-0299, Sandia Laboratories, Albuquerque, NM, to be published April 1979.

TABLE 1. REACTOR OPERATING SUMMARY.

DATES	LENGTH OF PERIOD	CUMULATIVE DAYS	ON/OFF	% POWER (APPROX.)	POWER (MW)	CALC. BURNUP (MWD)	ACTUAL BURNUP (MWD)
SEPT 23-30	8	8	ON	30	830	6640	6902
OCT 1-13	13	21	OFF				
OCT 14-27	14	35	ON	71	1980	27720	27720
OCT 28-31	4	39	OFF				
NOV 1-4	4	43	ON	90	2500	10000	10117
NOV 5-30	26	69	OFF				
DEC 1-10	10	79	ON	90	2500	25000	
DEC 11-21	11	90	OFF				49904
DEC 22-31	10	100	ON	90	2500	25000	
JAN 1-2	2	102	OFF				
JAN 3-14	12	114	ON	90	2500	30000	30[illegible]36
JAN 15-31	17	131	OFF				
FEB 1-10	10	141	ON	92	2550	25500	
FEB 11	1	142	OFF				68940
FEB 12-28	17	159	ON	92	2550	43350	
MAR 1-28 (4 AM)	27.1	186.1	ON	92	2550	69105	UNKNOWN
						262225 MWD	

1915 300

Attachment 2

Sandia Laboratories

Albuquerque, New Mexico
Livermore, California

date April 4, 1979

to Distribution

from B. L. Gregory, 2140

subject Total Dose Effects on Pressure Transmitters

The attached pages summarize my discussions with the Foxboro Company and Bailey Meter Company concerning their pressure transmitters.

Schematics are being sent to Sandia if required for further study. This information was relayed to Charlie Miller at NRC on 4/3/79.

BLG:2140:edd

Distribution:
4400 A. W. Snyder
4410 D. J. McCloskey
4442 L. L. Bonzon
4343 F. N. Coppage

POOR ORIGINAL

SUMMARY April 3, 1979

Discussions with Foxboro Company on possible radiation effects on Model E11GH pressure transmitter (Joe Childs (617) 543-8750, X2128)

Units in Consideration

There were two sales orders to the Three-Mile Island Plant

1971 - 3 units

Earlier model

Designed to meet LOCA

Temp and pressure environment - not qualified for radiation.

Uses teflon in sealant in wiring and in wiring hardness.

Tests performed on standard units.
Two units were tested to 10 megarads and survived. The same amplifier was tested to failure.

One unit performed at 50 megarads and failed at 90.

One unit functioned at 56 megarads and failed at 90 - functional failure.

1977 - 4 units were shipped, qualified for radiation and LOCA.

They use Kapton insulation, polysulfone and glass - filled diapthalate in place of teflon.

4 units of this type were tested to failure in radiation.

2 units failed @ 130 megarads

2 units functional @ 220 megarads.

Other Info -

Reactor unit #1 at 3-Mile Island uses the earlier, non-hardened model.

Reactor Unit #2 (failed unit) may use older or newer model (check storage shelves)

Parts in Amplifier

2N711 PNP Germanium Mesa

2N3712 NPN Silicon annular 1 watt

35 total parts

The circuit schematic is being faxed to me and is being mailed.

POOR ORIGINAL

1915 302

[illegible]

SUMMARY

Discussions with Dick Brooster from Bailey Meter
(216) 943-5500 X 2394

Description of unit - A bellows moves a core rod in and out of a differential transformer as the transducer.

Functional blocks

- 1000 Hz oscillator
- Feedback regulator
- Diode Bridges
- Differential amplifier (2-stage discrete transistor amplifier)

Differential amplifier operates -10V to +10V

Units are on PC boards which are potted in transformer resin.

A separate PC board (not potted) is also used.

The wires connecting the various boards are teflon insulated.

The unit employs the temp. coefficient of diode I/V characteristics to provide temperature compensation.

There have been radiation tests performed on this unit. In 1969 a unit was tested to destruction which failed at 4.8×10^7 rads. (Teflon insulation failure.) A second unit was tested to $2 \times 10^{[illegible]}$ rads which passed.

The circuit schematic is being faxed to me and is being sent Federal Express.

POOR ORIGINAL

1915 303

Discussion with Jim Graham, Foxboro, April 3, 1979

Foxboro Transmitters

1977 Transmitters - 0 to 2500 PSIG

	Serial #	Tag #
1.	371 6439	620-006-RC-22-PP-5
2.	371 6440	620-006-RC-22-PP-6
3.	371 6441	620-006-RC-22-PP-7
4.	371 6442	620-006-RC-22-PP-8

Installed in Reactor Unit #2 -Reactor coolant pump middle seal cavity pressures.

Graham believes they were installed. (air shipped to site)

1971 Units

Application

Reactor Coolant

Pressure wide range.

0-2000 PSIG

3 Transmitters on same measurement .

E11 GM's

4 each - different applications

4 - Reactor Coolant narrow range - 1700-2500 PSIG

4 - Reactor coolant low range

2 - Steam generator

Feed water

Turbine Throttle

POOR ORIGINAL

1971 Units - Potential failure mode - integrity of transmitter due to effects of radiation on seal (in new units there is a Viton O ring with DC 704 on it, viton retains elongation).

Both units - Conax power lead pressure seal.

In 1971 unit - both the seal and packing gland contains teflon (confined - not in air).

If failure occurred, the resulting signal could range over entire scale.

1915 304

Attachment 3

HYDROGEN BURNING/EXPLOSION POTENTIAL

Conditions attained by combustion and by detonation were calculated analytically for rich hydrogen/oxygen mixtures of various stoichiometries. For the reaction

$$(1-x)H_2 + xO_2 \rightarrow 2xH_2O + (1-3x)H_2 ,$$

we estimated the heat release and temperature rise in combustion to be

$$Q, erg/g = (4.84)(10^{12})\ x/(2 + 30x)$$

$$\Delta T, {}^{o}F = 208080\ x/(6.8 - 2.4x)$$

for $x \leqq 1/3$. Pressure was calculated by the ideal gas law at constant volume.

Ideal detonation of an ideal gas mixture leads to a Chapman-Jouguet pressure, P_{cj}, determined by

$$P_{cj} = P_o + 2\rho_o(\gamma - 1)Q ,$$

from which we find, for this problem,

$$P_{cj} - P_o \cong 0.74x \quad GPa .$$

Results of calculations performed with this approach are for P_o = 1000 psi and T_o = 280°F are tabulated below:

x	ΔT, °F	$P_{combustion}$, psi	P_{cj}, psi
0.02	610	1800	3,100
0.05	1560	3000	6,400
0.12	3830	6000	14,000

Efforts were undertaken to numerically calculate the progress and effects of ideal detonation. For the larger bubble size (1500 ft³), the detonation wave was assumed to be initiated at one point in the gas mixture, to propagate as a plane wave, and to provide an impulsive load to the vessel. The calculated impulse,

POOR ORIGINAL

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Attachment 4

Hydrogen Solubility and Chemistry

Principal NRC Contact: (to whom the following information was given verbally)	John Larkins NRC Silver Spring, Maryland FTS: 427-4323
Secondary NRC Contact:	Joe Murphy FTS: 492-8377

1. Solubility (Kepler, Dosch, Harrah)

The solubility of H_2 in water is a function of H_2 overpressure only in the temperature range of interest (circa 280°F). The solubility can be expressed approximately as

$$S_{H_2}(ccH_2 \text{ at } STP/gmH_2O) = 1.24 \cdot 10^{-3}\ P_{H_2}(psi\ H_2)$$

For two pressures of interest, the solubilities are:

Pressure	Solubility H_2
350 psi	.43 ccSTP/gmH_2O
1050 psi	1.3

Thus depressurization from 1050 to 350 would produce an H_2 bubble on the order of 600 ft^3.

Reference: "Solubility of Hydrogen, Oxygen, Nitrogen, and Helium in Water at Elevated Temperatures," Pray, Schweickert and Minnich, Industrial and Engineering Chemistry, 44, 1146-1151 (1952).

2. Degasification Times (Kepler)

It was assumed that the letdown system operates at 20 gal/min. This rate translates to processing one reactor coolant system volume every three days. It was further assumed that processing one RCS volume reduces the H_2 concentration by one-half.

Starting with the RCS saturated at 1050 psi, approximately 1.5 RCS volumes must be processed . reach saturation at 350 psi. This would require approximately 4.5 days.

To reach a level of 10 ccSTP per kg of H_2O starting with saturation at 1050 psi would require that between 6 and 7 RCS volumes be processed, taking 18-21 days.

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3. Radiolysis (Hughes)

Initial radiolysis rate of pure water in a radiation field of 10^8 rad/hr is very high, but rapidly decreases with increasing H_2 overpressure to a very low value.

The exact rate cannot be calculated since several factors affecting the equilibrium and rate are known.

It was concluded that present reactor coolant chemistry conditions approximate those during cold shutdown except for the excessive radioactive contamination.

Data was provided by NRC that the steady state H_2 level in the coolant during normal operation due to radiolysis is 35-45 ccSTP/kg H_2O. It was concluded that as long as an H_2 overpressure is maintained, the H_2 concentration due to radiolysis after shutdown (radiation levels down by 10^{-2}) would be at least a factor of 10 lower. These concentrations are a factor of 100 or more below the solubility.

Reference: 1. "Formation of H_2 During Core Melt Accidents in Nuclear Power Plants with Light Water Reactors," Kerntechnik (Kertaa), V 19 (11), P 473-7, 1977, ISSN, 03685276.

2. Hydrogen Production in Containment after a Loss-of-Coolant Accident Light-Water Reactors," MRR-117, NTIS, H. Jahn, March 1973.

4. H_2 Gettering by Zircaloy (Wahn, Healey, Douglass, Sallach)

The uptake of hydrogen into Zircaloy 4 is approximately 1/3 that of Zircaloy 2. Hydrogen pickup in 600°F water takes approximately 100 days to reach the 10 ppm range in Zircaloy 4. Concentrations as low as 5 → 10 ppm can cause stress corrosion cracking in zirconium alloys. Pickup of hydrogen when the material was exposed to hydrogen-water vapor would have been much greater (on the order of 5 percent in 2 hours at 300°C) and would provide the greatest source of hydrogen in the material, which would make the material susceptible to stress corrosion cracking.

Hydrogen can be absorbed into the zirconium from the vapor phase even through oxide layers at elevated temperatures. This process has an incubation time dependent on the oxide thickness layer. This incubation time is a result of the diffusion time necessary for the hydrogen to diffuse by oxygen vacancy through the oxide layer.

Graphs from Corrosion in Nuclear Applications by Warr n E. Berry show the uptake of hydrogen under a variety of conditions and zirconium alloys. The papers by Katsumi Une and Nichio Nagasaka, et al, show the hydrogen pickup in hydrogen vapor and hydrogen-water environments.

Reference: 1. Corrosion of Zr Alloys, A symposium presented at 1963 Winter Meeting ANS, New York, November 20, 1963, ASTM #368.

2. Effect of Hydrogen on Behavior of Materials, Proceedings of the International Conference, Jackson Lake, Wyoming, 1975, Anthony W. Thompson and I. M. Bernstein, Ed., The Metallurgical Society of AIME.

3. Corrosion in Nuclear Applications, Warren E. Berry, John Wiley & Sons, New York, 1971.

4. Hydrogen Uptake of Zr and its Alloys during the Early Stages of Corrosion in Steam, Freer, Silvester, Wanklyn, AERE-R-4531, 1964.

5. Solubility of Hydrogen and Deuterium in Ti and Zr Under Very Low Pressure, M. Nagasaka and T. Yamashina, Journal of Less Common Metals, 45, 53-62, 1976.

6. Kinetics of Reaction of Zr Alloy with H_2, Katsumi Une, Journal of Less Common Metals, 57, P. 9?-101, 1978.

5. Zircaloy-Steam Reaction (Douglass)

John Larkins, NRC, is trying to make a hydrogen balance in order to determine how much hydrogen might have reacted with the uncorroded Zircaloy 2 caldding during cooldown of the core, how much hydrogen went into solution in the high-pressure water, and how much hydrogen remained in the gas phase. He was specifically interested in the amount of hydrogen released from the reaction between steam and Zircaloy 2. My comments were that under normal conditions (when the reactor acts as a PWR), the fraction of hydrogen generated finding its way into the metal is on the order of 30 to 50 percent depending upon the temperature and time (the fraction ususally increases after the transition to linear kinetics). The hydrogen dissolves in the Zircaloy 2 and eventually precipitates out as zirconium hydride. However, with increasing temperature, the hydride becomes less stable and eventually dissociates. The equilibrium partial pressure of hydrogen at 1200°C, the temperature presumed to have existed during the incident, is not known. Beck[1,2] has reported some pressure-composition isotherms for temperatures up to 1100°C. His 1100°C isotherm extends only to a value of about 0.5 for the H/Zr ratio, having a pressure of nearly 1000 mm Hg. This temperature-pressure composition is in the beta-phase field. The pressure value would be much higher at 1200°C. I stated that the amount of hydrogen in the metal at 1200°C would be very low unless the external hydrogen pressure was very high, i.e., greater than the equilibrium value to maintain a given concentration in the metal, whatever the pressure value happened to be.

I further stated that hydrogen would be taken up by the unreacted metal during cooldown. This presupposes that the hydrogen can migrate across an oxide film or that the film was cracked, in which case ingress of hydrogen could readily occur. It is thought that even if the films were crack-free, hydrogen can still diffuse through the oxide readily.

It was also pointed out that oxygen is an alpha-stabilizer, whereas, hydrogen is a beta stabilizer. The oxide film forms and dissolves simultaneously into the Zircaloy, thus transforming the beta Zircaloy to alpha Zircaloy at high temperatures. This occurs because the alpha-stabilizing influence of oxygen is much greater than the beta-stabilizing influence of hydrogen. The solubility of hydrogen is much less in the alpha phase than in the beta phase. This means that when much oxygen exists (as it does in the case of runaway corrosion), that the amount of hydrogen remaining in the Zircaloy will be much less at a given temperature than when oxygen is not present.

A number of names of people working in the area of high-temperature Zircaloy 2 steam reactions pertaining to LOCA were given along with some references. These are follows:

Dr. Adrian Roberts, EPRI
Dr. Richard Pawel, ORNL
Dr. R. Westerman, Battelle NW
Dr. R. Biederman, Worcester Polytechnic

Roberts is the program monitor on all aspects of zirconium-steam, zirconium-pellet interactions at EPRI and can give an excellent overview of the problems and who is doing what. Pawel, Westerman, and Biederman are all working on various aspects of high-temperature Zircaloy-Steam reactions.

Reference:

1. Beck, R. L., "Zirconium-Hydrogen Phase Diagram," ASM Transactions Quarterly, 55, 1962, P 542.

2. Beck, R. L., "Thermophysical Properties of Zirconium Hydride," ibid, P. 556.

3. Yurek, G. J., Cathcart, J. V., and Pawel, R. E., "Microstructures of the Scales Formed on Zircaloy 4 in Steam at Elevated Temperatures," Oxid Metals, 10, 1976, P. 255.

4. Biederman, R., "A Study of Zircaloy 4 Steam Oxidation Reaction Kinetics," Report EPRI NP-734, April 1978.

5. Westerman, R. E., "Zircaloy Cladding ID/OD Oxidation Studies," Report EPRI NP-525, November 1977.

6. Kelpfer, H. H., and Douglass, D. L., "Factors Limiting the Use of Zirconium Alloys in Superheaded Steam," ASTM Special Publication No. 368, Corrosion of Zirconium Alloys, 1963, P. 118..

6. Calculation of Zirconium Oxidation to Produce H_2

It was assumed that the following quantities of H_2 were produced during the course of the accident:

70,000 SCF which has burned
48,000 SCF currently in containment
100 CF at 1000 psi (current bubble)
12,000 CF of water saturated with H_2 at 1.3 ccSTP/gmH_2O

For a 73.0 MTU core (core size initially supplied by NRC), which would contain $2.02 \cdot 10^4$ kg of Zr, approximately 40 percent of the Zr would be required to react with water to produce the above quantities of H_2.

Revising the core size to 81.893 MTU reduces the fraction of Zr reacted to approximately 36 percent.

7. Potential Hazard from Adding H_2O_2 as an H_2 Getter (Whan)

Transition metal oxides are used as catalysts to decompose H_2O_2 in oxygen generators. It has been demonstrated that transitron metal ions in solution at the PPM levels can cause the rapid decomposition of H_2O_2 to produce oxygen.

We expressed concern that the addition of H_2O_2 to the coolant to getter H_2 could instead generate free O_2 which could then form an explosive mixture with the H_2 present.

Reference: 1. W. C. Schumb, C. N. Satterfield, R. L. Wentworth, "Hydrogen Peroxide," American Chem Soc Monograph No. 128, New York, Rhenehold Publications Company, 1955.

2. "The Effect of Fe^{+3}, Cr^{+3}, Nr^{+2}, and Mn^{+2} Ions on Decomposition of Hydrogen Peroxide Solutions," Kirk L. Shanahan, SAND78-1778, February 1979.

COPY

Sandia Laboratories

Albuquerque, New Mexico

May 8, 1979

Dr. Raymond di Salvo
Probabalistic Analysis Staff
U.S. Nuclear Regulatory Commission
Maryland National Bank Bldg. M/S 3106
Washington, D.C. 20555

Re: Contingency Vent-Filter for Three-Mile Island Unit II Reactor

Dear Ray:

In responding to your request regarding a contingency vent-filter system for the Three-Mile Island Unit II Reactor, we did not have time on the first go-around to present anything more than rough sketches. Now that the dust has settled, figuratively speaking, I would like to present a more organized account of the basic system that we proposed for you on April 13 and the options that go with it.

The primary contributors to this task, beside myself, were Walt Murfin (4413) and Harold Walling (1114).

I hope that the enclosed material will provide you with useful information to support the sketches we sent you earlier.

Please feel free to call upon us at any time.

Sincerely,

Allan Benjamin

Allan S. Benjamin

Nuclear Facility Analysis
Division 4414

Enc: (1) "Contingency Vent-Filter for the Three-Mile Island Unit II Reactor," with attached appendix.

(2) Blueprint of Vent-Filter System

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Distribution:

NRC: Mark Cunningham (PAS)

BCL: R. S. Denning
P. Cybulskis

Sandia: 1110 J. D. Kennedy
1114 J. H. Davis
1114 H. C. Walling
4400 A. W. Snyder
4410 D. J. McCloskey
4412 J. W. Hickman
4412 S. V. Asselin
4412 D. D. Carlson
4412 D. J. Murphy
4413 W. B. Murfin
4414 G. B. Varnado
4414 A. S. Benjamin

CONTINGENCY VENT-FILTER FOR THE THREE-MILE ISLAND UNIT II REACTOR

To handle the unlikely but postulated escalation of the Three-Mile Island accident to a core meltdown, Sandia Labs was asked by the NRC Probabilistic Analysis Staff on April 10, 1979, to rapidly prepare preliminary designs of vented filter systems that could be retrofitted to the reactor within a week to ten days. The purpose of the vent-filter was to relieve pressure from the containment building without significant radioactivity release if the structural integrity of the containment were threatened. Sandia responded to this request by submitting a preliminary vent-filter design to NRC on April 13, 1979, together with a list of design options that could be considered. A more detailed description of the design and the options is presented here, with supporting calculations being given in an attached appendix.

It should be emphasized that this system represents the results of a very short-time but concerted study that was directed at a very specific situation (the Three-Mile Island accident) and with very specific objectives (to provide a contingency system for venting the containment on short notice). It does not derive from the benefit of a detailed engineering study and is not in any way recommended as a model for operational reactors.

A schematic of the first (maximum capability) of the options proposed is shown in Fig. 1 and a more detailed drawing of the piping and valving aspects is presented on the enclosed blueprint. The main inlet line to the filter media attaches outside containment to an existing hydrogen control line (6-inch diameter) which comes off a normal purge exhaust line (36-inch diameter). This path for exhaust from containment was mandated by the fact that the normal purge lines were the only lines which were located above the containment water level, were large enough to handle the anticipated steam flow, were structurally adequate in the event of a pressure excursion, and could be valved open at the inside valve to permit outward-directed flow. All other lines above the water level are valved to prevent exhaust.

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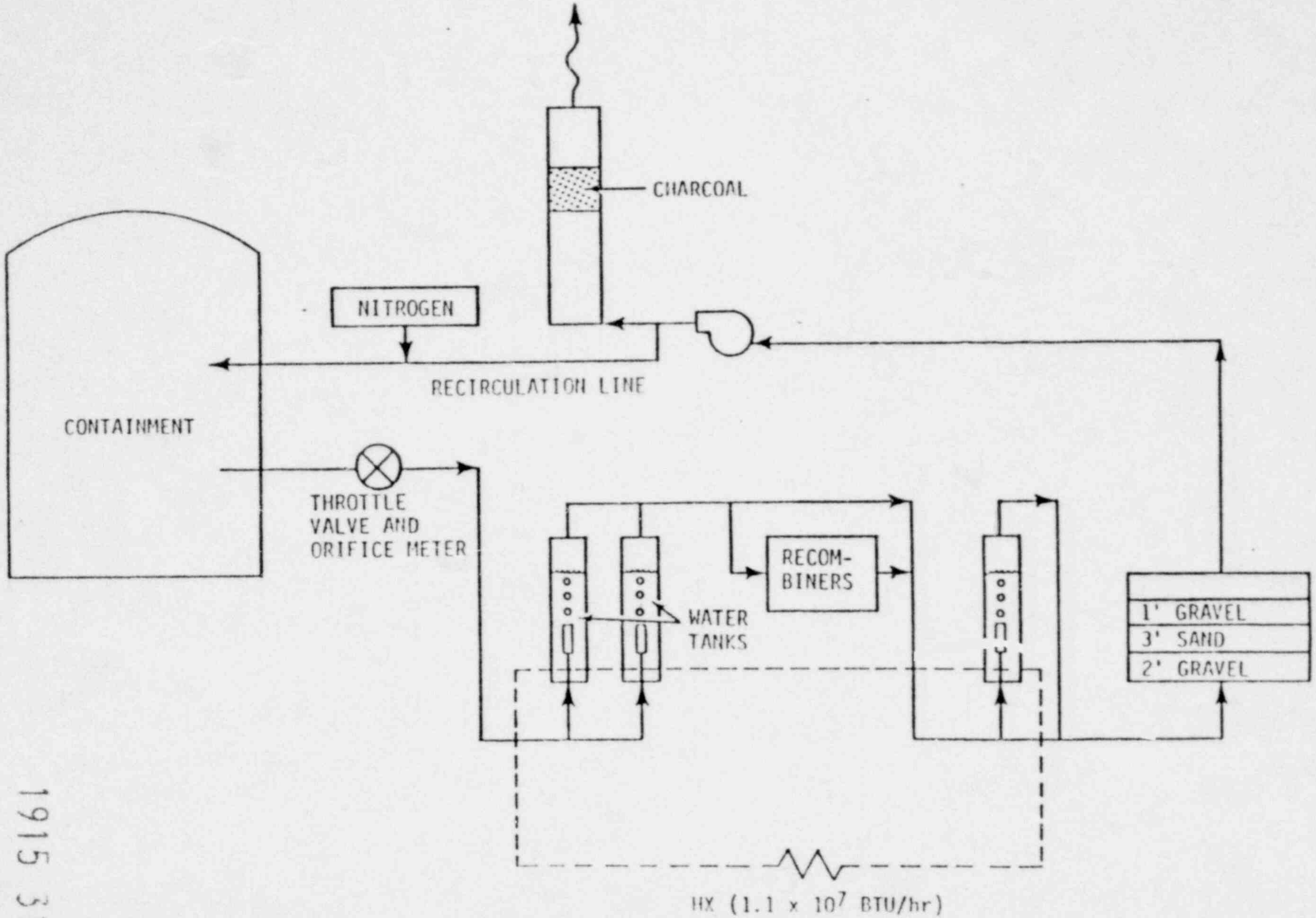

Figure 1. Schematic of Option 1 Design.

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There are four 36-inch normal purge lines that penetrate containment fairly close to one another at the interface with the auxiliary building. Two of these lines are normally used for exhaust and two for intake. At the time of the design effort, according to reports from the site*, one of the exhaust and one of the intake lines were being used for hydrogen recombination and recirculation. The plan for the vent-filter was to utilize the other containment exhaust line as an intake for the vent-filter and the other containment intake line as a means for optionally recirculating the filter exhaust back into the containment (see blueprint).

The major uncertainty regarding exhaust and intake was the possibility that the inside containment isolation valves would fail if the containment environment became too severe. The electrical insulation was believed to be sensitive to high radiation levels together with high temperatures, and a failure of the insulation would cause the spring-loaded valves to fail in the closed position. According to Thesing*, even if such a failure were to occur, it would be quite likely that the valves could be kept open with instrument air. As a precaution, however, it was considered important to utilize at least one of the other two normal purge lines, and preferably both of them, as a redundant intake for the vent-filter. In case all the valves should fail closed and the instrument air should fail to open them, a further precaution should be taken by making available the means for hot-tapping/drilling through an isolation valve.

At the intake to the vent-filter, a throttle and orifice meter are provided. The throttle is needed to allow careful control of the flow, since the system is not designed to accomodate large flow rates for indefinite periods. The orifice is provided to rule out the possibility of excessive flow rates that could blow out the system. The diameter of the orifice is specified as 3.3 inches, which would limit

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*Harold Walling of Sandia conducted several telephone conversations with Jim Thesing, Bechtel Corp., between April 11 and 13 to ascertain information pertinent to the vent-filter design effort.

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the exhaust flow to about 3000 cfm at a containment pressure of 45 psia.* Operating in the release mode, the time to reduce the containment pressure from 45 psia to 35 psia would be about eight hours, assuming that the water inventory in the containment sump is 700,000 gallons (i.e., approximately two refueling water storage tank loads). After 24 hours, the pressure would be reduced to about 28 psia. These calculations are based on utilization of the system at 21 days after power shutdown; the details are given in the attached appendix.

The primary filtering medium was selected to be water, appropriately treated with an additive (e.g., sodium hydroxide or sodium thiosulfate) to enhance iodine capture. The advantages of water pools in this instance are as follows:

(1) They have proven effective for their heat sink and steam condensation capabilities (e.g., BWR pressure suppression pool).

(2) They have a fairly high efficiency for particulate entrapment (>98%) and for capture of elemental iodine (>90%).

(3) They are inherently resistant to blasts, fires, and other hazards.

(4) They can be easily cooled by means of standard heat exchangers.

(5) Water is readily available and requires no special preparation, such as grading or purification.

The primary disadvantage of water pools is that they must be kept subcooled in order to maintain their capacity for condensing the steam and capturing the entrained fission products.

Sand filters with overlying and underlying gravel layers are also attractive for fission product decontamination under high flow loadings, but they have the following disadvantages: (1) they require large amounts of gravel in order to maintain their heat sink capability, since they cannot be readily cooled by heat exchangers, (2) they

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*The flow will not be choked under this condition. The maximum flow rate corresponding to isentropic choked flow is about 13,300 cfm. See appendix.

must be carefully graded and free from impurities that could cause excessive pressure drops, and (3) they have not been adequately tested in certain areas, such as their ability to drain the condensed water without becoming clogged. On the other hand, their efficiency for capturing particles is so great (>99.9%) that their use as backup to the water pools should be considered (see Fig. 1 and blueprint). Other considerations regarding the choice of filtering media and the design of the system may be found in a draft report* that was forwarded to NRC/PAS on April 11, 1979.

As a final filtering agent before release to the environment, the design calls for charcoal filters to be situated in the station vent. Assuming that the charcoal is impregnated with potassium iodide or triethylenediamine (TEDA) to enhance the capture of methyl iodide, the trilevel filtration capability of water, sand, and charcoal provides a high decontamination factor for all fission product species except the noble gases. It is important to point out, however, that the charcoal cannot be relied upon to provide the bulk of the filtering capability. Charcoal filters will be rendered useless by relatively small amounts of water or particulate and will ignite if heavily loaded with fission products or subjected to high gas temperatures. In addition, they usually require a prefilter to clean the incoming flow and an afterfilter to catch any contaminated charcoal particles that become dislodged and entrained in the flow.

For the Three-Mile Island situation, the water pools and gravel/sand filter were designed to be contained in portable tanks that were reportedly available on site. Initially, consideration was given to locating the water pool in a single large tank, such as the refueling water storage tank, the spent fuel storage pool, or the condensate storage tank for Unit I. For one reason or another, however, these tanks were all unavailable. The design shown in the blueprint utilizes ten 50,000-gallon water tanks in parallel to achieve a total water pool capacity of 500,000 gallons. Assuming that the tanks are

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*A. S. Benjamin, "Issues Affecting the Feasibility and Effectiveness of Vent-Filtered Containments," draft Sandia report dated March 1979.

initially half full, calculations show that the system can operate for at least 14 days in the recirculation mode and 28 days in the release mode before the tanks fill up and require draining (see appendix). Based on these results, it would be easily possible to operate with fewer tanks, if necessary.

The water tanks are designed with a single-loop direct heat exchanger that circulates water from the tanks through the exchanger and back into the tanks. A direct heat exchanger was considered to be preferable to an indirect one in this case because it is easier to design and construct. It was anticipated that the spent fuel cooling system could be diverted to the portable water tanks to provide the heat exchange capability, since it was not being used for its intended purpose of cooling spent fuel. Each of the two spent fuel coolers has a cooling capacity of 5.5×10^6 BTU/hr. If both coolers were operating, the system would be capable of removing enough heat to keep the water pools subcooled indefinitely, even at the maximum venting rate (see appendix).

Each water tank is equipped with level indicators and a thermocouple, so that the operator of the system can determine the water temperature and level. If the temperature approaches saturation in one of the tanks, the operator should be prepared to valve it out of the system until its temperature can be reduced by the heat exchanger. If the water level becomes too high due to steam condensation, the operator should drain it into a condensate tank provided for that purpose. Provision is made for the condensate tank itself to be drained into tank cars if its capacity is reached.

After water scrubbing, the remaining noncondensible gases are expected to consist largely of hydrogen, carbon dioxide, air, and noble gases. If one is operating in the recirculation mode, it would of course be inadvisable to inject a hydrogen-rich gas mixture back into the containment, due to the explosion hazard. Even in the release mode, it would be desirable to reduce the risk of a hydrogen explosion in the lines that could damage the system. For these

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reasons, hydrogen recombiners are provided in the design, followed by an additional water tank to cool the gases emitted from the recombiners. It would be preferable to have several recombiners operating in parallel, as opposed to a single one, so that relatively high rates of flow could be accomodated. It should also be mentioned that the piping and valving used between the recombiners and the cooling tank would be required to sustain very high temperatures.

As an alternative to hydrogen recombination, it has been suggested that an injection of E. I. du Pont's Halon 1301 into the line might prevent any possible hydrogen explosion because of its known anti-deflagration characteristics. Since it has not been tested in a radiation environment, however, its use at this time is probably ill-advised.

The gravel/sand filter is included as an option primarily for the release mode in order to provide an added filtration capability and to dry the water-saturated gas flow prior to its encounter with the charcoal filters. The required size of the gravel/sand filter is estimated to be 225 sq. ft. in area and 6 ft. in height, based on the assumption that all the steam exhausted from containment is condensed in the upstream water tanks.

The water tanks, hydrogen recombiners, and sand filter are located below grade to reduce radiation exposure to personnel at ground level. It was understood that significant excavation had already been accomplished for purposes of contamination control and that excavation was in general no problem. While placement below-grade is recommended, a complete burial of the system is considered unnecessary.

As mentioned earlier, the vent-filter system incorporates a dual capability for recirculating the noncondensibles back into containment or for venting them via the station vent to the atmosphere. In the recirculation mode, the flow is sustained by a blower, and a nitrogen injection capability is pr[illegible] to purge the containment and reduce the likelihood of hydrogen explosion. The blower can also be used in the release mode.

In the release mode, the vent-filter should be effective against a containment overpressurization for an indefinite period of time if the water tanks are drained periodically. In the recirculation mode, however, containment overpressurization may eventually occur as a result of the accumulation of noncondensibles in containment. The time to containment failure after the initiation of recirculation is calculated to be more than 20 days, as compared to about 7 days with no venting of any kind (see appendix).

Various design options have been identified and are listed in Table I. To simplify the system, one could delete the recirculation line, the sand filter, and/or the charcoal filters (Options 2, 3, and 4) at some expense in flexibility and filtration capability. As an alternative to release or recirculation, one could direct the exhaust into the containment of Unit I, which had been shut down for maintenance (Option 5). A summary of the advantages and disadvantages of these options is given in Table I.

It would be worthwhile to reemphasize that this system was designed for expedience to serve as a contingency at Three-Mile Island. It therefore lacks many of the automatic controls and sophistications that one would want to include in a system of this sort given more time to design and construct it. The operation of the system requires intelligence and forethought. The operator must be prepared to execute the following actions at certain times:

(1) Drain any water tank or sump in the system that accumulates too much water. Do not let the tanks overflow.
(2) Valve out any tank if the water temperature approaches saturation (212° F).
(3) Limit the flow rates to levels that do not jeopardize the system.
(4) Do not recirculate a hydrogen-rich mixture back into containment.

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TABLE I

Design Options

Option Number	Description	Advantages	Disadvantages
1	As shown in Fig. 1.	Maximum filtration. Can recirculate back into containment if pressure transient is not too rapid, otherwise can discharge through station vent. Nitrogen purge for containment.	Amount of construction required.
2	Delete recirculation line and blower.	Fewer containment penetrations.	Less flexibility. Can only be used in release mode.
3	Delete sand filter and/or charcoal filters.	Availability of material. Fewer construction problems. Minimal impact on recirculation mode.	Forfeits additional filtration capability needed for release mode.
4	Delete recirculation line, sand and charcoal filters, and recombiners.	Minimum construction impact.	Minimum flexibility, lower filtration capability. Greater chance for H_2 explosion in vent line.
5	Redirect recirculation line to Unit I containment.	Same advantages as Option 1.	Contaminates Unit I containment.

(5) Watch for leaks and monitor radiation.

Care must be taken as well in the construction of the system, particularily in the following areas:

(1) Provide adequate pipe diameters throughout system.
(2) Spike water tanks with appropriate additives.
(3) Watch materials in high-temperature section following recombiners.
(4) Connect to more than one outlet from containment.
(5) Have contingency available for hot-tapping containment.

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APPENDIX

Estimation of Vent-Filter Requirements as a Contingency for the Three-Mile Island Unit II Reactor

CONTENTS

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1. NOMENCLATURE

Symbols	Definitions
A	Area (ft^2)
B	Dimensionless function defined by Eqn. (10)
c_p	Specific heat at constant pressure (BTU/lbm-°R)
C_D	Discharge coefficient
F	Dimensionless function defined by Eqn. (11)
$f_{NC,conc}$	Mass fract'-n of noncondensible components in concrete
g	Acceleration of gravity (ft/sec^2)
g_o	Conversion constant (32.2 $lbm\text{-}ft/lb\text{-}sec^2$)
H	Height of water (ft)
h	Specific enthalpy (BTU/lbm)
H_{eff}	Effective heat of ablation of concrete (BTU/lbm)
h_{fg}	Latent heat of vaporization of water (BTU/lbm)
M	Molecular wei.ht
m_{sump}	Mass of water in the sump (lbm)
$\dot{m}$	Mass flow rate (lbm/min)
p	Absolute pressure (lb/ft^2)
Q_{vent}	Venting rate (ft^3/min)
$\dot{q}$	Heat rate (BTU/min)
R	Universal gas constant (1545 ft-lb/lb mole-°R)
T	Absolute temperature (°R)
t	Time (min)
V_c	Free volume in containment (ft^3)
γ	Ratio of specific heats
ρ	Mass concentration, or density (lbm/ft^3)
ω	Index indicating release mode (if ω = 0) or recirculation mode (if ω = 1)

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1. NOMENCLATURE (Continued)

Subscripts	Definitions
ch	Choked flow
circ	Circulated into the containment atmosphere
dec	Decay
HX	Heat exchanger
in	Injected into the containment atmosphere
NC	Noncondensibles
o	Outside (ambient)
or	Orifice
sat	Saturation
st	Steam
w	Water

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2. INTRODUCTION

This appendix provides supporting calculations for the estimates of design requirements for the Three-Mile Island contingency vent-filter.

The assumed situation inside containment is shown schematically in Figure A-1. The core is assumed to have melted through the reactor vessel and is interacting with the concrete in the reactor cavity. This interaction produces an injection of noncondensible gases (primarily H_2 and CO_x) into the containment atmosphere at a rate $\dot{m}_{NC,in}$. The containment temperature atmosphere, consisting of a mixture of condensible gases (mainly steam) and noncondensible gases (mainly air, H_2, CO_x, and noble gases), is vented at the rate ρQ_{vent} starting at time t_o. The steam is condensed outside containment, whereas the non-condensibles may be recirculated back into containment, at the rate $\rho_{NC}Q_{vent}$, or may be released through the station vent. The containment atmosphere is in equilibrium with the water in the containment sump, which replaces a portion of the vented steam at a rate of $\dot{m}_{st,in}$.

1915 326

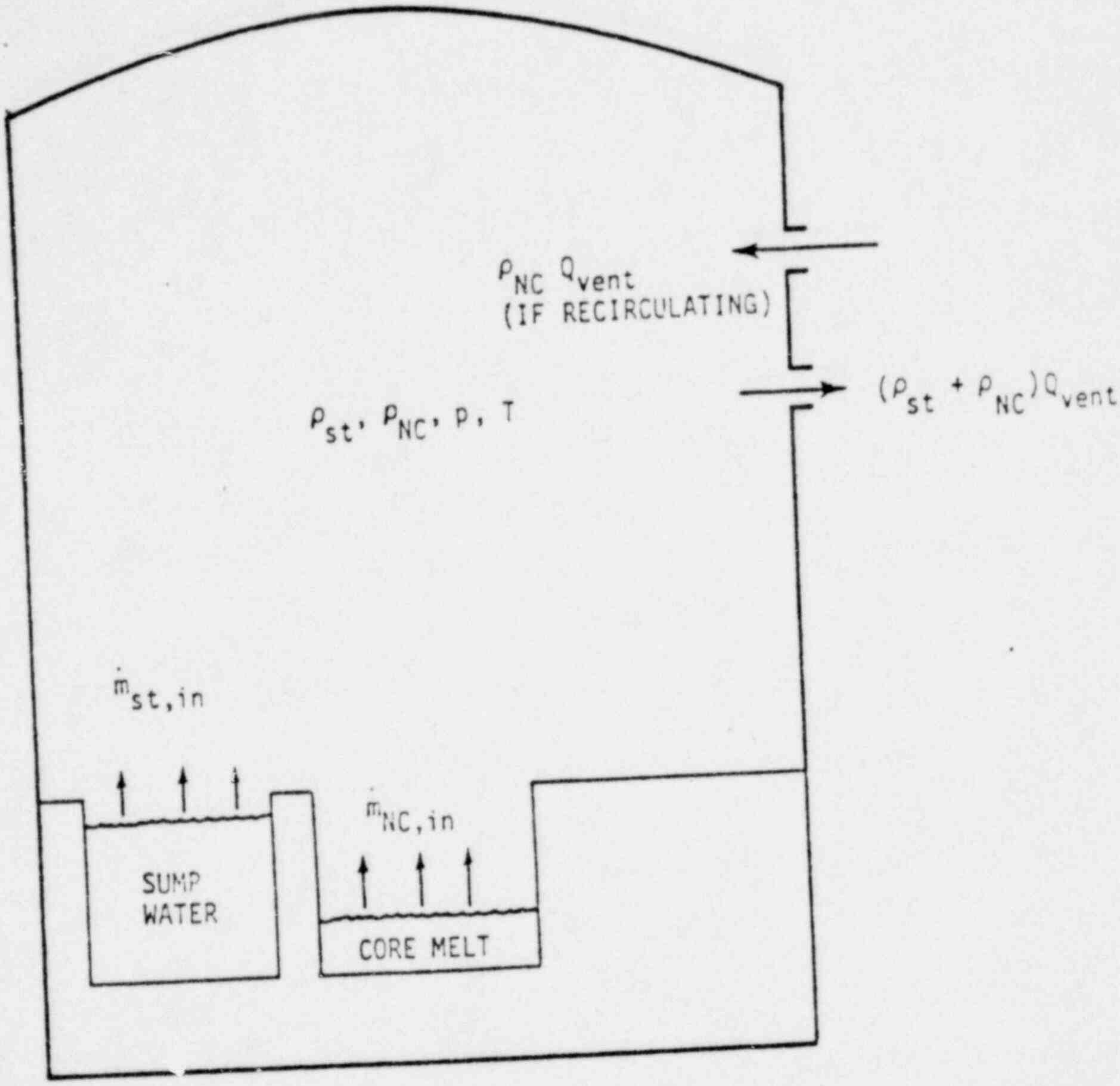

Figure A-1. Schematic of Containment During Core Meltdown

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3. DESIGN BASIS APPROXIMATIONS (RELEASE MODE)

Because the water content in the sump is expected to be much greater than the steam inventory in the containment (i.e., 700,000 gallons in the sump as compared to an equivalent 17,000 gallons water content in the atmosphere), it is possible for design purposes to make the conservative assumption that the sump reservoir is infinite. Assuming as well that the sump water maintains equilibrium with the containment atmosphere, the temperature and hence the partial pressure of water vapor in the containment remains constant. Thus any steam that is vented from containment is replaced by vaporization from the sump, viz.

$$\dot{m}_{st,in} \approx \rho_{st} Q_{vent}$$

The pressure variation in containment is then governed by the venting of noncondensibles, and the minimum design venting rate (i.e., that which arrests the containment pressure without reducing it) is

$$(Q_{vent})_{min} = \frac{\dot{m}_{NC,in}}{\rho_{NC}}$$

The mass influx of noncondensibles can be estimated by assuming that 100% of the decay heat in the molten core is transmitted to the concrete and that the concrete is thermally in quasi-equilibrium (i.e., that the penetration rate of the isotherms is equal to the melting rate of the surface). Thus

$$\dot{m}_{NC,in} \approx f_{NC,conc} \frac{\dot{q}_{dec}}{H_{eff}}$$

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Assuming that $\dot{q}_{dec}$ = 2.8 MW (from ORIGEN calculations corresponding to the Three-Mile Island core after 20 decay days), that H_{eff} = 1330 BTU/lbm (from considerations of the sensible heat and heats of decomposition and melting in concrete) and that $f_{NC,conc}$ = 0.27 (a representative value for limestone concrete), it follows that $\dot{m}_{NC,in} \approx 32$ lbm/min. Further assuming that ρ_{NC} = 0.067 lbm/ft^3 (from Battelle calculations at a 21-day decay time), it follows that $(Q_{vent})_{min}$ = 470 cfm.

The maximum design venting rate can be considerably larger than the minimum value as long as the heat exchanger is capable of removing the latent heat of the steam that is condensed in the water bath. (The decay heat produced by fission products captured in the water bath is calculated to be very small in comparison.) Thus

$$(Q_{vent})_{max} = \frac{\dot{q}_{HX}}{\rho_{st}\, h_{fg}}$$

Assuming a heat exchange capacity of 1.1×10^7 BTU/hr (the capacity of the spent fuel pool cooling system) and a steam density of 0.066 lbm/ft^3 (from Battelle's calculations), it follows that $(Q_{vent})_{max} \simeq 3000$ cfm. At this venting rate, the rate of water accumulation in the water tanks would initially be about 35,000 gallons/day, although this value would decrease during operation (see next section).

To prevent excessive flow rates that could compromise the system, an orifice plate should be included near the entrance to the system. The flow through this orifice, even at the containment venting pressure of 45 psia, is not expected to be choked, and so the flow rate can be expressed by the following isentropic flow relation modified by a discharge coefficient:

$$\frac{Q_{vent}}{A_{or}} = C_D \left| \frac{2\gamma}{\gamma - 1} \left[1 - \left(\frac{p'}{p}\right)^{(\gamma-1)/\gamma} \right] \frac{g_o p}{\rho} \right|^{1/2}$$

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The pressure upstream of the orifice, p, can be taken to be the containment pressure, but the downstream pressure, p', depends upon the details of the venting system (i.e., the pressure drops through the ducts, valves, pumps, water lutes, water pools, sand filter, etc.). If the system is designed to minimize pressure drops (i.e., large ducts, efficient lutes, etc.), then the major pressure drop in the system can be taken to be the hydrostatic pressure difference in the water tanks. Hence,

$$p' \approx p_o + \rho_w g\, H/g_o$$

Using H = 20 ft, p = 45 psia, ρ = .133 lbm/ft^3, Q_{vent} = 3000 cfm, γ = 1.4, and C_D = 0.6, it follows that A_{or} should be about 0.061 ft^2 and hence that the orifice diameter should be about 3.3 inches. The corresponding isentropic choked flow venting rate, given by

$$\left(\frac{Q_{vent}}{A_{or}}\right)_{ch} = \left[\frac{\gamma(\gamma + 1)}{\gamma - 1}\,\frac{g_o p}{\rho}\right]^{1/2}$$

is calculated to be 13,300 cfm.

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4. CONTAINMENT RESPONSE (RELEASE OR RECIRCULATION MODE)

Dropping the assumption about an infinite water sump, it is possible to write down the equations of mass and energy conservation in the containment in a fairly simple form if it is assumed that (1) the heat transfer to the building structures is small and (2) the sump water remains in equilibrium with the atmosphere. These equations are as follows:

Conservation of Species: Steam

$$V_c \frac{d\rho_{st}}{dt} = \dot{m}_{st,in} - \rho_{st} Q_{vent} \tag{1}$$

Conservation of Species: Noncondensibles

$$V_c \frac{d\rho_{NC}}{dt} = \dot{m}_{NC,in} - \rho_{NC} Q_{vent} (1 - \omega) \tag{2}$$

Conservation of Energy: Sump and Atmosphere

$$V_c \frac{d(\rho h)}{dt} + \frac{d(m_{sump} h_w)}{dt} = \dot{m}_{NC,in} h_{NC,in} - \rho Q_{vent} h$$

$$+ \rho_{NC} Q_{vent} h_{NC,circ} \omega \tag{3}$$

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where

$$
\left.\begin{aligned}
\rho &= \rho_{st} + \rho_{NC} \\
\rho h &= \rho_{st}(h_w + h_{fg}) + \rho_{NC}\, h_{NC} \\
h_w &= c_{p,w}\, T \\
h_{NC} &= c_{p,NC}\, T \\
h_{NC,in} &= c_{p,NC}\, T_{NC,in} \\
h_{NC,circ} &= c_{p,NC}\, T_{circ} \\
\dot{m}_{st,in} &= -\frac{dm_{sump}}{dt} \\
\omega &= \begin{cases} 0 \text{ for release mode} \\ 1 \text{ for recirculation mode} \end{cases}
\end{aligned}\right\} \quad (4)
$$

Introducing Equations (1), (2), and (4) into Equation (3), the equation for energy conservation can be written as follows:

$$
\begin{aligned}
&\left[(\rho_{st}\, c_{p,w} + \rho_{NC}\, c_{p,NC})\, V_c + m_{sump}\, c_{p,w}\right] \frac{dT}{dt} \\
&= -\dot{m}_{st,in}\, h_{fg} + \dot{m}_{NC,in}\, c_{p,NC}\, (T_{NC,in} - T) \\
&+ \rho_{NC}\, Q_{vent}\, c_{p,NC}\, (T_{circ} - T)\omega
\end{aligned}
\quad (5)
$$

Together with these is the equation for phase equilibrium:

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$$
\frac{dp_{sat}}{dt} = \frac{dp_{sat}}{dT} \frac{dT}{dt} \quad (6)
$$

as well as the perfect-gas equations of state:

$$p_{sat} = \frac{\rho_{st}}{M_{H_2O}} RT \tag{7}$$

$$p = \left(\frac{\rho_{st}}{M_{H_2O}} + \frac{\rho_{NC}}{M_{NC}}\right) RT \tag{8}$$

By combining Equations (1), (6), and (7), solving for dT/dt, and substituting into Equation (5), one can derive the following equation for $\dot{m}_{st,in}$:

$$\dot{m}_{st,in} = \rho_{st} Q_{vent} \frac{1 + BF}{1 + F} \tag{9}$$

where

$$B = \frac{\dot{m}_{NC,in} c_{p,NC} \left(T_{NC,in} - T\right)}{\rho_{st} Q_{vent} h_{fg}} + \frac{\rho_{NC} Q_{vent}\, c_{p,NC} \left(T_{circ} - T\right) \omega}{\rho_{st} Q_{vent} h_{fg}} \tag{10}$$

and

$$F = \frac{\left[\frac{T}{p_{sat}} \frac{dp_{sat}}{dT} - 1\right] \rho_{st} V_c h_{fg}}{\left[(\rho_{st} V_c + m_{sump}) c_{p,w} + \rho_{NC} V_c c_{p,NC}\right] T} \tag{11}$$

Closure is formed by borrowing the following expressions from the preceding section:

$$\dot{m}_{NC,in} = f_{NC,conc} \frac{\dot{q}_{dec}}{H_{eff}} \tag{12}$$

1915 333

$$Q_{vent} = c_D A_{or} \left\{ \frac{2\gamma}{\gamma - 1} \left[1 - \left(\frac{p'}{p} \right)^{(\gamma-1)/\gamma} \right] \frac{g_o p}{\rho} \right\}^{1/2} \tag{13}$$

$$p' = p_o + \rho_w gH/g_o \tag{14}$$

The solution of these equations involves a straightforward first-order time integration. At a particular time, calculate $\dot{m}_{NC,in}$ and Q_{vent} from Equations (12) through (14), obtain dp_{sat}/dT from steam tables, calculate $\dot{m}_{st,in}$ from Equations (9) through (11), update ρ_{st}, ρ_{NC}, and T from Equations (1), (2), and (5), calculate p from Equation (8), and update m_{sump}. The procedure is repeated at each time step.

Results of these calculations are shown in Figures A-2 and A-3. Values of p and T initially were taken to be 45 psia and 244° F respectively, from Battelle's calculations. The initial value of m_{sump} was taken to be 5.6×10^6 lbm, the equivalent of approximately two refueling water storage tank loads. The orifice diameter was taken to be 3.34 inches and the hydrostatic water head to be 20 feet. The decay heat in the core was taken as 2.8 MW, and no credit was taken for the expected reduction of this value with time. The molecular weight of noncondensible gases was taken to be 29, their temperature emerging from the molten core was taken as 5000° F, and their temperature upon recirculation was taken as 150° F. These values are, of course, rather gross estimates.

It is to be understood that these calculations give an indication of the likely effectiveness of this vent-filter in either a release or recirculating mode, but have several shortcomings that would have to be corrected in a more detailed analysis. The primary areas where improvements are needed are as follows:

(1) Better representation of the pressure drop in the vent-filter system.

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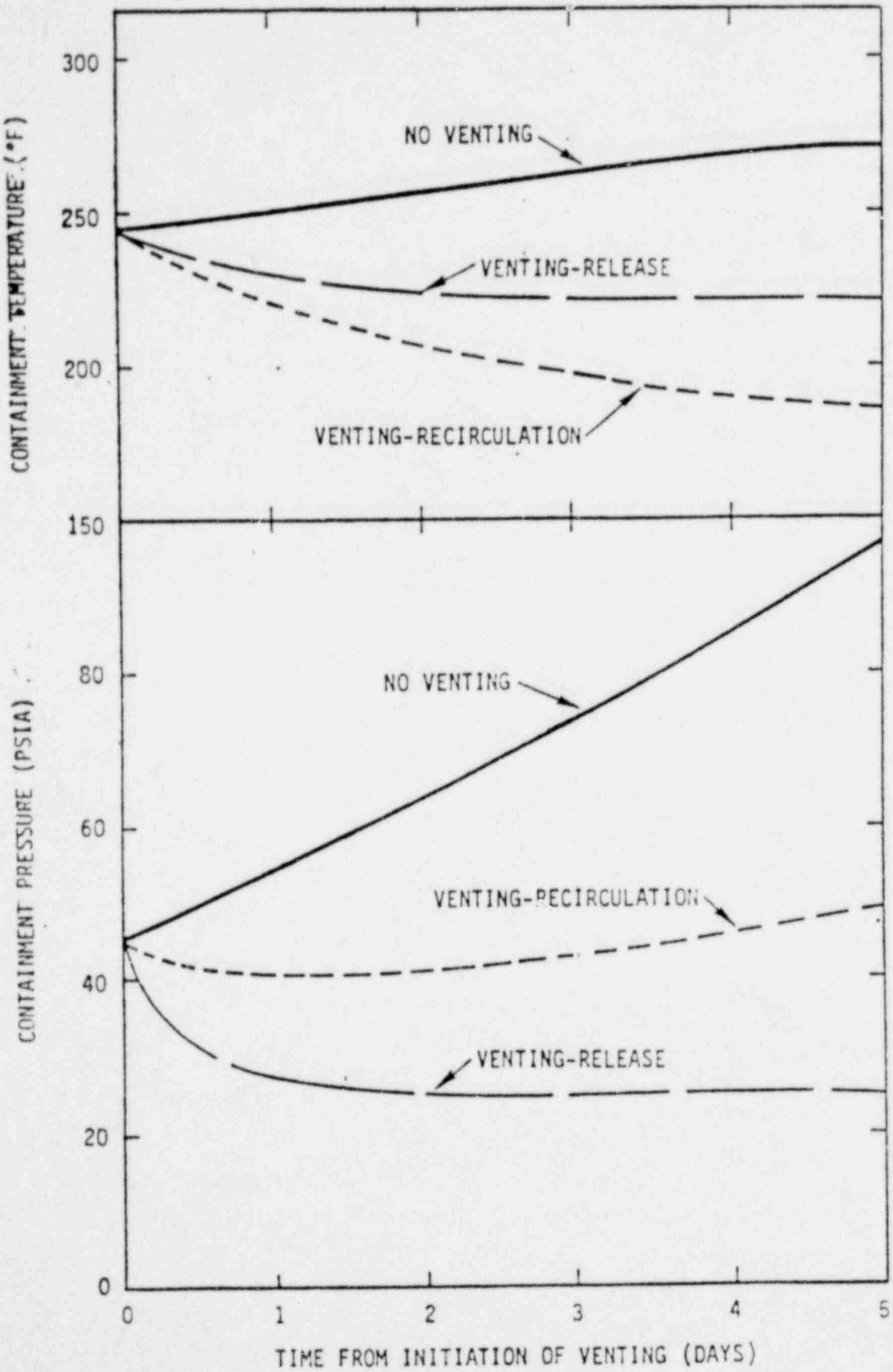

Figure A-2. Containment Temperature and Pressure as a Function of Time

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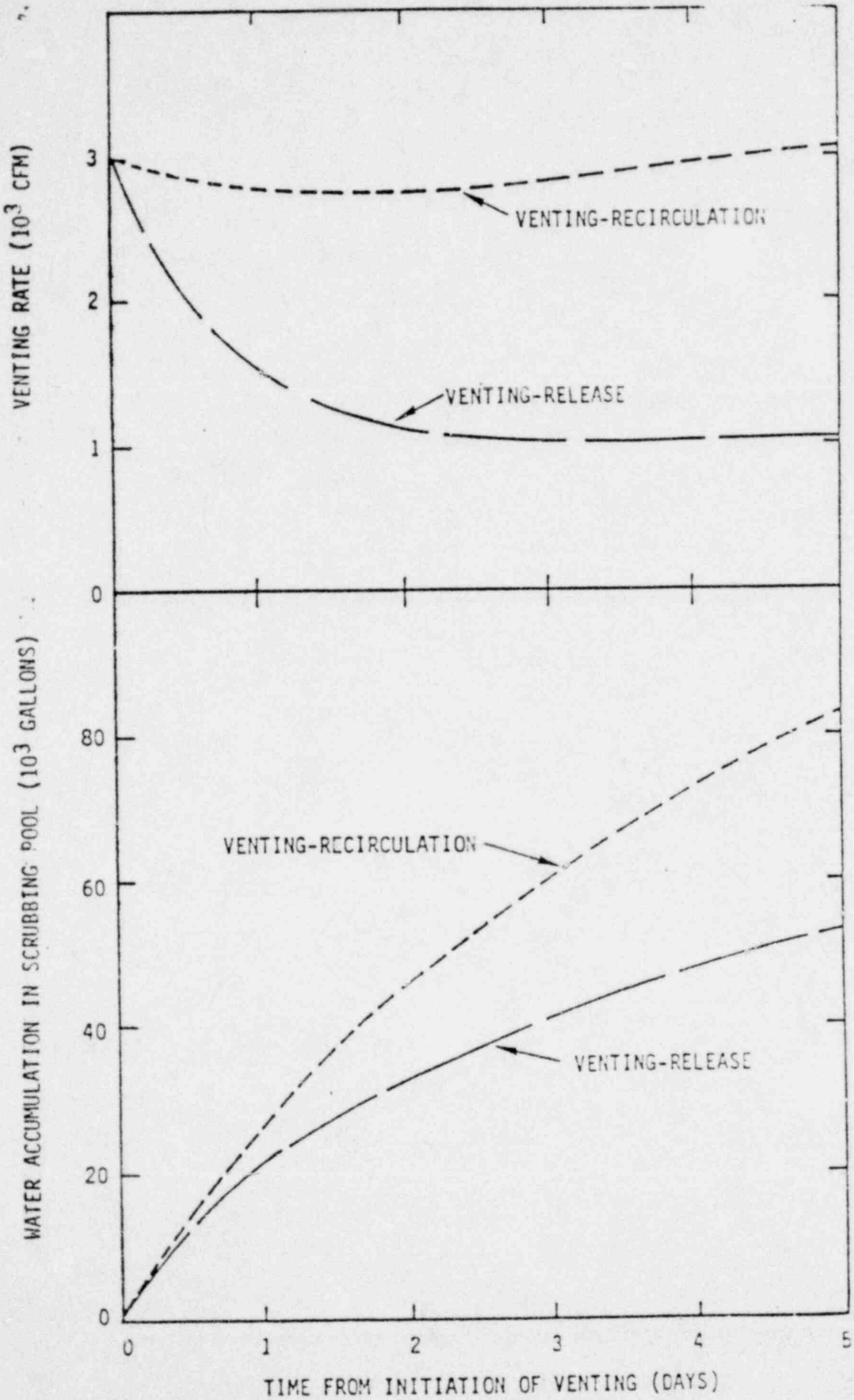

Figure A-3. Venting Rate and Condensate Accumulation as a Function of Time

(2) Coordination with concrete thermal response model to give better estimate of noncondensible gas generation.

(3) Better characterization of noncondensible species, particularly to define equivalent molecular weight.

(4) Better estimates of $T_{NC,in}$ and $T_{NC,circ}$.

(5) Inclusion of containment structural heat sinks.

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(2)

Department of Energy
Chicago Operations and Regional Office
9800 South Cass Avenue
Argonne, Illinois 60439

JUN 6 1979

Robert L. Ferguson, Program Director for Nuclear Energy
Office of Nuclear Energy Programs, HQ

INFORMATION REQUESTED BY THE PRESIDENT'S COMMITTEE ON THREE MILE ISLAND

Attached to this memorandum are several internal memorandums from Argonne National Laboratory (ANL), which describe the input personnel had in activities at Three Mile Island. As discussed between Mr. Feinroth and Mr. Jascewsky, of my office, by telephone, the ANL's Radiological Assistance Team's response activities are not included since this information has already been provided to Headquarters.

Any questions concerning the material provided, should be directed to Edward J. Jascewsky on (FTS) 972-2253.

Original Signed by
Fred C. Mattmueller

Robert H. Bauer
Manager/Regional Representative

OES:EJJ

Enclosures:
1. Memo., Rest to Honekamp, dated 4/1[illegible]/79
2. Memo., Gehl to Weeks, dated 5/29/79
3. Memo., Korbus to DeLorenzo, dated 5/2[illegible]/79
4. Memo., Frost to File, dated 5/30/79
5. Memo., Frost to File, dated 5/31/79
6. Memo., Frost to DeLorenzo, dated 5/31/79
7. Memo., McConnell to DeLorenzo, dated 5/31/79
8. Memo, Cunningham to Burris, dated 6/1/79

ARGONNE
NATIONAL
LABORATORY

INTRA-LABORATORY MEMO

RECEIVED
MAY 30 1979
Office of Operational Safety
File
Action

May 29, 1979

TO: C. A. DeLorenzo Director, OOS

FROM: H. M. Korbus *H M Korbus* Director, Plant Systems

SUBJECT: Plant Systems-Reclamation Activities at Three Mile Island

REFERENCE: Memo C. A. DeLorenzo to Distribution, subject Information Requested by the President's Commission on Three Mile Island

Warren J. Tyrrell, Plant Systems Supervisor engaged in activities at the subject area from April 16, 1979 through April 27, 1979.

Mr. Tyrrell was assigned to the Waste Management Group under the direction of Mr. Benjamin Rusche.

His assignments were as follows:

1. Inspection of decontaminated areas by Westinghouse and Vychem decon personnel.
2. Reviewed report on protective coatings received by Mr. Rusche from company in New Orleans. Arranged to have company come to TMI for meeting and test demonstration.
3. Consulted with Battelle NW representatives regarding electropolishing as a method of removing radioactive contamination from metal surfaces. As a result of this meeting TMI obtained funding from DOE for the installation of a decon facility estimated to cost 300K.
4. Advised Health Physics as to best method for the decontamination and disinfecting of respirators.
5. Discussed ANL waste disposal procedures.
6. Met with local metal fabricator to prepare estimate for fabrication of the M III bin.
7. Worked with Health Physics regarding survey and smear results.
8. Worked with the Waste Management Technical Support Organization Group - chart attached.

HMK:hpb
Attachment
cc: J. F. Bartusek w/a
R. L. Vree "
306 RF "
PS File "
-- "

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Ext. 251, 252

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WASTE MANAGEMENT
B. Rusche (SC-ERI)

RADWASTE OP ORGANIZATION
J. Seelinger (Met-Ed)

Planning/Scheduling
M. Rininger (Met-Ed)
T. Hillard (Met-Ed)

H. P. Contacts
B. Brannock

Personnel Coordination
B. Gunderson (GPU)
D. Fick (GPU)

Construction Coordination
W. Hirst (GPU)

D. Nagle (UE&C)
S. Dam (B&E)
W. Gunn (GPU)
S. Levin (GPU)
J. Kindzierski (GPU)
T. Uilkema (GPU)
A. Peters (B&R)

Clerical Support
J. Smith (GPU)

Gas
J. McConnell (GPU)

R. Dunn (Gilbert)
J. Robison (Gilbert)
W. Itschner (GPU)
C. Montgomery (GPU)
A. Nawaz (Bechtel)

Liquid
R. L. Williams (GPU)

W. Tyrrell (ANL)
M. Ross (GPU)
S. Kraft (GPU)
D. Garman (Gilbert)
A. Larson (Gilbert)
C. Edwards (Gilberts)
B. Lutz (Gilbert)
P. Arthur (Gilbert)
B. Reinmann (GPU)
B. McCutcheon (GPU)
K. Pastor (GPU)
J. McDonald (GPU)
B. Turpin (Duke)

Clerical Support
M. Babb (Met-Ed)

Technical Support

R. Brooksbank/ORNL Lia.
J. Snider (Liquid) (ORNL)
W. Rodger (NSA)
J. Lieberman (NSA)
L. King (ORNL)
W. A. Shannon (ORNL)
Martin (ORNL)

B. Gunderson
4/19/79

ARGONNE
NATIONAL
LABORATORY

INTRA-LABORATORY MEMO

30 May 1979

TO: File

FROM: B. R. T. Frost, Director, MSD

T. F. Kassner (ANL-MSD) participated in a meeting of nuclear fuel experts, held on April 12, 1979 at NRC-Bethesda, to update estimates of the damage to the TMI-2 core and to consider its effect on the desirability of initiating natural-convection cooling of the core. ANL-MSD has been involved for the past three years in an NRC-sponsored program to investigate the effect of steam oxidation on the mechanical properties and extent of embrittlement of Zircaloy cladding under loss-of-coolant accident conditions. A summary of the above meeting was transmitted to D. Ross, Deputy Director, Division of Project Management, Office of Nuclear Reactor Regulation by W. V. Johnston, Chief, Fuels Behavior Research Branch, Division of Reactor Safety Research, Office of Nuclear Regulatory Research. There have been subsequent telephone calls with NRC-RSR staff, particularly M. Pickelsheimer (Zircaloy expert and program manager for LWR cladding).

BRTF:rw

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ARGONNE
NATIONAL
LABORATORY

INTRA-LABORATORY MEMO

31 May 1979

TO: File

FROM: B. R. T. Frost, Director, MSD

SUBJECT: Assistance to EPRI by MSD Ceramics Group

At the request of Dr. Adrian Roberts (EPRI) Roger Poeppel of MSD conducted a quenching experiment on unirradiated UO_2 pellets - annealed in 100 CO_2 : 1 CO for one hour at 1600°C and quenched into water. Examined pellet for cracking behavior - first time it held together in spite of cracking. Repeat of experiment at least twice and pellet fell apart each time into a few large pieces. Transmitted results to Dr. Roberts on April 27.

Note: Dr. Poeppel has been under contract to EPRI to study cracking behavior of UO_2 for the past two years. The purpose of these quenching experiments was probably to throw light on the likely size of UO_2 particles in the upper section of the TMI-2 core.

ANL did not transmit this information to NRC.

BRTF:rw

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ARGONNE
NATIONAL
LABORATORY

INTRA-LABORATORY MEMO

31 May 1979

TO: C. A. DeLorenzo Director, OOS

FROM: B. R. T. Frost B. Frost. Director, MSD

SUBJECT: Information Relating to TMI

In response to your memo of May 22, I enclose four memos that describe MSD involvements with Three Mile Island. This does not give you the information in the format that you requested but I have only just received these memos and rather than rewriting them to meet your format I am sending them as written.

In addition, Dick Weeks has accompanied John Honekamp and others to California this week to talk to EPRI and Prof. Pigford of the Commission on key questions to be answered and what assistance ANL can provide to these two studies. I suggest that you contact John Honekamp for information when he is at the Laboratory next Monday, June 4.

BRTF:rw
Att.
Distribution:
R. W. Weeks
L. A. Neimark
S Greenberg
J Rest
R. B. Poeppel
T. F. Kassner
J. R. Honekamp

RECEIVED
Office of Operational Safety

[illegible] '79

	INFO	ACTION
CAD	✓	
EAW		
FOP		
HB		
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SK		
JL		
Circ.		

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ANL-20 (11-68)

ARGONNE
NATIONAL
LABORATORY

Office of Operational Safety

INTRA-LABORATORY MEMO

ADE	300	INFO	ACTION
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	EAW	-	-
	FOP	-	—
	HB		
	RGT		
	SK		

May 31, 1979

TO: C. A. DeLorenzo Director, OOS

FROM: R. J. McConnell Associate Director, EBR-II Engineering

SUBJECT: Three-mile Island

REF: Memo to Distribution from C. A. DeLorenzo, "Information Requested by the President's Commission on Three Mile Island," dated May 22, 1979

In response to your reference request, I participated in the following meeting.

Subject

TMI-2 Meeting - Core Damage Assessment, April 5, 1979.

Location

Lynchburg, Va.

Participant

R. J. McConnell, ANL/EBR-II

Purpose

1) Evaluate the information available to determine a best guess of the extent and nature of core damage.

2) Propose future efforts to gain more knowledge.

Other Participants

See attached.

To the best of our knowledge, this represents the only EBR-II participation.

RJM:km

cc: D. W. Cissel

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ARGONNE NATIONAL LABORATORY

INTRA-LABORATORY MEMO

April 12, 1979

To: J. R. Honekamp OTD

From: J. Rest J Rest MSD

Subject: Assistance to the NRC on Matters Related to the Accident at the Three Mile Island Nuclear Reactor

On April 9, Rich Sherry of the NRC called me to request assistance in estimating the fuel temperatures for the uncovered core that occurred as a result of the accident at the Three Mile Island Nuclear Reactor #2 on March 28, 1979. Rich said that George Marino of the NRC had done some calculations to estimate the rate of heating of the uncovered core due to decay heat assuming adiabatic conditions. George's calculations indicated heating rates of from 1-10° F/S. (The next day George called to revise the heating rates down to 1-3° F/S). In addition, measurements had been made (how the measurements were performed and when they were made was not clear) on the amount of released isotopes of xenon. The measurements indicated that 22%, 24%, and 31-38% of the total core inventory of X_e^{135}, X_e^{133m}, and X_e^{133}, respectively, had been released. The NRC requested that I use the GRASS-SST code to perform calculations for assumed accident scenarios to estimate the fuel temperatures required to explain the release of from 20-40% of the total core inventory of fission gas.

Two suggested scenarios were used to perform the calculations. The first scenario consisted of an irradiation at 6 KW/ft (average core power rating) at an average fuel temperature of 1200°F for 60 full power days followed by (1) a relatively instantaneous reduction in power (1.2% of nominal due to reactor scram) and a fuel cooldown to 550°F occurring in about 1 hour's time, and (2) a heatup of the fuel at a heating rate of 1° F/S. The second scenario was similar to the above, but differed in that fuel from approximately 25% of the core which was irradiated for 60 full power days at about 12 KW/ft with an average temperature of 2550°F was considered. The objective of these calculations was to determine at what fuel temperatures GRASS-SST would predict 20-40% total gas release.

The results of the analysis, which were transmitted verbally by me to the NRC (George Marino) on April 11, subsequent to separate discussions I had with Steve Gehl and yourself, are as follows:

1. Predictions made with GRASS-SST for the 6 KW/ft fuel indicate that a maximum of from 5-10% total gas release would occur at fuel temperatures between 4700°F and the fuel melting point (~5160°F). This result was obtained assuming that no extensive grain-boundary separation occurs in the fuel. A substantial amount of gas release from the grains was calculated to occur as a result of the heatup, but this gas was trapped on the grain surfaces and edges, and hence was not released to the exterior

Letter was also sent to Marino describing results

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of the fuel. (The predicted fuel swelling due to the retained gas on the grain surfaces and edges was too small to cause appreciable long-range interlinkage of the porosity).

2. For the 12 KW/ft fuel 20% and 40% total gas release was predicted to occur at fuel temperatures on the order of 4500° and 4800°F, respectively. Again, this result was obtained assuming that extensive grain-boundary separation did not occur.

3. Assuming that extensive grain-boundary separation does occur, GRASS-SST results indicate that for the 6 KW/ft fuel 50% or more total gas release could occur at fuel temperatures on the order of 2400-2700°F. Results for the 12 KW/ft fuel indicate that 50% gas release could occur under these conditions at fuel temperatures on the order of 4500°F. The calculations indicated that the substantially greater fractional release of fission gas predicted to be released from the lower rating fuel than from the higher rating fuel at temperatures of about 2400-2700°F in the event of extensive grain-boundary separation was due to the fact that the lower operating temperatures of the lower rating fuel resulted in much smaller (and hence more mobile) bubbles being generated within its grains.

4. Experience with fuel from the H. B. Robinson Reactor (30,000 MWd/ton burnup compared to the ~1000 MWd/ton burnup of the Three Mile Island fuel) during Direct Electrical Heating (DEH) transient tests at heating rates substantially higher than those calculated to occur at Three Mile Island indicates that ~20% and 40% release occurs in fuel regions where the temperatures reached 2750 and 3650°F, respectively. However, it is expected that the much higher concentration of gas in the H. B. Robinson fuel (very little gas was released during the irradiation) facilitated the formation of the observed grain-surface and grain-edge channels, and this enabled the gas released from the grains to escape to the exterior of the fuel. In addition, fairly extensive grain-boundary separation was observed to occur in this fuel as a result of the transient heating.

On April 12, I called George Marino to ask him if the NRC needed any additional assistance in the interpretation of the above results, which I had transmitted to him on April 11. I again reiterated that these results were based on assumptions about the properties of the fuel (e.g., grain-size), assumptions on the irradiation history of the fuel rods, and on the suggested scenarios of the accident. For example, from the above results it is clear that the irradiation history of the fuel (e.g., 6 KW/ft vs. 12 KW/ft) significantly affects the predicted gas release during the accident.

George told me that the NRC feels that significant grain-boundary separation could have occurred during the accident. Their thinking is based on the results of a pressure transient in the PBF reactor where appreciable grain-boundary separation was observed. In particular, George was referring to a region of the

PBF-tested fuel where almost complete separation (powdering of the fuel) was observed. Analyses indicated that this type of separation (differing by its powdery nature from the type of separations observed in DEH-tested fuel) was the result of the stresses generated as a result of requenching. Based on this hypothesis, the NRC is using in their calculations, the 2700°F fuel temperature predicted to occur in the low rating fuel during the accident under the assumption of the formation of extensive grain-boundary separation. (Result #3, above).

George was reluctant to provide me with any information on what the NRC calculations were to be used for (e.g., the calculation of metal-water reaction rates?), only saying that the work was of an urgent nature.

George and I both agreed that more detailed calculations (e.g., calculations with GRASS-SST and LIFE-LWR) could be useful. I indicated that I would attempt these calculations in the near future.

Note (added later): J. Rest is continuing telephone discussions with NRC-RSR personnel on fission gas release calculations.

ARGONNE
NATIONAL
LABORATORY

INTRA-LABORATORY MEMO

29 May 1979

TO: R. W. Weeks, Associate Director, MSD

FROM: Steve Gehl, Irradiation Performance Group, MSD

RE: Assistance to NRC, etc. regarding TMI

As the following list indicates, most of the assistance I have provided has been indirect, e.g., with Honekamp and Rest. I have included all information for completeness, so that you can do the editing.

1. Week of 4/2. Discussion with J. Honekamp on the likely fission-gas behavior in TMI prior to accident (how much fission gas release before accident?) Provided literature references.

2. Week of 4/9. Several discussions with J. Rest regarding all aspects of fission-gas behavior prior to and during accident. Discussed the implications of GRASS calculations, which were then transmitted to NRC (G. Marino).

3. 4/12. Phone call S. Gehl to G. Marino (NRC). Discussed temperature history during accident; role of fuel microcracking and chemical reactions in fission gas release. George asked that we consider running DEH (Direct Electrical Heating) tests in steam atmosphere.

4. Week of 4/9. Phone call from EPRI, referred by Honekamp. Provided references on fission gas behavior in LWR fuels.

5. 5/10. Internal Memo (Gehl to Neimark) "Core Thermal Conditions and Fission-gas Release in TMI-2" (can be provided if needed).

Note: L. Neimark, S. Gehl and J. Rest have been carrying out a program to study fission gas release from UO_2 under transient conditions for NRC-RSR (G. Marino) for the past three years.

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TMI-3 Meeting Core Damage Assessment 5 April 1979

Attendance List

Name	Organiz.	Subject of Interest/Experience	Telephone
Garry R. Thomas	EPRI	Transient over temp fuel behav.	(415) 855-20
Gilbert Robichaud	B&W	Fuel Marketing	
Harley Wilson	B&W	Fuel Design / core cond Task Force	
Kim Stein	B&W	Fuel Structural Analysis	
Ray King	B&W	Fuel Design / core cond Task Force	
Lew Walton	B&W	FUEL ENG / THERMOCOUPLE DATA	
Mike MONTGOMERY	B&W	FUEL ENGR / Fuel MTL PERF / Core cond TF	
GARY CLEVINGER	B&W	LRC/NUC. MAT. TECH / CORE COND. TASK F	
RICHARD E. PAWEL	ORNL	ZIRC-OXIDATION / O_2 DIFF	615-574-5
David O. HOBSON	ORNL	Zirc-oxidation / embrittlement / tube rupture	615-574
[illegible] J. McCONNELL	[illegible]	CORE DAMAGE	
R. A. Lorenz	ORNL	Fission Product Release	615-574-
Russell Ball	B&W	Instrumentation data	
Ralph Friederichsen	Bettis	Fuel Element Engineering	412-462-5
Charles J. Baroch	B&W	Core Perf Task Force / Fuel Performance	
Reid M Hiatt	B&W	Thermal Hydraulics	
V. O. Uotinen	B&W	Fuel Design	
C. S. Caldwell	B&W	Fuel Condition and F.P. Dist. in coolant/ga	
GEORGE MEYER	B&W	Thermal Hydraulics	
ROBERT J. WALLER	B&W	THERMAL / FLUIDS	
MELVIN F. SANKOVICH	B&W	MGR., FUEL MARKETING	
RICHARD V. DeMARS	B&W	Mgr. Fuel Mech. Design / Core Cond. Task F.	
Art Lowe	B&W	Materials Behavior, CCTF	(804) 384-5
JOHN MAHER	B&W	FUEL PERFORMANCE / RADIOCHEMISTRY	
GORDON BOND	GPU	NUCLEAR ANALYSIS - MGR	(201) 263-480
Jim TULENKO	B&W	Mgr, Fuel Engineering / Fuel Performance	804-38
Robert N. Duncan	CE	Director Fuel Dev. / Fuel Perf - [illegible]	203/
JOE LONGO	C-E	MGR ECCS ANALYSIS	
Peter M. LANG	DOE	Fuel behavior and development	301-353
W. P. CHERNOCK	C-E	V.P. DEVELOPMENT	(203) 688-1911
John C. Comine	KAPL (GE)	PC Radiolysis	(518) 393-6611 x 73
Ed Zebroski	EPRI & IAG	Dir. Systems & MATERIALS	415 855-2061
Ralph O. Meyer	NRC	Reactor Fuels Section (NRR)	301-492-7
W. V. JOHNSTON	NRC	Fuel Behavior Research Branch	301-427-

1915 349

US Department of Energy
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439
Telephone 312/972- 4473

L. BURRIS (CEN)

Filed ....
Noted ....
Copies ....

June 1, 1979

TO: L. Burris — Chemical Engineering

FROM: P. T. Cunningham — Chemical Engineering/ACL

SUBJECT: Assistance to NRC on the T.M.I. Accident

Attached is a report describing our assistance to NRC on the T.M.I. accident, per my conversation with Don Webster on Wednesday.

Paul

P. T. Cunningham

PTC/jes

Attachment

cc: D. Webster
F. Cafasso
R. Meyer
C. E. Johnson
R. Heinrich
R. Larsen (RES)

1915 350

Assistance to NRC on the T.M.I. Accident

Type of Assistance: Review of experimental data resulting from γ counting of a sample of primary coolant water to determine what, if any, inferences might be drawn from the data.

Who Provided: Data was reviewed by R. Larsen, Radiological and Environmental Research Division, and C. E. Johnson, R. Heinrich, and P. Cunningham, Chemical Engineering Division, Argonne National Laboratory.

To Whom Provided: Conclusions were transmitted by telephone to Bud Cherry, Vice President, Corporation Planning, General Public Utilities Corporation.

When Provided: Data received on Saturday, March 31, 1979, and conclusions transmitted by telephone on that afternoon.

Results: The data provided is summarized as follows. A 100-mL sample of primary coolant water was obtained and γ counted to determine the fission products present. Counting results, obtained from Mr. D. Henderson (BAPL) were:

Isotope	Decay Rate, d/m/mL*
^{131}I	3.0×10^{10}
^{132}Te	4.5×10^{8}
^{133}I	1.5×10^{10}
^{134}Cs	1.4×10^{8}
^{136}Cs	3.9×10^{8}
^{137}Cs	6.1×10^{8}
^{140}Ba	4.7×10^{7}
$^{89.90}Sr$	1.2×10^{7}

*All count rates corrected to noon, 3/30/79.

The sample was reportedly reading about 1 R/mL and contained trace levels of U. Other information available included:

- reactor had approximately 60 full power days equivalent of operation;
- reactor core 93×10^3 Kg;
- primary coolant volume 4×10^3 ft^3;
- gas (probably hydrogen) "bubble" approximately 1.2×10^3 ft^3 at 1000 psig.

Based on this data, and other reasonable assumptions or estimations as necessary, it was concluded that:

A. About 3% of the core inventory of Cs was in the coolant (perhaps 25% of the fuel was exposed to coolant).

B. Fission product distribution was not consistent with that expected if fuel vaporization or meltdown had occurred; therefore, fuel probably still intact (at least as pellets). (The concentration of ^{132}Te, which, at the time of the accident, was comparable to that of ^{131}I in the fuel, was about 10^2 less than that of ^{131}I in the water. Tellurium is a relatively volatile fission product. The concentration of ^{140}Ba, which, at the time of the accident, was also comparable to that of ^{131}I in the fuel, was about a factor of 10^3 less than that of ^{131}I in the water. It is expected that ^{140}Ba would be found in the water if fuel melting had occurred.)

C. Hydrogen appeared to be the only reasonable major constituent for the bubble. A water-metal reaction was postulated as the source for hydrogen and to achieve the stated volume would have required involvement of about 25% of the cladding inventory of Zr.

Assistance to NRC on the T.M.I. Accident

Type of Assistance: Data provided on the solubility of hydrogen in water at high pressures and temperatures.

Who Provided: D. S. Webster and M. Blander of Chemical Engineering Division, Argonne National Laboratory.

To Whom Provided: Joseph Murphy of the Nuclear Regulatory Commission (apparently part of a group assembled for the emergency).

When Provided: Initial telephone conversations on Sunday night, April 1. Data sought on Monday, calculations made and results telephoned on Tuesday (4/3).

Results: At the time of this inquiry by NRC, there was presumed to be a 1000 cu. ft. "bubble" of hydrogen at the top of the reactor, in solubility-equilibrium with recirculating water. In order to dissipate the bubble, it was believed possible to draw off 20 gpm of water (through a cooler, let-down orifices, filters) and spray it into the makeup storage tank at about 1 atmosphere pressure. Dissolved hydrogen was expected to come out of solution, whereupon it would be discharged to the containment; the water would be pumped back into the reactor primary system. The questions were: What is the solubility of hydrogen at 1050 psia in water at 280°F, and how long will the H_2 removal take?

The most probable value for the solubility of H_2 at 1050 psia in water at 280°F was judged to be 1600 ml H_2 (S.T.P.)/kg water (Reactor Handbook, 2nd Ed., Vol. 1, p. 851, Interscience Publishers, N.Y.C.). Furthermore, as the water temperature fell during passage through the cooler, the solubility would fall to about 1300 ml H_2 (S.T.P.)/kg water at 120°F.

At the 1600 ml/kg value, 20 gpm of water (75 kg/min) would carry 120 liters H_2 (S.T.P.)/min, or 0.09 cu. ft./min at 1050 psia and 280°F, to the disengagement vessel. If it is assumed that the returned water saturates with hydrogen again in the reactor, the 20 gpm purge will have to continue for about a week ($\frac{1000 \text{ cu. ft.}}{0.09 \text{ cu. ft./min}}$ = 11000 min) to remove the 1000 cu. ft. bubble.

After the bubble is gone, purging should continue in order to remove most of the large amount of H_2 in solution. The relationship for this falling-concentration period is $t = \frac{M}{q} \ln \frac{Co}{c}$, where M is the mass of water in the primary system (taken as 230,000 kg), q is the pumping rate (75 kg/min), and Co/c is the ratio of initial H_2 concentration to the desired final concentration (taken as 1600/10). The time for this phase of purging is about 11 days.

If, as suggested by NRC, only half the H_2 disengages during spraying (?) both times will be doubled.

1915 353

Mr Crawford

7

Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

MAY 25 1979

R. L. Ferguson, Program Director
Office of Nuclear Energy Programs, HQ

INFORMATION REQUESTED BY THE PRESIDENT'S COMMISSION ON THREE MILE ISLAND

As requested by your TWX to A. G. Fremling, Manager of RL, on May 15, 1979, subject as above, information on RL or RL contractor assistance to the NRC, State of Pennsylvania, or to the utility company following the Three Mile Island Accident is provided in the enclosure to this memorandum. While RL and RL contractors have received numerous contacts from DOE, NRC and the utility company, to discuss Hanford capabilities, the only actual assistance provided this far is as noted in the enclosure. I have established a single point of contact at RL (J. D. White) to coordinate any new or continuing activities related to the Three Mile Island accident and we have advised our prime contractors of this action.

F. R. Standerfer

F. R. Standerfer, Assistant Manager
for Technical Operations

NFCP:JDW

Enclosure

cc: H. Feinroth, ET-HQ,
w/encl.

1915 354

ASSISTANCE PROVIDED BY RL AND RL CONTRACTORS TO THE NRC, STATE OF PENNSYLVANIA, OR THE UTILITY COMPANY FOLLOWING THE ACCIDENT AT THREE MILE ISLAND (TMI)

Provision of Air Distribution Manifolds

1. Two breathing air distribution manifolds, plus extra filters and spare parts, were provided to Metropolitan Edison Company on April 17, 1979. The estimated value of the equipment was $1000. The utility company agreed to reimburse DOE for the cost.

2. The manifolds were requested on the afternoon of April 17, 1979, and were shipped air freight, by 10:00 p.m. on April 17, 1979.

3. Personnel and organizational contacts were as follows:

   - On April 17, 1979, R. L. Ferguson called R. P. Fasulo, (RL) and requested that we provide fresh air breathing equipment to TMI. J. L. Tew was the DOE contact; he put us in direct contact with Joe Barrett of Metropolitan Edison Company.

   - The Metropolitan Edison Company's request was for standard MSA multiport breathing air supply manifolds.

   - Two manifolds were provided by Hanford Engineering Development Laboratory (HEDL). W. C. Cravens, Nuclear Facilities Administrator, Chemical Engineering Department, was the shipment coordinator.

   - The manifolds were shipped to Metropolitan Edison Company at TMI.

4. This activity was completed on April 17, 1979.

Reactor Physics Calculations

1. Battelle's Pacific Northwest Laboratory (PNL), under existing Request for Services agreement with NRC, performed physics calculations for NRC following the accident at TMI. Calculations were performed to estimate the amount of core blockage which had occurred, using temperature data provided by NRC.

2. The calculations were requested by Sol Levine, Director of the Office of Regulatory Research, NRC, through the home office of Battelle Memorial Institute. PNL contact was Dr. D. S. Trent, Manager of the Fluid and Thermal Engineering Section.

2

3. Calculational results were provided to NRC at TMI. NRC should be contacted for information on results.

4. Work has been completed, except for final documentation, which is currently being prepared for transmittal to NRC.

1915 356

DOE Form AD-10A
(12-77)

U.S. DEPARTMENT OF ENERGY

memorandum

DATE: MAY 23 1979

REPLY TO ATTN OF: ET-930

1979 MAY 25 PM 1 09

SUBJECT: Analysis of Three-Mile Island Charcoal Filter Samples

TO: John M. Deutch, Acting Assistant Secretary
for Energy Technology

NRC recently requested the assistance of Dr. V. Dietz, Naval Research Laboratory, in analyzing samples of charcoal filter media from Three-Mile Island. This limited effort, short-term work relates to work Dr. Dietz has done under an ETW/P supported contract. The necessary arrangements were made and the analyses are underway (no impact on ETW/P program). We will receive a copy of the NRL report prepared for NRC.

The attached memorandum to me from Goetz Oertel provides additional information.

We are notifying you of this because of your interest in DOE support efforts related to Three-Mile Island.

Original signed by
SHELDON MEYERS
Sheldon Meyers, Program Director
Office of Nuclear Waste Management

Attachment

cc: B. Ferguson, Dir., ETN, w/attach.

1915 357

STNI 4

DOE Form AD-10A
(12-77)

U.S. DEPARTMENT OF ENERGY

memorandum

DATE: MAY 17 1979

REPLY TO
ATTN OF: ET-952

SUBJECT: ETW/NRC Cooperation on Three-Mile Island Iodine Release Measurements

TO: Sheldon Meyers, Program Director
Office of Nuclear Waste Management

Informal discussions with NRC personnel at Three-Mile Island (TMI) have determined that much of the iodine leakage during and after the accident was due to faulty charcoal filter media (KI_x impregnated activated charcoal). It passes the current NRC efficiency test satisfactorily, but the organic iodine bleeds off slowly after the initial capture, a phenomenon which the test does not measure.

This phenomenon was revealed in an ETW/P-supported paper presented by Dr. V. Deitz (NRL) at the August 1978 "15th DOE Nuclear Air Cleaning Conference." The responsible NRC staff participated in this Conference. NRC recently asked us to permit Dr. Deitz to carry out iodine "bleed off" tests on samples of the TMI charcoal. Results to date confirm previous results, i.e., they do bleed off as much as 70-80% of the organic iodine initially absorbed.

Backup beds impregnated with TEDA (a proprietary amine compound patented in England), now in place at TMI, are retaining the iodine satisfactorily.

It is our understanding that most power reactors now in service also utilize the KI_x charcoal.

DOE has used the TEDA treated media at SR for several years. About three years ago our legal department received a request from Suttcliffe-Speakman (a UK charcoal company) indicating they were the sole licensee of Harwell and requesting an exorbitant license fee. At that time, we embarked on a program -- also with Dr. Deitz -- to find a competitive product. Another amine compound HMTA proved adequate and it cost less, as well. This new product (a DOE patent is in place) may now find wide-scale use.

Goetz K. Oertel

Goetz K. Oertel, Director
Division of Waste Products

cc: John Whitsett, ID

1915 358

Titles of TMI related Activities at
Los Alamos Scientific Laboratory (LASL) and Sandia

Los Alamos:

"Resperative Studies for NRC" (NRC contract #A7005)

"TRAC Code Applications" (NRC contract #A7049)

"Fuel Pin Transient Behavior Modeling and Analysis" (NRC contract #A7046)

"Accident Delineation" (NRC contract #A7014)

"Source Terms for Decay Heat Calculation" (DOE)

Sandia:

"Accident Energetics" (NRC contract #A1016)

"Systems Interaction Methodology Applications" (NRC contract #A1113)

"Physical Protection for Nuclear Facilities" (NRC contract #A1060)

"Development and Analysis for Vent Filtered Containment Conceptual Design" (NRC contract #A1220)

1915 359

③ Mr Crawford

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MESSAGE CONTAINS WEAPON DATA
☐ YES ☒ NO

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TYPE OF MESSAGE: ☒ Multiple Address

FROM
DEPARTMENT OF ENERGY
DAYTON AREA OFFICE
HARRY N HILL, AREA MANAGER
MIAMISBURG, OHIO

OFFICIAL BUSINESS: Harry N. Hill (Signature of certifying official)

TO ROBERT L. FERGUSON, USDOE, GERMANTOWN, MD.
INFO CY: J. R. ROEDER, USDOE, ALBUQUERQUE, N.M.

UNCLASSIFIED/N O N W D

INFO REQUESTED BY THE PRESIDENT'S COMMISSION ON THREE MILE ISLAND. THE FOLLOWING MOUND FACILITY INFORMATION IS PROVIDED IN RESPONSE TO YOUR TWX OF 5/15/79:

1. A LIST OF EACH ACTIVITY OR TYPE OF ASSISTANCE PROVIDED:
   a. A RADIOLOGICAL ASSISTANCE TEAM CONSISTING OF SIX MRC (MONSANTO RESEARCH CORP.) PEOPLE AND EQUIPMENT WAS DISPATCHED TO ASSIST AND COMPLEMENT THE RADIOLOGICAL ASSISTANCE TEAM FROM THE DOE CHICAGO OPERATIONS OFFC.
   b. THE MOUND EMERGENCY RADIO SYSTEM (ERS) WAS MANNED TO PROVIDE PHONE PATCH CAPABILITY FOR EG&G PERSONNEL.
2. DATES AND NAMES OF INDIVIDUALS AND ORGANIZATIONS PROVIDING THE ASSISTANCE:
   a. RADIOLOGICAL ASSISTANCE TEAM
      DATE: MARCH 30, 1979 THROUGH APRIL 3, 1979
      NAMES: CLYDE W. TAYLOR, GERALD L. PHILLABAUM, PHILLIP J. RHUDE, WALTER R. WALLACE, JAMES M. ANDERSON

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TEST TARGET (MT-3)

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MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION TEST TARGET (MT-3)

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IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

AND LOWELL C. HOPKINS. ALL EMPLOYED BY NRC.

b. ERS

DATE: MARCH 30 AND 31, 1979

NAME: CARL T. SHAWHAN, NRC

POOR ORIGINAL

3. THE RESULTS OF ANY WORK DONE, INCLUDING RESULTS OF ANALYSES, DATA FROM PHYSICAL SAMPLES, ETC.

   a. THE MOUND RADIOLOGICAL ASSISTANCE TEAM PLACED DOSIMETER BADGES IN A TWENTY-MILE RADIUS AROUND THE THREE MILE ISLAND REACTOR FOR THE PURPOSE OF ASSESSING POTENTIAL EXPOSURE LEVELS. THE DOSIMETERS WERE NOT COLLECTED, PROCESSED OR ANALYZED BY MOUND PERSONNEL.

   b. THE MOUND ERS WAS ON STANDBY AND WAS NOT ACTUALLY USED BY EG&G PERSONNEL.

4. INFORMATION ON THE ORGANIZATION AND INDIVIDUALS TO WHOM THE RESULTS WERE COMMUNICATED:

   a. THE MOUND RADIOLOGICAL ASSISTANCE TEAM SERVED AS A COMPLEMENT TO THE DOE CHICAGO OPERATIONS OFFICE TEAM AND REPORTED ALL ACTIVITIES AND RESULTS TO EDWARD J. JASCEWSKY AND PAUL NEESON OF DOE/CH. THE MOUND TEAM WAS ASSIGNED TO ASSIST THE U. S. FOOD & DRUG ADMINISTRATION (FDA) IN PLACEMENT OF DOSIMETER BADGES IN A 20-MILE RADIUS AROUND THE REACTOR AND REPORTED ON THIS ACTIVITY TO RICHARD GROSS OF THE FDA.

5. INFORMATION ON ANY ASSISTANCE THAT IS STILL CONTINUING: MOUND PERSONNEL ARE NOT CURRENTLY PROVIDING ASSISTANCE TO ANY ORGANIZATION CONCERNED WITH THE THREE MILE ISLAND INCIDENT.

THE REQUESTS FOR MOUND RADIOLOGICAL ASSISTANCE TEAM PERSONNEL CAME FROM E. J. JASCEWSKY OF CH, AND THE REQUEST FOR MANNING THE ERS CAME FROM THE EOC AT HEADQUARTERS. THEREFORE, BOTH ACTIONS ARE PROBABLY BEING REPORTED BY THE RACT. HOWEVER, ACCORDING TO A TELECON WITH R. F. INROTH, HE PREFERS TO HAVE THIS INFORMATION AVAILABLE, TO ASSURE A COMPLETE REPORT.

(8) Mr Crawford

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FROM

J. B. Knox
LLL

7. OFFICIAL BUSINESS
Joseph B. Knox
(Signature of certifying official)

(TIME) A.M. P.M.

8. DATE 5/23/79

TO

Robert L. Ferguson 3465
U.S. Dept. of Energy X-200 24/0000
Germantown, MD

INFO J. LaGrone
U.S. Dept. of Energy/SAN X-200 - 24/0048

M. Wu
U.S. Dept. of Energy/SAN

R. Becker/LLL (L-20)

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REFERENCE TWX 1351741 DTG 151741Z MAY 79, FOLLOWING INFORMATION CONCERNS ASSISTANCE PROVIDED BY LLL IN RESPONSE TO THE THREE MILE ISLAND REACTOR ACCIDENT.

1. ACTIVITY OR TYPE OF ASSISTANCE PROVIDED - THE ATMOSPHERIC RELEASE ADVISORY CAPABILITY (ARAC) WAS ACTIVATED DURING THE PERIOD 28 MARCH, 1979 THROUGH 18 APRIL 1979 FOR THE PURPOSE OF PROVIDING REAL-TIME ASSESSMENT ADVISORIES CONCERNING THE ATMOSPHERIC RELEASES OF RADIONUCLIDES.

2. DATE AND NAMES OF INDIVIDUALS AND ORGANIZATIONS PROVIDING THE ASSISTANCE.

INDIVIDUAL, ORGANIZATION		MAR 28-31 (DAYS)	APR 1-18 (DAYS)
DICKERSON, M.H., PHYSICS DEPT.		3	7
ELLIS, J.S.	"	2	13
ELLSAESSER, H.W.	"	-	4
GREENLY, G.D.	"	3	12
GUDIKSEN, P.H.	"	1	12
KNOX, J.B.	"	1.5	6
LANGE, R.	"	3	13
MACCRACKEN, M.D.	"	2	4
PETERSON, K.R.	"	3	13
RODRIGUEZ, D.J.	"	3	13
SHERMAN, C.A.	"	3	13
SULLIVAN, T.J.	"	3	12
TUERPE, D.R.	"	2	12
VEITH, C.R.	"	1	13

1916 003

INDIVIDUAL, ORGANIZATION		MAR 28-31 (DAYS)	APR 1-18 (DAYS)
WALTON, J.J.	"	1	3
WEICHEL, F.L.	"	-	2
WEICHEL, R.L.	"	2	9
GUNN, B.S., COMPUTATIONS DEPT.		-	3
HILL, K.L.	"	-	13
LAWSON, L.A.	"	3	12
LICUANAN, A.R.	"	-	4
WALKER, H.	"	2	12
ALTON, M.S., ELECTRICAL ENGR. DEPT.		3	13
CASSARO, E.S.	"	2	13
FAULKNER, D.F.	"	2	13
GREEN, T.J.	"	2	13
LANVER, B.S.	"	2	12
OBERMAN, A.Z.	"	-	1
BARBER, L.M., HAZARDS CONTROL DEPT.		-	1
GIBSON, T.A.	"	-	6
HANKINS, D.E.	"	-	1
MYERS, D.S.	"	-	2
POWELL, T.	"	-	2
SMITH, T.H., MECHANICAL ENGR. DEPT.		-	5

1916 004

3. DURING THE INCIDENT, THE REAL-TIME ARAC CONCENTRATION CALCULATIONS WERE USED TO (A) DEPLOY GROUND SAMPLING TEAMS, (B) ESTIMATE THE SOURCE TERM (IN CONJUNCTION WITH THE HELICOPTER MEASUREMENTS), (C) PROVIDE ESTIMATES OF INTEGRATED EXPOSURE AND (D) PROVIDE THE FAA WITH

INFORMATION BEARING ON POSSIBLE CHANGES IN FLIGHT CORRIDORS NEAR THE HARRISBURG AIRPORT.

4. RESULTS OF THE ARAC CALCULATIONS WERE SENT TO (A) DOE COMMAND POST, (B) DOE EMERGENCY OPERATIONS CENTER (EOC), (C) NRC EOC AND NRC KING OF PRUSSIA, PENN.

5. NO FURTHER ACTIVITIES AT THIS TIME.

B-107 (5)

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FM USDOE MAHLON E GATES MGR NVOO LAS VEGAS NV

TO RHEGGTN/USDOE ROBERT L FERGUSON OFF OF NUC ENERGY PROGS GTN (ET)

INFO RHEGGTN/USDOE R SHULL MA HQ GTN (DP-292)

AE

BT

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SUBJ. THREE MILE ISLAND (TMI) SUPPORT. THE FOLLOWING INFORMATION IS PROVIDED IN RESPONSE TO YOUR TWX OF MAY 15, SAME SUBJECT. ALL ASSISTANCE PROVIDED IN SUPPORT OF THE TMI ACCIDENT BY NV AND ITS CONTRACTORS WAS IN DIRECT SUPPORT OF THE EMERGENCY ACTION COORDINATION TEAM (EACT). IT IS NV'S UNDERSTANDING THAT DETAILED INFORMATION ON THIS SUBJECT HAS BEEN ASSEMBLED BY EACT.

END MSG. NSD:EWA-404

BT

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