7

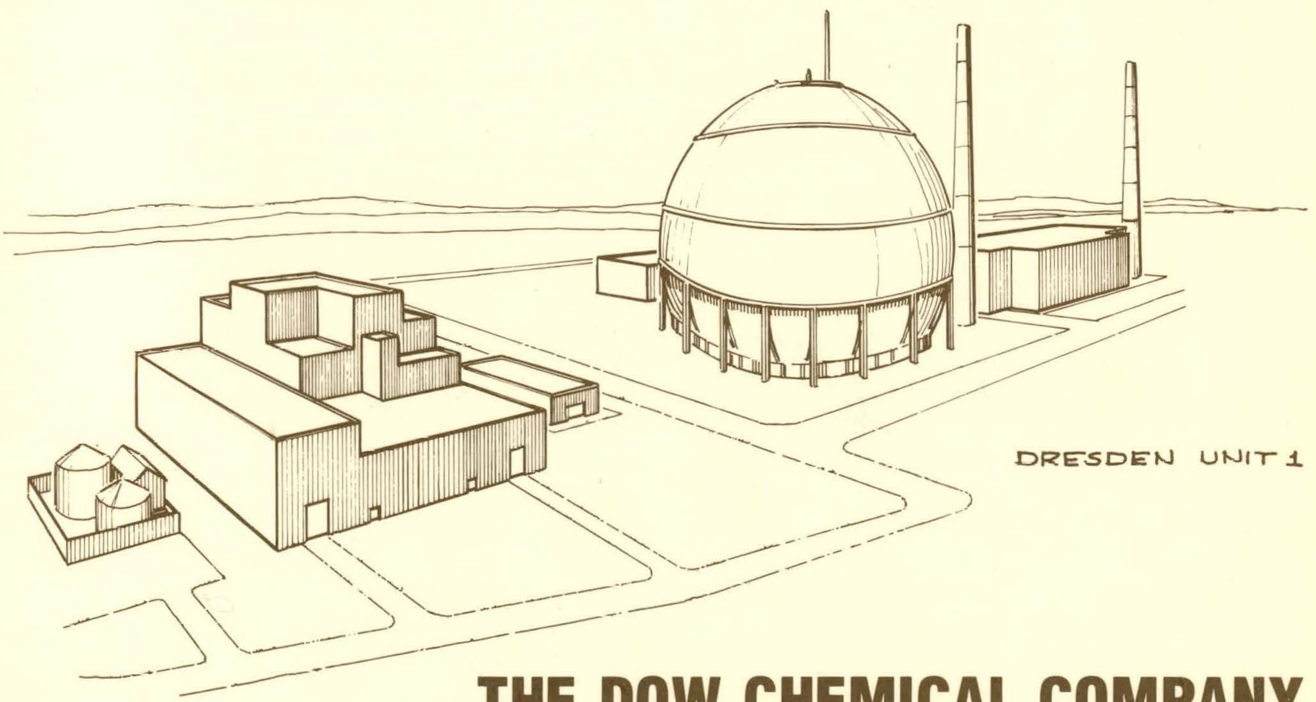
DOW

NUCLEAR SERVICES

TECHNICAL STUDY FOR
THE CHEMICAL CLEANING OF DRESDEN-1
DNS-D1-016

MASTER

VOLUME VII
APPENDICES IX THRU XIV



THE DOW CHEMICAL COMPANY

MIDLAND, MICHIGAN 48640

TECHNICAL STUDY FOR
THE CHEMICAL CLEANING OF DRESDEN-1
DNS-D1-016

VOLUME VII
APPENDICES IX THRU XIV

JUNE 15, 1977

Prepared by:
DOW NUCLEAR SERVICES
THE DOW CHEMICAL COMPANY
MIDLAND, MICHIGAN

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VOLUME VII

APPENDIX IX

THE DECONTAMINATION OF DRESDEN-1

WARREN I. KIEDAISCH
GENERAL STAFF ENGINEER
COMMONWEALTH EDISON COMPANY
CHICAGO, ILLINOIS

Introduction:

The power industry has recognized for many years that chemical cleaning of boilers and other power plant equipment is a necessary part of prudent plant management. We have cleaned boilers to remove deposits which cause loss of heat transfer and result in tube failures. We have cleaned heaters to regain heat transfer efficiencies. We have never, to my knowledge, cleaned power plant equipment primarily to permit access to the equipment for inspection or maintenance. We are now facing the reality that in nuclear plants chemical cleaning is necessary to permit access to equipment for operation, maintenance and inspection at minimum radiation exposure.

The Problem:

We have been watching the rise in man-rem of exposure which has been experienced at our nuclear stations from year to year. Figure 1 shows this trend for Dresden Station.(1) Dresden started operating as a single unit station in 1960. Dresden 2 was added in 1970 and Dresden 3 in 1971. The rise in man-rem exposure, for the years prior to 1970, is a product of increasing dose rates for routine operation, maintenance and inspection work and an increased "in service" inspection program on the primary system. The years 1971 through 1974 show a steeper rate of rise due to the operation, maintenance and inspection of units 2 and 3.

The trend of increased man-rem of radiation exposure per year of operation is not peculiar to Dresden Unit 1. This same trend is being experienced on all light water reactors (both BWR's and PWR's). The information on Figure 2 was extracted from WASH 1311.(2) This upward trend will continue unless some positive actions are taken.

Causes:

The primary cause of the increased man-rem exposure with time is the increased dose rate in the areas where work is performed. This increased area dose rate during shutdown is due to activation products (mainly cobalt 58 and 60) plated out in equipment and piping. The deposit containing the activation product is in the form of a thin film on the interior of piping and equipment.

Solution:

The best solution to any problem is to eliminate the cause of the problem. Ideally, we should keep the thin film of activated corrosion products from forming. The techniques for accomplishing this have not been developed. The next best solution is to remove the film after it forms. The techniques for removing deposits from metal surfaces have a history of success, and chemical cleaning the reactor system, in our opinion, is a good alternative for reducing the dose rates.

Evaluation:

Once the decision is made that chemical cleaning is the technique preferred to reduce the radiation dose rate, many new problems appear.

1. Is there a solvent which will effectively remove the deposits?

2. Will the solvent degrade the components of the system and cause future problems in the operating life of the reactor?

3. How much will it cost?

In order to answer these questions, we entered into a contract with Dow Industrial Services to study the feasibility of a chemical decontamination of Dresden 1.

In addition, several consultants are evaluating the Dow Industrial Service program to suggest areas of concern where they feel additional study is necessary. These consultants are General Electric under Dale Bridenbaugh, Dr. Roger Staehle from Ohio State University, Craig Cheng of Argonne and Tom Hendricksen of Burns and Roe. Later in this program you will hear of the progress in some of these studies.

Benefit:

We feel that the technique of chemical cleaning of nuclear reactors is one which the power industry requires, just as the techniques for cleaning boilers in fossil stations had to be developed many years ago.

I can still remember what a cool reception the acid cleaning of boilers received thirty or more years ago. Today chemical cleaning of boilers is a commonly used tool enabling us to keep boilers with high heat inputs "on the line", and reduce costs from outages due to tube failures because of deposit build-up.

The same barriers must be faced again in the nuclear age. Chemical cleaning of an all stainless steel system is not something one does without adequate preparations. There is a well warranted concern that stress corrosion cracking of stainless steel can be accelerated by some chemicals. We must establish a cleaning system that can do the job without short or long term damage to the reactor or piping. Once this research work is completed, the final test is to clean a reactor and put it back into service. This is what we propose to do with Dresden Unit 1. We intend to monitor the performance of this reactor very closely by nondestructive testing after various periods of operation. We will have numerous metallurgical specimens placed in the reactor during cleaning.

Some of these specimens will be removed and tested immediately after the cleaning, others will remain in the reactor during operation and be removed for testing during subsequent refueling outages. It is our intent that specimens representing materials of construction for newer reactors as well as those in Dresden 1 be a part of this program. It is our desire to maximize the amount of information to be gained from chemical cleaning a nuclear reactor.

Conclusions:

In our opinion, the power industry needs to develop a method for chemical cleaning of reactors to reduce the worker radiation exposure which are the result of high radiation fields due to activation products deposited on the surfaces of the piping and equipment in the primary system. In addition to the reduction in radiation exposure to the workers, such a procedure will also result in greatly reducing the cost of maintenance and inspection of the systems.

While the cost of the first chemical cleaning will be high, future cleanings can be expected to be lower in cost as the techniques are developed and improved.

A thorough knowledge of the materials of construction and the piping arrangement of the system to be cleaned is of utmost importance, a careful screening of behavior of the solvent on those materials is necessary. A comprehensive and detailed cleaning procedure can then be made. A follow-up program of the metallurgical performance of the reactor after the reactor has been cleaned and put back into service is definitely required.

If all of these considerations are carefully evaluated, chemical cleaning of a nuclear reactor should be possible. The development of a chemical cleaning procedure for reactors will benefit the entire nuclear power industry.

References:

[1] J. C. Golden and R. A. Pavlick "A Review of Effluents, General Population Doses, and Occupational Doses Resulting from Commonwealth Edison's Use of Nuclear Power" dated August, 1974

[2] Thomas D. Murphy "A Compilation of Occupational Radiation Exposure from Light Water Cooled Nuclear Power Plants 1969 - 1973" WASH 1311 dated May, 1974

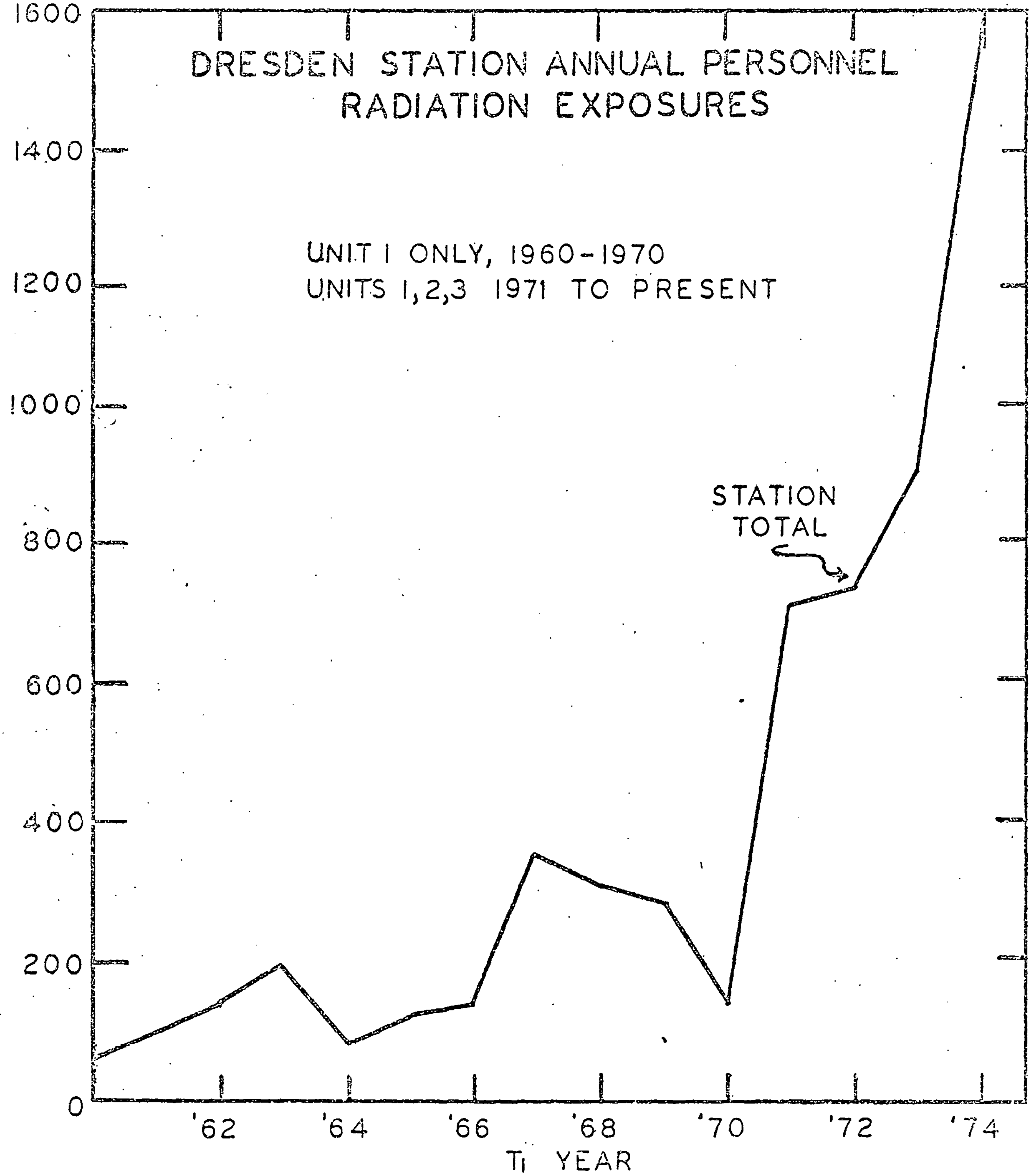
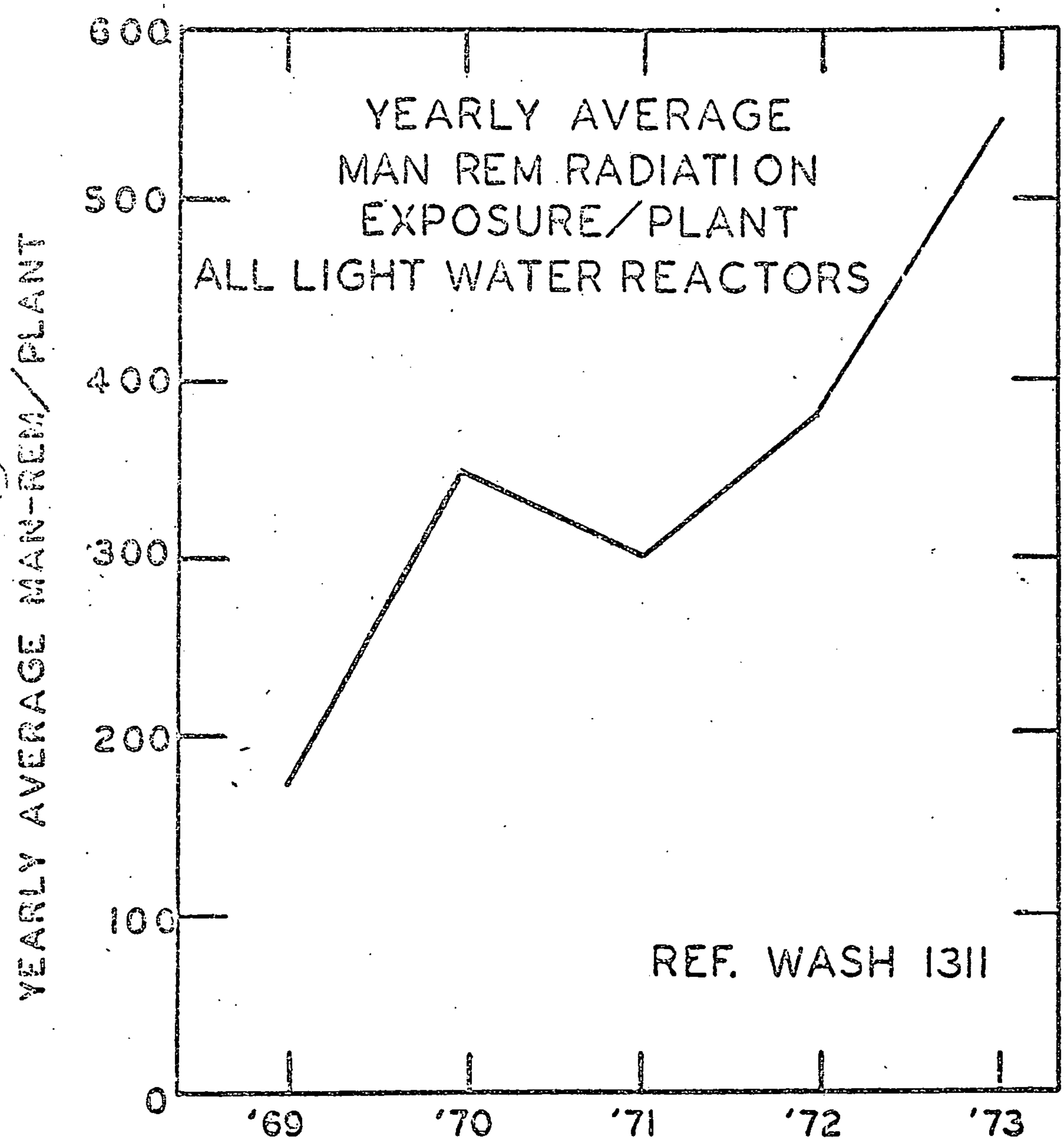

DRESDEN STATION ANNUAL PERSONNEL
RADIATION EXPOSURES
UNIT 1 ONLY, 1960-1970
UNITS 1,2,3 1971 TO PRESENT
STATION
TOTAL
1600
1400
1200
1000
800
600
400
200
0
'62
'64
'66
'68
'70
'72
'74
YEAR

FIG. 1



600
500
400
300
200
100
0
YEARLY AVERAGE MAN-REM/PLANT
YEARLY AVERAGE
MAN REM RADIATION
EXPOSURE/PLANT
ALL LIGHT WATER REACTORS
REF. WASH 1311
'69
'70
'71
'72
'73

FIG. 2

APPENDIX X

CONSULTANT'S OPINIONS

[illegible] TT/TEA [illegible]
[illegible] 18, 1974

Mr. W. P. Worden
Commonwealth Edison Company
One First National Plaza
PO Box 767
Chicago, Illinois 60690

Dear Bill:

Subject: DRESDEN 1 RADIATION LEVEL REDUCTION PROGRAM -- SUMMARY EVALUATION OF WORK STATUS

The purpose of this report is to present the status of work being performed by the Nuclear Energy Division to support Commonwealth Edison Company in the subject program.

Presently most of our effort has been in the area of reviewing and evaluating the chemical decontamination program proposed by Dow Industrial Service. This effort has been aimed at determining the following:

1. What is the effectiveness of the proposed cleaning solvent, i.e., decontamination factor--will it reprecipitate?
2. What is the effect of the cleaning solvent on the materials in the reactor plant? Will it cause corrosion, cracking, etc.?
3. What is the effect on subsequent service?
4. What will the proposed cleaning procedure involve, i.e., plant preparation, monitoring clean-up, safety precautions, etc.?

In order to make the above determination it was necessary for two General Electric people to visit with Dow Industrial Service (DIS) personnel at Midland, Michigan in order to obtain background information and Dow's overall approach to the program. Here we obtained an overview of the program, a brief review of the status of material studies, and various tests performed by DIS Midland. We also obtained most of the Progress Report prepared by DIS, thus providing a more detailed review of the DIS proposed program.

We also visited Dow Chemical Freeport, Texas Contract Research personnel and were given a similar review of their corrosion studies.

The present status of our review and evaluation is as follows:

Effectiveness of Cleaning Solvent

The specimen data viewed by GE personnel indicated that Dowcon-1, the proposed cleaning solvent, was effective as a cleaning solvent on the specimens viewed. The most significant of these specimens were the hand hole plates taken from two secondary steam generators. The fact that DIS indicates that the deposits found on these hand hole plates are the same as those found on pipe samples taken from the Dresden 1 plant over a year ago adds credibility to Dowcon-1 effectiveness as a cleaner. Due to the constraints on GE personnel our investigation of Dowcon-1 has been mostly after-the-fact. It was necessary to rely on written reports and viewing of specimens previously tested. However, based on this type of information it appears that Dowcon-1 is:

1. Effective as a cleaner on the type of scale in the Dresden 1 plant.
2. It dissolves the deposits rather than causing them to flake off.
3. It does not promote reprecipitation.

Effect of Solvent on Reactor Plant Materials

We were impressed with the amount of work DIS has done in this area. We were particularly impressed with the number and different alloys of material tested.

After reviewing the corrosion test data generated by DIS and examining many of the test specimens shown to us by DIS, we have not observed any gross gap in their corrosion studies. However, there are at present three additional tests that GE desires to have performed on 304SS. These three tests are described below:

1. IGSCC Tests under Simulated Cleaning Conditions

These tests would consist of exposure of simple rectangular bent beam samples - roughly 3" x 1/2" x 1/16" - strained 1%-2% by bending over a radius block, with both ends anchored to maintain residual stresses. The exposure conditions would be the time and temperature anticipated for the cleaning operation in fresh Dowcon-1, and in Dowcon-1 containing Fe^{+3} and Ni^{+2}. Data would be obtained by optical and metallographic examinations for IGSCC. These tests could easily be performed in Dow facilities.

2. IGSCC Tests in Residual Dowcon-1 at 550°F

These tests would consist of exposure of constant load uniaxial tensile samples to refreshed air-saturated Dowcon-1 containing Fe^{+3} and Ni^{+2} at 550°F. If self-loaded samples (limited unload following capability) were used, the tests could probably be performed in Dow facilities. If dynamically loaded samples (unlimited unload following capability) were used, the tests would probably have to be performed in General Electric-NED facilities. Dynamically loaded samples would be preferable because of their unlimited unload following capability and the ability to measure time-to-failure directly. Data would be obtained from self-loaded samples by optical and metallographic examination, and from dynamically loaded samples by direct readout of time-to-failure.

3. IGSCC Tests in Subsequent Service Environment

These tests would consist of exposure of unstressed samples to Dowcon-1 containing Fe^{+3} and Ni^{+2} for the time and temperature anticipated for the cleaning operation. The samples would then be rinsed clean with water and exposed to oxygenated 550°F water as both bent beam and constant load uniaxial tensile specimens. Data would be obtained from a statistical analysis of the IGSCC behavior of the samples exposed to Dowcon-1 in comparison to control samples which had not seen the prior exposure to Dow-Con 1. These tests would most probably have to be performed in General Electric-NED facilities.

Effect On Subsequent Service

The effect of the cleaning solvent on reactor materials during subsequent plant operation is, of course, our most important concern. It is also the area in which we have the least physical data. None of the test data or specimens observed to date suggest a serious problem in this area. However, many of the tests performed to date were not necessarily designed to provide long-term post cleaning service data. The additional tests described in the previous section should provide a significant data point for subsequent service.

Proposed Cleaning Procedures

The work in this area is just being initiated since the order of priority placed the work previously discussed ahead of this work. We have just recently received a marked-up P&ID and a preliminary Decontamination Facility Description. We will be reviewing this material along with the cleaning procedures as they are developed by DIS.

The above discussions represent our preliminary observation of the proposed chemical cleaning of Dresden 1. There of course will be additional in-house review and evaluation of existing data and new data as it is obtained. In addition, we expect to start reviewing the licensing and safety implications and requirements in the near future.

Very truly yours,

GENERAL ELECTRIC COMPANY

T. E. Adams, Manager
Performance Improvement Engineering
MC 140, X-6421

ROGER W. STAEHLE, Ph.D.
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GAHANNA, OHIO 43230
TEL: 614-471-6402

ADDRESS REPLY TO

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Dept. of Metallurgical Eng
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Columbus, Ohio 43210

25 October 1974

Mr. W. Worden
Commonwealth Edison Co.
One First National Plaza
P. O. Box 767
Chicago, Illinois 60690

Subject: Evaluation of Studies by Dow to Qualify Decontamination of Dresden 1

Dear Bill:

You requested my evaluation prior to the 28 October meeting in Chicago concerning the state of the decontamination studies in support of Dresden 1.

This letter summarizes my overall evaluation of the program and identifies specific areas in which uncertainties may exist. The comments in this letter are based upon my evaluation of the Dow Program through their August report, accessibility to the composition of the Dow decontaminating solution, and preliminary work in our own laboratory.

In summary, the Dow decontaminating solution appears to cause no catastrophic effects which would imply that the program would be unsuccessful or substantially altered. They have exposed a wide range of alloys, heat treatments, and geometrical circumstances to their nominal solution composition as well as to various modifications of that solution. At present the most serious concern relates to the stress corrosion cracking of heavily sensitized type 304 stainless steel. However, the intensity of the cracking is not significantly different from that which occurs in pure water.

My review of the composition of the Dow solution suggests that there does not appear to be any inherently catastrophic nature of possible interactions between this solution and key materials in Dresden 1.

Despite the generally satisfactory flavor which has arisen from the above, there are a number of important specific items which need very careful consideration and which could be limiting if negative results were obtained from appropriate experiments. These specific items are outlined below:

1. Sensitized Stainless Steel

The experimental work has shown that type 304 stainless steel in the sensitized condition is subject to stress corrosion cracking in the decontaminating solutions under circumstances involving the following:

a. sensitization at times of 50 hours or greater
b. the presence of heat treating scale
c. accentuation in the double crevice condition
d. Ni^{++} ions
e. Fe^{+3} ions
f. Cr^{+6} ions

Experiments also show generally that the susceptibility in variations of the decontaminating solutions do not appear to be more aggressive than pure water.

Since this area is an extremely crucial one, the following very specific questions should be answered:

a. A precise definition of the state of sensitization of welding and normal heat treating of the Dresden 1 should be established for comparison with experimental conditions.
b. Welding sensitization should be evaluated more carefully.
c. The possible existence of crevices which incorporate sensitized regions should be more carefully examined, especially with respect operation after the decontamination operation.
d. The possible interaction of various impurities in the solution (e.g. F^-, Cl^-, Pb) with oxygenated solutions if oxygenation appears possible.

2. Behavior of Crevices in Reactor Operation After Decontamination

The crevice area has been identified already as one of major concern by both consultants and Dow. However, there are certain circumstances in this regard which may need more careful consideration. These circumstances imply that the following experiments should be considered:

a. The solution chemistry of sequestered decontaminating agent should be evaluated after operation at reactor operating temperatures. Particular attention should be paid to the interaction of the decontaminating solution with iron oxides in the crevice.
b. Since a relatively high heat transfer circumstance exists in these crevices, the possible effect of wetting and drying or boiling out effect on this crevice should be evaluated.
c. The materials in the crevice should be evaluated with respect to general attack and stress corrosion.

3. Dissolution of Copper

It appears that a substantial amount of copper exists in the deposits. The possibility that the copper may be dissolved but subsequently plated on carbon steel surfaces is a substantial concern. Should this occur, the possibility of subsequent loosening of such copper deposits and producing flow blockages is possible.

Dow has suggested several techniques for removal of copper. The possible techniques with which I am familar suggests that very serious consideration should be given to which technique is chosen and the implications of these techniques on the corrosion behavior during cleaning as well as the subsequent corrosion behavior produced by deposit sequestered as a result of this action.

Thus the entire copper question should be reviewed very carefully with respect to agents which can solubilize and remove the copper.

4. An Effect of Decontamination Solution of Copper Base Alloys

A review of the Dow composition suggests that particular attention should be paid to the crevice and stress corrosion behavior of copper base alloys over a range of variations in solution. It is not presently clear to me as to the extent of exposure of copper base alloys. Thus, this matter should be carefully assessed. If it is possible to prevent the solution from being exposed to copper base alloys, it would be desirable. On the other hand, testing may show that despite apparent possible sensitivities that, in fact, the presence of inhibitors may obviate potential concerns.

5. Behavior of Irradiated Material

There continues to be a concern for the effect of the cleaning solution on irradiated materials. Since these experiments will inevitably be expensive; and, also, since the availability of material is restricted, particular attention should be paid to the choice of experimental conditions in which the irradiated materials are exposed. I would propose that a specific experimental program be outlined and agreed to in this area. Certainly items which should be considered in this are the following:

a. stresses
b. crevice geometry
c. sequestered solution following decontamination

6. Effect of Variations of Solution Composition

The decontaminating solution consists of a variety of species. The solution contains various inhibiting agents. It is most important that variations in the solution composition be incorporated as testing environments. To some extent this has already been done. However, specific areas in which careful attention should be given are the following:

a. presence or absence of inhibitors
b. variations of oxygen content together with item "a"
c. impurities possibly introduced by the decontaminating solutions
d. substantial variations in the major components of the solution

Specifically, I wish to suggest that a more careful consideration be given to the range of circumstances of the major components of the solution. While the test solutions might be code named for convenience, nonetheless certain critical worse case circumstances should be evaluated in these circumstances.

7. Interaction of the Solution with Existing Deposits in Chemistry Prior To Decontamination

The actual solution which exists in the decontaminating enviroment will include not only those species which are introduced intentionally but will involve the interaction of these species with deposits which already exist. These include, certainly, the iron oxides, copper oxides, and may also include deposits which have resulted from water treatment such as phosphate or sulfites and their appropriate derivatives. A careful analysis of these circumstances should be developed to ascertain possible additional variations in the actual solution composition which should be considered. To date Dow has already considered such contaminants as fluoride, lead, and chloride. However, it is not clear that this comprises the full set.

Further, the interaction of these impurities is significant when the oxygen concentration is changed. To have studied, for example, the fluoride contamination in the absence of oxygen may not be so crucial as the same interaction in the presence of oxygen.

On the other hand, a concept of reasonability in defining such environments also needs to be established. A specific proposal from Dow in this regard should be considered.

8. Identification of Gasket Materials

It is not clear that the gasket materials in the final decontaminating hook-up have been properly or extensively identified if they exist. I think a more careful consideration is possible for materials introduced by gaskets or other similar materials should be evaluated.

9. Quality Control of the Decontaminating Solution

Since the corrosion behavior of the alloys has been specifically established over very definite ranges of impurity contamination, it is crucial that the decontaminating environment stay within the ranges for which performance parameters have been established. A very specific set of impurity analyses and other environmental parameters should be established and means demonstrated by which these will be controlled.

10. Effectiveness of Decontaminating Solutions

There remains some question in my mind whether the decontaminating solution is as effective as Dow considers it to be. While it seems to be a very powerful system, there is still a question in my mind as to whether it will remove deposits especially in long vertical crevices and tubes. This matter should be reevaluated.

11. High Area Ratio Crevices

There is clear evidence that the area ratio in the crevices produces a substantial increase in corrosion rate. One wonders whether there should be additional stress corrosion tests in these high area ratio crevices. This specific issue should be evaluated to see whether such a condition exists and its basis for an appropriate test.

The above constitute areas which are worthy of serious concern. I suggest that these be reviewed by Dow and specific answers prepared. Naturally, I would be happy to work with both Commonwealth and Dow in the consideration of the above items.

Sincerely,

Roger W. Staehle

R. W. Staehle
Consultant

RWS:cgw

Burns and Roe, Inc.

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October 28, 1974

Mr. William P. Worden, Administrative Engineer
Office of Vice President L. F. Lischer
Commonwealth Edison Company
One First National Plaza
P.O. Box 767
Chicago, Illinois 60690

Subject: Decontamination of Unit No. 1
Dresden Nuclear Power Station

Dear Mr. Worden:

At your request I attended a meeting in your offices on October 21, 1974 with you, members of the Industrial Service Division of the Dow Chemical Company, and Dr. Craig F. Cheng of the Argonne National Laboratory. During the meeting we reviewed the results of studies conducted to date by Dow relative to the decontamination program for Unit No. 1 of the Dresden Nuclear Power Station. As you requested during the meeting, I am summarizing my comments and recommendations in this letter.

As I understand the information presented during this and the previous meetings I have attended on this subject, the status of the decontamination program for Dresden Unit No. 1 is as follows:

1. Because of high after-shutdown radiation levels, there is a need to decontaminate Dresden Unit No. 1 before undertaking certain modifications and inservice inspection items required to meet Atomic Energy Commission requirements for long term continued operation. In addition, there is always the possibility at any time of repairs or maintenance work that current after-shutdown radiation levels would make difficult or impractical.

2. Cleaning solutions previously used for decontamination of pressurized water nuclear power plants have proven to be ineffective in removing the type of scale deposits found in the Dresden boiling water nuclear power plant. In particular, the Newport News Industrial Corporation has recently attempted to clean scale deposits from a coupon removed from the Dresden plant with cleaning solutions of the type used for decontamination of Navy shipboard

pressurized water nuclear power plants. I understand the test conducted by Newport News was unsuccessful in that the scale deposits were not completely removed and the decontamination factors were low.

3. Extensive development work has been performed by Dow on a new and proprietary cleaning solution. The major aspects of the development program are:

   (a) A survey was made of the materials in the primary coolant systems of the Dresden plant which would be exposed to the cleaning solution during decontamination. This survey was based on all available construction drawings and vendor component drawings.

   (b) General corrosion, bi-metallic corrosion, and stress corrosion tests were made in the cleaning solution of categories of materials used in the primary coolant system of the Dresden plant to determine whether the cleaning solution could have a detrimental effect on the Dresden plant. Although not all materials were extensively tested, screening tests and extensive tests of representative materials from each category give confidence that the test program includes the parameters of importance for the Dresden plant.

   (c) These tests indicate, in general, that corrosion rates are low; that cracking is not induced in any of the materials tested with the exception of heavily sensitized and severely cold-worked type 304 stainless steel. Although type 304 stainless steel is the major structural material of the primary coolant systems of the Dresden plant, it is unlikely that the material in the Dresden plant is both heavily sensitized and severely cold-worked. Heavily sensitized and severely cold-worked type 304 stainless steel will also crack in an environment as mild as demineralized water.

   (d) Decontamination tests with radioactive coupons removed from the Dresden plant indicate that very high decontamination factors are achieved with the cleaning solution.

   (e) The cleaning solution has recently been used to clean a stainless steel heat exchanger at the CP5 reactor facility at Argonne National Laboratory. Not only was the heat exchanger successfully cleaned, but it has been returned to service and there is no evidence of detrimental effect from the cleaning operation.

Without regard to the question of decontamination, there is a need for improved radioactive waste facilities for Unit No. 1 of the Dresden Nuclear Power Station. Facilities and equipment are required as part of the decontamination program to concentrate the used cleaning solutions and package the residual radioactive materials for offsite shipment. These same facilities could also be used subsequent to decontamination for normal radioactive waste disposal operations associated with the Dresden plant.

In consideration of the information presented during this and previous meetings, I believe that the decontamination program for Dresden is soundly conceived and can proceed with minimum risk. However, it is still a very large step from laboratory testing of the decontamination process and a field test on a heat exchanger to full-scale decontamination of the Dresden plant. The risk could be reduced by trying out the decontamination process on an experimental boiling water reactor now decommissioned such as EBWR. There are several reasons why such an intermediate step would be prudent, including the following:

1. The major structural material of the EBWR primary coolant system is also type 304 stainless steel, the material of prime concern in the Dresden decontamination program. It is likely that sensitized areas, configurations such as crevices and deadlegs, and material combinations of concern in the Dresden decontamination program would occur in the EBWR plant.

2. It would be feasible to remove sections of the EBWR primary systems for destructive examination after decontamination, a step which would not be possible at Dresden.

3. Decontamination of EBWR would provide full scale experience with procedures and methods, and training of contractor personnel, which would be of substantial value in a subsequent Dresden decontamination program.

4. In any developmental program there should always be concern for unanticipated difficulties. A prototype test of the decontamination program at EBWR might reduce, or even eliminate, many difficulties when decontamination of Dresden is undertaken. It is conceivable that the decontamination program at Dresden could be expedited to such an extent that the cost of a prototype decontamination program at EBWR would be partially amortized.

Enclosure 1

1. - PRELIMINARY REPORT, Feasibility Study for Chemical Cleaning of Dresden Unit-1, DIS-NS-D1-008, dated October 18, 1974

2. - Drawing No. A-101, Primary System Decontamination Plot Plan

3. - Drawing No. A-102, Equipment Arrangement Radwaste Facility & Tank Farm

4. - Drawing No. A-103, Equipment Arrangement Rad Waste Facility Sections

5. - Drawing No. A-104, Equipment Arrangement Rad Waste Facility Sections

6. - Drawing No. R-201, Process & Instrumentation Diagram Primary System Decontamination

7. - Drawing No. A-202, Process & Instrumentation Diagram Lead Sheet

8. - Drawing No. R-203, Process & Instrumentation Diagram Rad Waste Facility

9. - Drawing No. A-204, Utility Tie-Ins

10. - Drawing No. 16001, Electrical, Single Line Diagram

In the spirit of the proprietory agreement I made with Dow, I am returning herewith the documents indicated in Enclosure 1, which I received at the Octobpr 21, 1974 meeting. My review of the documents has been limited to the points described in this letter. I have not reviewed these documents from the standpoint of chemical process, flow sheets, or arrangement of physical equipment, as such a review must involve additional personnel at Burns and Roe and would not be in accordance with the present proprietory agreement. Burns and Roe is capable and willing to assist you in reviews of this type under an expanded proprietory agreement. On the other hand, you may want to consider having such reviews made by the architect-engineer responsible for the design of Unit No. 1 of the Dresden Nuclear Power Station, since tie-in with existing facilities is involved. In addition, you may want to have such documents reviewed by personnel from the Newport News Industrial Corporation as they have extensive experience in the decontamination of Navy shipboard pressurized water nuclear power plants which they believe is applicable to the decontamination of commercial nuclear power plants.

I hope you consider this an adequate summary of my comments and recommendations as the Dresden decontamination program stands at present. I look forward to assisting you further in any way I can, either in future technical meetings or through review and comment on technical data, reports, or procedures involving decontamination.

Very truly yours,

Tom A. Hendrickson
Deputy Director
Power Engineering Division

TAH:bam
Enclosures

CRAIG F. CHENG
332 King's Cove
Lisle, Illinois 60532

October 25, 1974

Mr. William P. Worden,
Administrative Engineer
Commonwealth Edison Company
One First National Plaza
P.O. Box 767
Chicago, Illinois 60690

Dear Bill:

Decontamination Program--Dresden-I
Review of Status and Recommendation

I have reviewed the data and the experiments in progress on the captioned program. In my opinion, Dow Chemical Company has done enough tests on non-irradiated metals and alloys to show that there is no ill effects of Dowcon-1 on BWR systems, provided the same data applies to irradiated materials and there is no essential disagreement with Dr. Roger Staehle's experiments on applied potential measurements and short-term tensile tests. In the case of irradiated Type 304, I have suggested a test program, which Dow Chemical Company can incorporate into their planning (see attachment A).

Furthermore, I agree with Dow Chemical Company that the corrosion product film in BWR systems is different from that in PWR systems as witnessed by the unsuccessful result of Newport News Decontamination Test using the two-step standard U. S. Navy procedure (APAC-HEDTA - re: my letter dated August 18, 1974). In addition, I have successfully decontaminated sections of the recirculation piping in Dresden-II and Quad Cities-II even though their corrosion film are similar but not identical to Dresden-I (see attachment B). However, regardless how successful Dowcon-1 performs in laboratory static and dynamic tests, I still feel it is prudent to conduct a pilot-plant test prior to full scale decontamination of Dresden-I. There are three locations, where you may want to explore in conducting the pilot-plant test. They are (a) The lower part of of the reacotr vessel of BWR at Argonne National Laboratory, (b) Some components of VBWR of GE Company at Valtecitos, California, and (c) A part of the secondary steam generator loop in Dresden-I BWR systems.

Sincerely,

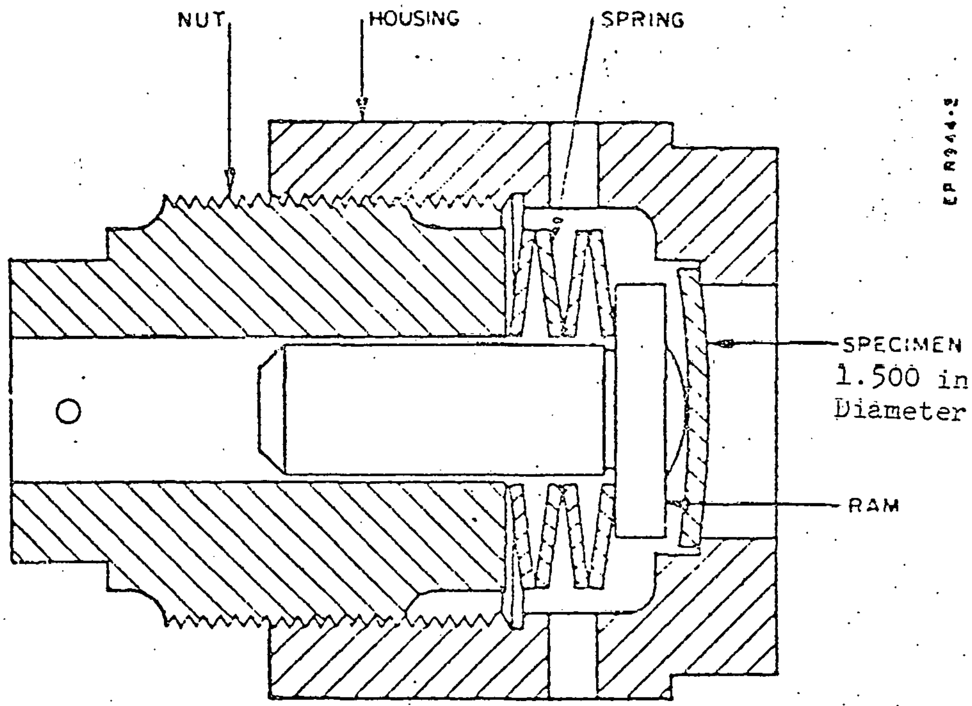
Craig F. Cheng

Craig F. Cheng

CFC:jln

Encs: A. A Proposed Outline on Decontamination of Irradiated Materials From Dresden-I.

B. Notes on Decontamination of Recirculation Piping in Dresden-II and Quad Cities-II.



Disc Specimen, 0.090-Inch-Thick, Biaxially Stressed
(Ref. WAPD-TM-944)

ATTACHMENT A

A Proposed Outline on Decontamination of Irradiated Materials From Dresden-I

The proposed outline is a test for determining the corrosive effects of Dowcon-1 on irradiated Type 304 stainless steel.

(I) Test Facility

Conduct decontamination program in the two dead legs of the dynamic loop to be transferred to Midland, Michigan from Freeport, Texas.

(II) Irradiated Type 304 SS - (Dresden-I BWR)

a. Hexagon can at core periphery ($\sim 3 \times 10^{12}$ nv > 1 Mev)
b. 8-in. steam line to emergency condenser
c. Hand hole cover from steam generator

(III) Non-irradiated Materials

The same 1020 carbon steel, 405 SS, 17-4 PH SS, and Stellite coupons or specimens used at Freeport, Texas.

(IV) Stress Corrosion Test

a. Conduct similar jig loaded tensile test with irradiated Type 304 SS (II-a) in either rod or strip form.
b. Conduct biaxially stressed disc specimens (0.090-inch thickness) as described in WAPD-TM-944 with irradiated Type 304 SS (II-b and II-c). Use the oxide scale surface for the tensile side.

(V) Crevice Corrosion Test

Use multi-facet washer (10 slots) of teflon and non-irradiated materials in contact with the oxide-scale surface of irradiated Type 304 SS and (IIa and IIb or IIc). Insert carbon steel washer only in the second leg (Carbon steel tends to convert to collodial iron in Dowcon-1, as observed in Midland test dated 9-5-74).

(VI) Stressed Crevice Specimen Test

Use WOL specimens of non-irradiated Type 304 SS originally intended for testing at Freeport, Texas.

ATTACHMENT B

Notes on Decontamination of Recirculation Piping in Dresden-II and Quad Cities-II

In the course of investigating the cracked recirculation piping in Loop B of Dresden-II and Quad Cities-II BWR Systems, it was necessary to clean up the oxide-spinel scale on the fractured surface and inside diameter (ID) surface. Preliminary X-ray deffraction data indicated that the scale on Dresden-II piping is predominately spinel of nickel-ferrite with substitutions of nickel-chrome spinel, plus FCC iron. The scale on Quad Cities-II piping is similar to Dresden-II except there is an addition phase of possibly colbalt ferrite. (The scales on Dresden-I piping in the cleanout fitting and demineralizer supply line were chrome and nicket ferrites)

Dowcon-1 was selected to remove the scale at the fractured surface and the inside diameter surface. The results are summarized below:

a. Fractured Surface

After two successive cleaning cycles of 100 hours in Dowcon-1 at 250°F followed by ultrasonic cleaning in acetone for 1 hour, the scale on the ID surface was completely removed, but only 10% of the scale on the fractured surface was cleaned off. The radioactivity remaining on these specimens measured about 7 mr/hr @ 2-in. of beta & gamma and 5 mr/hr @ 2-in. of gamma. FCC iron probably prevented the scale from sloughing from the irregular fractured surface.

b. ID Surface

Sections of the above mentioned piping were decontaminated with Dowcon-1 at 250°F for 114 hours and followed by ultrasonic cleaning in acetone for 1 hour. Each section contains the weld-metal, the counterbored region and the undisturbed straight run. The ID surface was brightly cleaned, except for some black scale at the weld-bead after decontamination. The latter scale was removed during ultrasonic cleaning. The radioactivity measurements at the ID surface are listed below:

		DRESDEN-II	QUAD CITIES-II
As Received	Beta & Gamma	80 mr/hr @ 2-in.	60 mr/hr @ 2-in.
	Gamma	27 mr/hr @ 2-in.	22 mr/hr @ 2-in.
After Dowcon-1	Beta & Gamma	3 mr/hr @ 2-in.	9 mr/hr @ 2-in.
		14 mr/hr @ 1-in.	37 mr/hr @ 1-in.
	Gamma	1 mr/hr @ 1-in.	4.8 mr/hr @ 1-in.
After Ultrasonic Cleaning	Beta & Gamma	8 mr/hr @ 1-in.	12 mr/hr @ 1-in.
	Gamma	0.5 mr/hr @ 1-in.	1 mr/hr @ 1-in.

The gamma activity was mostly due to Co^{60} and the beta activity was attributed to Co^{58} decay, which penetrated to a depth of about 10 mils in the substrate.

APPENDIX XI

PROCEEDINGS OF THE AMERICAN POWER CONFERENCE

VOLUME 37 - 1975

PROTECTING DRESDEN I REACTOR COOLANT MATERIALS FROM CORROSION DURING DECONTAMINATION AND AFTER RETURN TO SERVICE

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INTRODUCTION

Commonwealth Edison has instituted a plan to decontaminate Dresden Unit 1 in order to reduce radiation exposures. The procedure chosen involves a chemical decontamination wherein the decontaminating solution will be circulated, followed by a rinsing. A new solvent was formulated for this procedure since previously available ones were ineffective. The full-scale cleaning of Dresden Unit 1 is scheduled for 1977.

This paper is concerned primarily with the corrosion aspects of the decontaminating process and describes important issues which were considered in developing the decontaminating solution. Various tasks involved in assessing possible corrosion problems are described and available results are summarized.

This paper is exemplary and does not include all the data which have been amassed to date. The work under way in support of Dresden I involves an extensive program coordinated by Commonwealth Edison and is being conducted by Dow Industrial Services, Argonne National Laboratory, General Electric, and private consultants.

In developing effective solutions for decontamination it is important that they do not compromise the materials of construction during the decontaminating process nor lead to aggravation of corrosion at a later time during normal operation or layup.

In order to assess the latter, an extensive testing program has been undertaken to evaluate the possibilities of corrosion damage which might occur subsequent to the decontaminating procedure.

IMPORTANT ISSUES IN CORROSION TESTING

The corrosion testing program was organized to assess the following issues.

1. The corrosion behavior of structural materials during the decontaminating process with respect to the following:
   a. all alloys and heat treatments which the decontaminating solution would wet;
   b. all special geometrical circumstances of bimetallic joints, crevices, and welds;
   c. stressed metals in the open and crevice circumstances;
   d. variations in temperature from the nominal;
   e. prolonged times beyond those originally planned;
   f. changes in the composition of decontaminating solutions resulting from accumulation of corrosion product impurities;
   g. various states of aeration;
   h. inadvertent change in the chemistry of the decontaminating solu-

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TABLE I
MATERIALS FOUND IN DRESDEN 1 THAT WILL CONTACT CHEMICAL CLEANING SOLUTION

WETTED MATERIALS:

AISI	ASTM	ASME
302	A53-B	SA48-25, 30
303	A105-2	SA53
304	A106-B	SA105-2, 1
316	A155-KC70	SA107-1137, 1141
347	A167-3	SA108-1035
410	A182-F304, 304 ELC, 316, F22	SA113
416	A193-B8, 416	SA120
B16	A194-8, 1	SA132-304
B113	A212-A, B	SA155-2¼, CL1
C1040	A213-304	SA182-F11, F6, F304
C1045	A216-WCB	SA194-C12H
C1213	A240-304, 304L, 405	SA216-WCB, WCA
	A249-304L	SA217-WCL, WC9
	A264-304L	SA234-WP22, WP22W, WPB, WPBW
	A268-405, 410	SA266-2
	A269-321	SA269-304
SAE	A270-304, 410, 410H, 420	SA278-25
	A296-CA15	SA285-C
SAE 40 (Brass)	A298-304L, 308, 309	SA298-308L, 209
SAE 64	A302-B	SA335-P11
SAE 660 (Bronze)	A312-304, 316	SA336-F8
SAE 1112	A335-F1, P22	SA351-CF8
	A336-F1	SA358-S
	A371-309	SA403-WPW 304, WPW 326
	A376-304	SA511-MT321
	A479	SB30
	A516-70	SB62
	A582	SB143-A2
	B371	SB145-4A
		P-3442B

OTHERS

Asbestos	Graphitar 14
Carpenter Mirromold	Hastelloy C
Cast Iron C130	Haynes - 25, 21
Co-Cr-W Alloy (AWS-5.13)	Inconel
Copper	Monel
Copper and Neoprene	Nhrile
Everdur	Si-Bronze
Flexrock 401	Stellite 6
Garlock 24	TP 17-4PH

tion, e.g., absence of the inhibitor;

i. propagation of preexisting cracks; and

j. effects of residual gamma activity in structural materials near the core.

2. The corrosion behavior of crevices after the decontaminating solution has been removed in either the conditions of reactor operation or layup.

3. The possibility that preexisting cracks with sequestered decontaminating solution would propagate during reactor operation.

TABLE II

BIMETALLIC JUNCTIONS OF WETTED MATERIALS IN DRESDEN 1

Junction Material (asterisk indicates crevice)	Equipment Piece (see legend at bottom of table for explanation)
303-304*	G-17
304-304L*	C-2, C-3
304-316*	G-17, Pipe: C-1, Valves: SP115M2, 108M2, SP213-M2
304-347*	G-17, Valve: 108M2
304-405*	F-4, F-16
304-410*	C-2
304-410H*	F-16
304(ANN)-410	G-125
304(ANN)-420 (H.T.)*	G-125, G-39
304-1020*	C-2, E-2, E-4, E-7, G-39
304-1112	C-2
304-H25 Alloy*	Valve: 208M2
304-TP17-4PH*	G-4, G-17
304-Copper*	G-17
304-Chrome*	G-17
304-Graphitar*	G-17
304-Flexatallic 304*	Valves: SP115M2, SP213M2
304-Inconel*	G-4, G-17
304-Stellite #6*	G-39, Valve: A208M2
304-Monel	Valve: A208M2
304L-405*	C-2
304L-410*	C-2
304L-1020*	C-2, C-8, E-2, E-3, E-4
308L-1020*	F-4, C-2
309-1020*	F-4, C-2
316-347*	Valve: 108M2
316-17-4P4*	Valve: SP115M2
347-Copper*	G-17
347-2¼ Cr 1 Moly*	Valves: 198M1, 110
347-Stellite #6*	G-17, Valves: 108M1, 110
347-Hastelloy C*	Valves: 108M1, 110
405-410*	C-2
405-410H*	C-2, F-16
405-416	C-2
410-410H	G-125
410-230	G-125
410-1020*	Valves: MV10, MV6, SPA116, 223
410-Carpenter Mirromold*	Valve: MV10
410-Stellite #6*	G-39
410-Flexatallic 304*	Valve: MV6 SPA116
410-Tungsten, Cobalt, Chrome Alloy	Valves: 223, 401
416-A1S1 C1213*	G-54
410-18 Cr 8 Ni: StSt1*	Valve: 401
416-Cast Iron #30*	G-54
416-SAE 660*	G-54
410-420*	G-39
1020-1137/1141*	E-7
1020-1112	G-125
1020-Nitrile (O-rings)*	G-39
1020-Flexatallic*	G-125, Valve: MV6-SPA116
1020-70/30 Cu/Ni*	E-7
1020-2 ¼ Cr 1 Moly*	Pipe: C-1, C-2, C-3, C-4, Valves: 108M1, 110, 108M2
1020-Cast Iron C130	G-125
1020-Asbestos*	Valve: 223
1020-Carpenter Mirromold*	Valve: MV10

TABLE II CONT.

Junction Material	Equipment Piece
1020-Stellite*	Valves: MV10, MV6SPA116, 110, 108M2
1112-Cast Iron C130*	G-125
2 ¼ Cr 1 Moly-Tungsten Cobalt Cromium Alloy	Valve: 401
2 ¼ Cr 1 Moly-18 Cr 8 Ni StnStl	Valve: 401
2 ¼ Cr 1 Moly-Stellite #6*	Valves: 108M1, 108M2
H25 Alloy-Stellite #6*	Valves: 108M1
Graphitar 14-Stellite #6*	G-17
#40 Brass-B1113*	G-54
B-62-Everdur*	G-125
B-30*	G-125
B-30 · B-62*	G-125
Cast Iron #30-SAE660*	G-54

Equipment Piece Number Identification:

C-2	Reactor Pressure Vessel
C-3	Drum, Primary Steam
C-8	Tank, Reactor Cleanup Demineralizer
E-2	Secondary Steam Generator
E-3	Heat Exchanger, Regenerative, Cleanup Demineralizer
E-4	Heat Exchanger, Regenerative, Cleanup Demineralizer
E-7	Heat Exchanger, Reactor Unloading
E-111	Cooler, Reactor Enclosure Drain Tank
F-4	Turning Vane, Reactor Pressure Vessel
F-6	Vessel Thimble
F-15	Diffuser Basket with Poison Sparger
F-16	Guide, Grid
F-17	Plate, Core Support
F-21	Control Rod Drive Tube Assembly
G-4	Pump, Cleanup Demineralizer Recirc.
G-17	Pump, Reactor Recirculating
G-39	Pump, Unloading Recirculating
G-54	Pump, Reactor Enclosure Drain Tank
G-125	Pump, Reactor Area Sump

4. The behavior of cleaned surfaces during reactor operation.

In selecting the experimental arrangements, it was assumed that no reasonable environmental condition could be overlooked. The occurrence of stress corrosion cracking in relatively innocuous conditions has been all too frequent in the past.

Further, it was assumed that a set of worst-case conditions could be arranged which incorporated conditions of heat treatment, alloy, geometry, and stressing. Should such tests be satisfactory, they would provide the greatest confidence in the corrosion testing program.

The following sections describe the status or important results of the work which was undertaken to answer the concerns outlined above.

MATERIALS

An extensive study was conducted by Dow engineers to identify all wetted materials in Dresden I. These materials, together with their various designations, are listed in *Table I*. These materials were identified by a detailed review of the design drawings and by examining each of the piping runs individually.

In addition, the existence of bimetallic junctions and crevices was identified; these are summarized in *Table II*. Where the bimetallic junction is a crevice, it is noted with an asterisk.

GENERAL, GALVANIC, AND CREVICE CORROSION

Results from the extensive corrosion testing program showed that the Dow solvent, NS-1, did not cause significant corrosion. These results are summarized in *Table III*. These data were normalized to a period of 300 hours since this represents twice that of the expected duration of the decontaminating operation.

The data of *Table III* were obtained in static tests using Teflon-lined or glass-lined cells. Specimens generally were ¾-in. wide by 6 in. long by 1/16 in. thick.

In addition to the static tests, a series of dynamic tests was conducted in a flowing loop where the flow velocity was about 7.5 ft/sec. In all cases the attack was not extensive. These results are shown in *Table IV*.

STRESS CORROSION CRACKING TESTING

An extensive program of testing for stress corrosion cracking was conducted using a wide range of metallurgical and environmental conditions. The previous results from general, bimetallic, and crevice attack showed the expected low rates. However, stress corrosion cracking is always the most serious concern owing to the rapidity of crack propagation, often under relatively innocuous circumstances.

Special emphasis was placed upon evaluating the stress corrosion cracking behavior of sensitized stainless steel since it was considered to be the most sensitive material. Studies of other materials were also conducted. Results from a general screening test for a variety of alloys are given in *Table V*. These tests were conducted in a solution of NS-1 which was air saturated at room temperature.

Two-stage single U-bends were fabricated according to ASTM method G-30. The specimens were ¾ of an inch wide and 6 inches long before bending. Bending was done after any heat treatment.

The austenitic alloys were sensitized at 1200 F for 50 hours with a furnace cool. The 400 series alloys were subjected to an 885 F temper embrittlement heat treatment for 100 hours with air cool. The 17-4 pH alloy was given an 1150 F heat treatment for 4 hours with air cool. The Inconel alloys were sensitized at 1200 F for 50 hours and furnace cooled. Heat treatment scales were removed prior to testing. This set of tests constituted reasonable worst-case conditions for important alloys.

The U-bend specimens were placed in Teflon-lined test spools and statically tested for 300 hours at 275 F in air-saturated Dow Solvent NS-1. (Solvent air saturated at room temperature.) No stress corrosion cracking was observed visually or metallographically.

The results in *Table V* show clearly that the normal chemistry of NS-1 does not cause stress corrosion cracking under reasonable worst-case conditions.

Certain conditions were found which cause stress corrosion cracking of stainless steel. These were associated with special conditions wherein the ferric (Fe^{+3}) and nickelous (Ni^{+2}) ions were

TABLE III
SUMMARY OF CORROSION DATA FOR REPRESENTATIVE ALLOYS

AISI Type Alloy	General Corrosion*	Crevice Corrosion	Galvanic Corrosion*
304	0.009	None	304L (0.0001) to 1020 (0.35)
304 Sen.	0.106	None	304L (0.0027) to 1020 (0.824) (1:10)
304L	0.039	None	304L (0.0011) to 1020 (1.406) (10:1)
304L Sen	0.089	None	304L (0.0009) to 1020 (2.8396) (1:30)
347	0.023	None	347 (0.0011) to 1020 (0.3385)
405	0.123	None	405 (0.4759) to 1020 (0.0802)
410	0.057	None	410 (1.14) to 1020 (0.34)
446	0.008	None	446 (0.0013) to 1020 (0.2032)
Inconel 600	0.011	None	—
17-4 PH	0.037	None	17-4 (0.0009) to 304 (0.0013)
Hastelloy B	0.033	None	—
Copper 122 and 715	0.051	None	—
4419	1.4	None	4419 (6.67) to 304 (0.010) (1:20)

Comments

General Corrosion—Each number represents an average weight loss from at least 5 specimens converted to penetration (mils)/300 hour test time. All testing done under air saturated condition, with test temperature set at 275 F.

Crevice Corrosion—Each alloy has been tested with artificial Teflon crevices and in a double U-bend configuration (stressed crevice). (No crevice initiation occurs on stainless alloys, copper alloys, or nickel based alloys.) Tests were run at 275 F from 100 to 300 hours.

Galvanic Corrosion—Each alloy couple has a (1:1) area ratio except where noted. Couples were made by rubber banding coupons together. The galvanic tests were made in air-saturated conditions at 250 F for 100 hours.

All corrosion tests were in Dow Solvent NS-1 without added impurities

* NOTE: Corrosion numbers represent total mils penetration (1 mil = 10^{-3} inches). Penetration is assumed to be uniform over the surface of the specimen.)

TABLE IV
CORROSION DATA FROM THE DYNAMIC TEST LOOP

Specimen	Corrosion Rate, Total mils penetration in 300 hours
Galvanic Couples (Welded 1:1 area ratio unless otherwise noted)	
*410 - 304 Sensitized (1200 F for 50 Hours)	0.30 (Average of 6)
316 - *304 Sensitized (1200 F for 50 Hours)	0.24 (Average of 5)
304 - 304 Sensitized (1200 F for 50 Hours)	0.35 (Average of 5)
316 - 316	0.04, 0.05
304 Sensitized - 304 Sensitized (1200 F for 50 Hours)	0.32, 0.32
304* - Hastelloy B	0.47
304 - 4419* (20:1 area ratio) Specimens were welded with a crevice present to simulate a cladding crack	20 (Average of 3)
304* - Graphitar (2:1 area ratio)	0.03, 0.01
General Weight Loss	
304 (U-Bends)	0.09 (No cracking)
Stellite #6	0.03
446 (Flow Rate of 7.5 ft/sec)	0.002 (Average of 3)
446 (Stagnant Conditions)	0.02
410 (Flow Rate of 7.5 ft/sec)	0.02 (Average of 3)
410 (Stagnant Conditions)	0.05
CN FM Cast Stainless Alloy	0.01 (Average of 5)
Crevice Specimens (4-in. by 4-in. plate with Teflon crevice)	
304	0.01 (No crevice attack)
Monel 400	0.05 (No crevice attack)
Incoloy 800	0.08 (No crevice attack)
Stress Specimens	
Sensitized, cold worked tensile rods, 1250 F for 6 hours, strained 5%, 1200 F for 50 hours, stressed to 2% strain	No cracking

NOTE: * — Signifies the anodic member of a galvanic couple.
Conditions of the loop test:
Time—300 Hours
Oxygen Conc.—Air saturated initially, and at 150 hours the loop was opened and the Dow Solvent NS-1 was again air saturated with oxygen
Flow Rates—7.5 ft/sec in high velocity tests
Temperature—250 F
Solution—Used Dow Solvent NS-1 with iron concentration of 0.12 - 0.18 weight percent

TABLE V
STRESS CORROSION SCREENING DATA FOR 300 HOURS AT 275 F IN DOW SOLVENT NS-1

Two-Stage Single U-Bends of:	Number Tested	Results
304	14	No cracking
304 sen. and descaled	22	
304 L	17	
304 L sen. and descaled	15	
316	5	
17-4 PH (1150 F heat treatment)	5	
Hastelloy B	5	
Hastelloy C	5	
Inconel 600	5	
Inconel 600 sensitized	5	
Incoloy 800 sensitized	5	
405	5	
405 - Temper embrittled	5	
410	5	
416	6	
446	5	
442 - Temper embrittled	5	
446 - Temper embrittled	5	
2 ¼ Cr-1 Mo	5	
4142	5	
1020	5	
Red Brass, CDA-230	5	
70/30 Cu Ni, CDA-715	5	
Cu, CDA-122	5	

present together with extensive sensitization of the alloy. When these results were obtained, it was necessary to determine whether such conditions were more aggressive than pure water conditions to which the reactor had already been exposed and which were already known to cause stress corrosion cracking. To assess the significance of these conditions and the relative behavior of pure water and NS-1, tests using smooth and precracked specimens were conducted. These experiments showed that the NS-1 solvent is no more aggressive in pure water and, in fact, may be less aggressive. Results from U-bend tests are shown in *Table VI*. In no case did stress corrosion cracking occur in NS-1 and not in pure water.

The relative effect of pure deionized water and NS-1 were evaluated using precracked specimens; here the effect of an initially sharp crevice could be evaluated. In the precracked specimens, the stress corrosion cracking was more pronounced in the specimens exposed to deionized water than to NS-1.

Other environmental impurities were studied separately including up to 100 ppm Cl^-, 100 ppm F^-, 100 ppm $S^=$, 100 ppm Pb^{+2}, 1000 ppm Cu^{++}, and 650 ppm Ni^{++}. None of these caused stress corrosion cracking of the sensitized stainless steel when exposed to NS-1.

The effect of cold work on the stress corrosion cracking of sensitized specimen was also evaluated. These tests involved tensile specimens that were sensitized six hours at 1250 F then cold worked to 5 percent plastic strain followed by a sensitization of 50 hours at 1200 F with a furnace cool. Next, the specimens were loaded into jigs and stressed to a level that would produce up to 5 percent permanent deformation. At this stress, the specimens were locked into position and tested for 300 hours in Dow Solvent NS-1. This type of test has been carried out on Dow Solvent NS-1 with ferric ion impurities to 0.12 percent, nickel ion impurities to 0.065 percent, chloride, lead, and sulfide impurities to 100 ppm. In no case has stress cracking of the sensitized stainless occurred in this specimen configuration.

ELECTROCHEMICAL STUDIES

Electrochemical studies using sensitized stainless steels were conducted in the corrosion laboratories of the Ohio State University. These studies were conducted at room temperature and 260 F in the sensitized (50 hours at 1200 F) and quench annealed conditions. (See *Table VII*.)

TABLE VI
RESULTS FROM U-BEND STRESS CORROSION TESTS

	Dow Solvent NS-1	Deionized Water
Total Number of Single U-Bends	58	—
Total Number of Double U-Bends	157	18
Total Number of Cracked U-Bends		
Single	6	—
Double	72	10
Conditions Causing Cracking		
Presence of heat treatment scale		
Single	2	—
Double	5	2
Presence of heat treatment scale with 0.12% Fe+++ + 0.065% Ni++		
Single	4	—
Double	51	6
No heat treatment scale but 0.12% Fe+++ + 0.065% Ni++		
Single	0	—
Double	15	2
(All cracks were intergranular)		

* Specimens prepared to ASTM G-30 3/4" X 6" before bending
— 112 hours at 250 F in Teflon lined test spools
—sensitized species: 1200 F for 50 hours and furnace cooled
— all solution air saturated at room temperature

These studies are the subject of a separate report. The results show that there is no unexpected acceleration in corrosion which results from changing the state of aeration or from changing the inhibitor concentration. Further, the heavy sensitization treatment (50 hours at 1200 F) does not accelerate the corrosion rate significantly.

No evidence of pitting or other unstable breakdown was observed.

ADDITIONAL WORK

Work on corrosion testing continues. Special emphasis is being placed upon work in the following areas:

1. The effect of decontaminating treatment on subsequent stress corrosion cracking in BWR environments will be evaluated by exposing stressed specimens in water containing 0.2 ppm oxygen at 550 F. These will be conducted statistically to compare the effects of decontaminating treatments with those of pure water.
2. Specimens will be exposed during the decontaminating operation itself. Some of these will be left in during plant operation to evaluate effects of reactor operation over longer periods of time.

SUMMARY

Not all of the work which has been conducted is reported in this paper. However, the results shown in the tables are typical and cover the broad range of conditions considered in this study.

The results show clearly that there is no accelerated corrosion of metals associated with the NS-1 decontaminating solution, nor does this solution accelerate stress corrosion cracking.

The NS-1 solution appears to be

TABLE VII
MEASUREMENTS ON SCC - WOL SPECIMENS

Spec. No.[a]	Test[b] Environment	Initial Load, lb	Estimated K, ksi√in.[c]	Fatigue Crack Length a, in.[d]	Initial K, Ksi√in.[e]	Final Load, lb[d]	Results[f]
1	Deionized H_2O	787	10.0	0.928	11.5	714	Stress cracks
2	Dow Solvent NS-1	787	10.0	0.893	10.6	692	No effect
3	Deionized H_2O	1970	25.0	0.880	25.4	1210	Crevice attack
4	Dow Solvent NS-1	1970	25.00	0.897	26.5	1475	No effect
5	Deionized H_2O 0.12% Fe^{+++} 0.065% Ni^{++}	1970	25.0	0.881	25.4	1190	Pitting, stress cracks
6	Dow Solvent NS-1 0.12% Fe^{+++} 0.065% Ni^{++}	1970	25.0	0.881	25.4	1280	Crevice attack
7	Deionized H_2O	2500	32.0	0.890	32.9	1320	Crevice attack
8	Dow Solvent NS-1	2500	32.0	0.881	32.2	1625	No effect

a. From plate 0.75" thick sensitized 50 hours at 1200 F with a furnace cool.

b. Initially expand from 225 base at 250 F in environments outlined below. All specimens were then expanded in rocking autoclave to 500 F from 300 hours at about 0.3 ppm O_2.

c. Based on nominal crack length of 0.84 in.

d. Measured after SCC test.

e. $$K = \frac{P}{BW^{1/2}}\left[29.6\left(\frac{a}{w}\right)^{1/2} - 185.5\left(\frac{a}{w}\right)^{3/2} + 655.7\left(\frac{a}{w}\right)^{5/2} - 1017\left(\frac{a}{w}\right)^{7/2} + 638.9\left(\frac{a}{w}\right)^{9/2}\right]$$

where:
P = Load, lb.
B = Specimen thickness, in.
W = Distance from loading line to back of specimen, in.
a = Distance from loading line to crack tip, in.

f. Cracks propagated in regions other than at crack tip, e.g., on the notch face or on the fatigue crack face.

stable and well behaved over temperature, impurities in the solution, and length of exposure.

ACKNOWLEDGMENTS

Only the electrochemical work was performed under the direct supervision of R. W. Staehle. The overall program of decontaminating studies is being supervised by Mr. William Worden of Commonwealth Edison. The work at Dow Industrial Services is being directed by Dr. David Harmer, and the corrosion testing work is being conducted by Mr. Tom Boyce.

The information in this paper is taken from licensing submittals prepared by Dow for Commonwealth Edison: (1) Dresden I Chemical Cleaning Licensing Submittal, dated 16 December 1974; (2) Dresden I Chemical Cleaning Licensing Submittal, dated 14 April 1975.

APPENDIX XII

HEALTH PHYSIC REPORTS

6670

R & D REPORT

CRI NUMBER

OFFICE COPY

LABORATORY REPORT CODE: HEH2.13-22-3(2)

DATE ISSUED: March 12, 1974

DOW CHEMICAL U.S.A.

DEPARTMENT: Health & Environmental Research

ACCOUNT NO.: 4202

PROBLEM NUMBER: 0039116

TITLE

HEALTH PHYSICS COVERAGE DURING THE PRELIMINARY FEASIBILITY STUDY OF THE DRESDEN NUCLEAR POWER STATION FOR CHEMICAL CLEANING OF UNIT I BY DOW INDUSTRIAL SERVICE NUCLEAR

AUTHOR(S) SIGNATURE(S)
G. W. Engdahl Gordon Engdahl 3/5/74
REVIEWER'S SIGNATURE
L. G. Silverstein 3/5/74

DESCRIPTIVE SUMMARY WITH CONCLUSIONS: (Include in this space references to data books, and to earlier related reports, patents and publications.)

A preliminary feasibility study was conducted at the Dresden nuclear power station, Morris, Illinois, by Dow Industrial Service, a division of The Dow Chemical Company. The Dresden station is owned and operated by Commonwealth Edison Company of Chicago, Illinois. The study was performed in relation to decontamination of the Unit 1 reactor primary systems. During the study, Dow Health Physics monitored the activities of DIS to ensure that safe practices were followed as stated in the DIS Radiation Protection Manual. Dow Health Physics personnel supplemented Dresden Radiation Protection Services while in the Unit 1 containment vessel in a cooperative effort for the safety of Dow and Suntac Nuclear employees. This report contains a summary of all DIS work permits for entries into the Unit 1 containment vessel. Radiation surveys and purposes for entries are included in the summary.

sjl

DISTRIBUTION:

J. A. Sterling, DIS, 47 Building (for distribution to DIS)

FORM C-40340

PERSONNEL ASSOCIATED WITH DIS ACTIVITIES AT THE DRESDEN NUCLEAR POWER PLANT, MORRIS, ILLINOIS

	DRESDEN FILM BADGE ASSIGNMENTS						
Names	10/22/73-11/4/73	11/5/73-11/18/73	11/19/73-12/2/73	12/3/73-12/16/73	12/17/73-12/30/73	12/31/73-1/13/74	1/14/74-1/28/74
Dow Personnel							
Frausen, F.	5516	5873	6079	6608	6804	7169	7608
Snyder, M.	5517	5872	6078	6689	6805	7170	7609
Sterling, J.	5551	5803	6080	6665	6806	7171	7610
Silverstein, L.	5552	--	6081	6605	6802	7167	7606
Engdahl, G.	5553	--	6082	6688	6803	7168	7607
Boyle, R.	5574	--	--	--	--	--	--
Anders, O.	5575	--	--	--	--	--	--
Bohl, R.	--	--	--	6604	--	7457	--
Roush, G.	--	--	--	6732	--	--	--
Vaccaro, J.	--	--	--	--	7045	--	--
Stevenson, M.	--	--	--	--	--	7407	--
Burroughs, D.	--	--	--	--	--	7408	--
Suntac Nuclear							
Mullett, W.	5571	5794	--	--	--	--	--
Lohkamp, L.	5572	--	--	--	--	--	--
Sawyer, T.	5573	--	--	--	--	--	--
CiannauYi, T.	5518	5903	6155	6442	7015	7158	7611
Stephans, D.	--	5905	6158	6445	7019	7161	7614
O'Loughlin, P.	--	5906	6159	6446	7020	7162	7615
Hadley, D.	--	5907	6157	6444	7017	7163	7616
Simmons, D.	--	5908	6160	6447	7021	7163	7616
Coen, J.	--	--	6156	6443	7016	7159	7612
Mika, R.	--	--	--	6448	7022	7164	7617
Fiedler, R.	--	--	--	6449	--	--	--
Raval, G.	--	--	--	6450	7024	7166	7618
Mullarkey, T.	--	--	--	6761	--	--	--
Williams, P.	--	--	--	--	--	7431	--

DIS WORK PERMITS WHICH INCLUDE RAD SURVEYS

#1	Primary Steam Drum Area
#2	SSG Room D
#3	SSG Room C
#4	SSG Room A
#5	SSG Room B
#19	Radwaste Tunnel (Survey Sheet Needed)
#37	Unloading Heat Exchanger Room
#42	SSG Room Pipe Chase
#43	Crud Samples Loading
#45	Dermin Room B

DIS WORK PERMITS WHICH WERE USED FOR VIDEOTAPING

WorkSheet #	Videotaping
DIS 16 & 41	Unloading Room Videotaping
DIS 17	"A" Clean up Demin Videotaping
DIS 18	"B" Clean up Demin
DIS 21	"A" Clean Up Videotaping
DIS 22	"B" Clean Up Videotaping
DIS 23	"B" Clean Up Room Videotaping
DIS 24	"D" SSG Room Videotpaing
DIS 25	"B" SSG Room Videotaping
DIS 26	"C" SSG Room Videotaping
DIS 28	"A" SSG Room Videotaping
DIS 32	649' Level (emergency condenser) Videotaping and Primary Steam Drum
DIS 34	REDT Room "A" Videotaping
DIS 36	REDT Room "B" Videotaping and Subpile Room Videotaping.

SUMMARY OF DOW INDUSTRIAL SERVICE & SUNTAC ENTRIES IN THE DRESDEN I CONTAINMENT VESSEL

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
11/30/73	Primary Steam Drum	1	Steam drum to check relief valves, level gauges, vents and drains. Radiation survey was also performed.
11/30/37	Secondary Steam Generator D	2	Check S/G drain lines, decontamination bypass lines, pump vent, heat exchanger and S/G vent lines. Radiation survey.
12/5/73	Secondary Steam Generator C	3	Check S/G vents, drain lines, valves, pump vent and heat exchanger vent. Radiation survey.
12/3/73	Secondary Steam Generator A	4	Check S/G steam generator and pump vents, drains, etc. Radiation survey.
12/3/73	Secondary Steam Generator B	5	Check S/G pump vent and drain lines. Radiation survey.
12/6/73	Radiation Waste Facility	7	Check lines 1400-C3-3", S/G blowdown; 4410-4"G1, REDT drains; 4410-4"G1, REDT and clean up room bypass; 4414-2"G1R clean up demineralizer resin transfer.
12/4/73	Demineralizer Room B	8	Check drain lines, vent lines and relief lines.

CONTINUED

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
12/6/73	Secondary Steam Generator Room D	9	Check decon connection flange.
12/6/73	B Cleanup Demineralizer Room and Pump Room	11	Check drains and vents in cleanup room.

Note: DIS Permits 12, 13 and 14 were cancelled due to unavailability of Dresden Radman.

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
12/13/73	Subpile Room 488' Level	15	Check flanges and control rod drives.
12/13/73	Unloading Heat Exchanger Room	16	Videotaping.
12/13/73	A Cleanup Demineralizer Room	17	Videotaping.
12/13/73	B Cleanup Room	18	Entry delayed; batteries on video power pak pack went dead.

CONTINUED

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
12/14/73	Radiation Waste Tunnel	19	Check and tag pipes.
12/14/73	517'6" Level Pipe Run	20	Check connections to down-comer.
12/14/73	A Cleanup Room	21A	Videotaping the pump room and demineralizer tank area.
12/14/73	Demineralizer and Pump Room B	21B	Check pump room wall for possible penetration by 4" pipe.
12/14/73	B Cleanup Demineralizer Room	22	Videotaping started but stopped due to technical difficulties.
12/14/73	B Cleanup Demineralizer Room	23	Videotaping completed.
12/17/73	D Secondary Steam Generator Room	24	Videotaping of entire room.
12/17/73	B Secondary Steam Generator Room	25	Videotaping of room.

CONTINUED

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
12/17/73	C Secondary Generator Room	26	Videotaping of room.
12/20/73	Unloading Heat Exchanger Room	27	Check vents and drains and injection connection.
12/18/73	A Secondary Steam Generator Room	28	Videotaping of room.
12/19/73	Instrument Room (Steam) at 546', 564' & 574' Levels	29	Check equipment drains, vents, etc.
12/19/73	Pipe Chase 517'6" Level	30	Check for various piping.
12/19/73	Demineralizer Room A	31	Check whole area for modifications of systems.
12/19/73	Emergency Condenser and Primary Steam Drum Drum	32	Videotaping the areas.

Note: DIS Permit 33 entry cancelled due to lack of radiation protection.

NTINUED

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
12/ 0/73	Reactor Enclosure Drain Tank Room A	34	Videotaping of room.
1/2/74	Unloading Heat Exchanger Room	35	Visual inspection of piping.
1/3/74	Subpile Room 488' Level and Reactor Enclosure Drain Tank Room B	36	Videotaping of rooms.
1/3/74	Unloading Heat	37	Check and locate vents and drains. Radiation survey in proposed work area.
1/3/74	Subpile Room (Upper Level)	38	Check flanges for control rod drive reactor vessel drains.
1/4/74	Below Primary Steam Drum	39	Check equipment drains.
1/4/74	Secondary Steam Generator D	40	Check pipe run and penetrations through wall and containment.

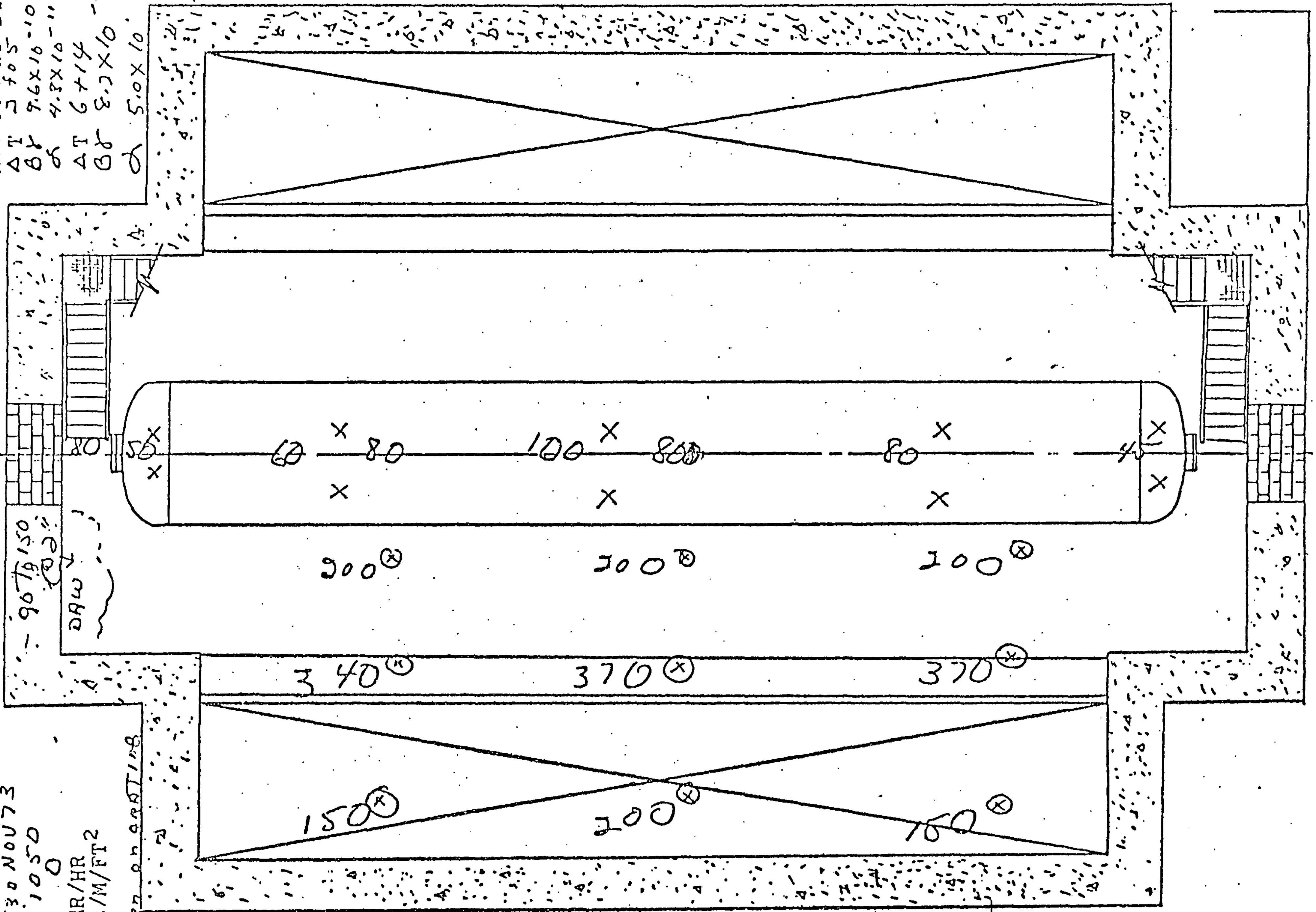
CONTINUED

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
1/7/74	Unloading Room	41	Check steam pressure line and videotaping.
1/8/74	517'6" Pipe Chase Below SSG Rooms A and C	42	Check discharge piping from REDT's and demineralizer bypass loop numbers. Radiation surveys on Line 4415 4" REDT "B" and line 4410 discharge from REDT "A".
1/9/74	Decontamination Room B	42	Check crude samples collected from D-1 reactor vessel which had been stored since 11/4/73. Samples transferred to packing barrel at entrance to personnel hatch. Samples packed and shipping drum surveys for radiation levels.
1/10/74	517'6" Pipe Chase Below SSG Rooms A and C	44	Check drain on main supply to steam generator.
1/15/74	SSG D and Demineralizer Room B	45 & 46	SSG D: Check dimensions on decontamination flange. Survey room for pipe run for solvent flushing. Demineralizer Room B: Check for vents, drains and for hand operated valves. Radiation survey was made.

CONTINUED

Date of Entry	Area Entered	DIS Work Permit No.	Purpose
1/16/74	Unloading Heat Exchanger Room	47	Check heat exchanges, demineralizer water supplies, vents and drains.
1/18/74	Unloading Room, Secondary Steam Generator Room D, Primary Steam Drum and Demineralizer Room A	48	Supervisors' inspection to verify work and inspections previously performed.

GWE:sjl



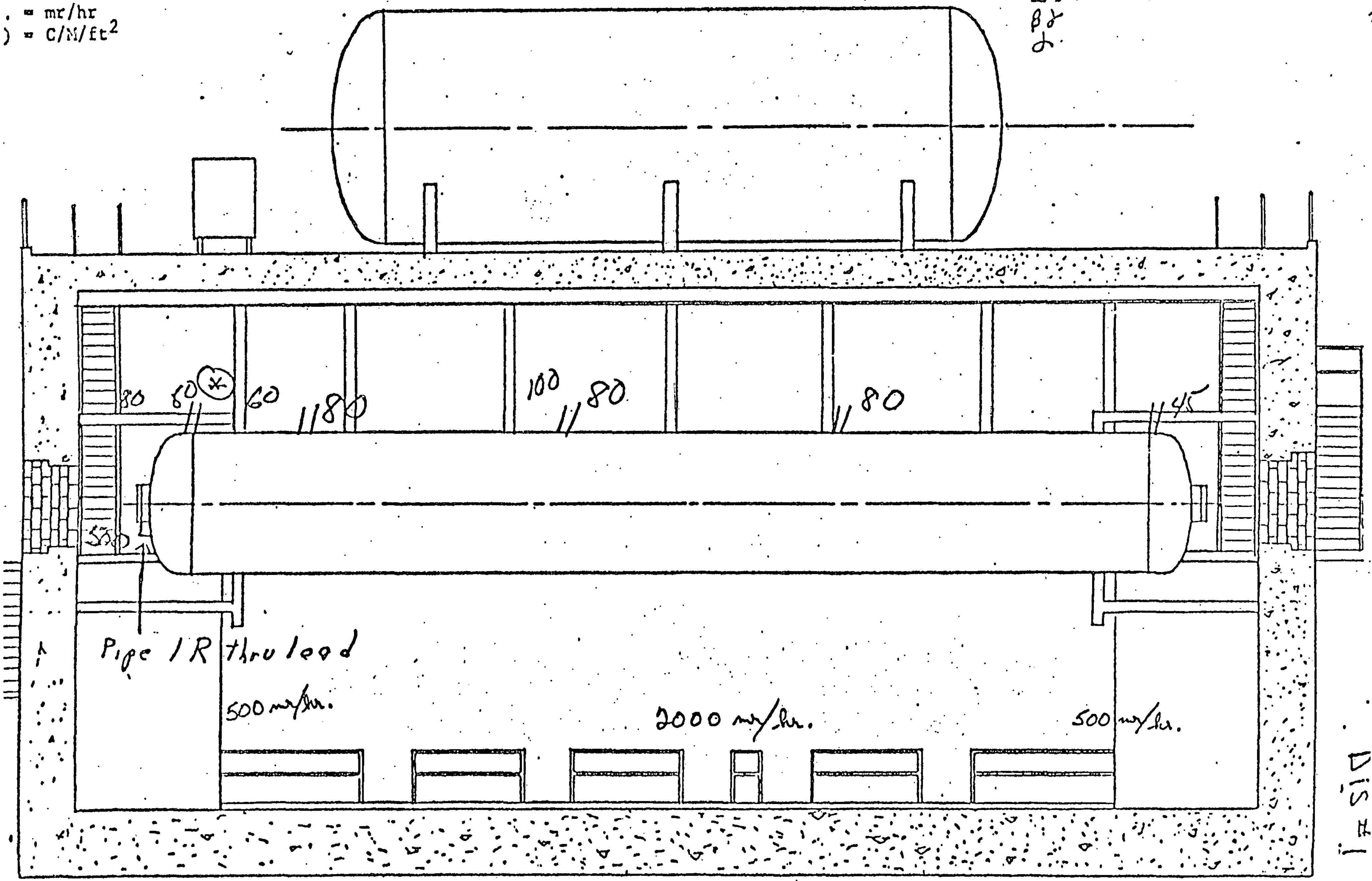
Air Sample Data

SR # 12339
Date 30 NOV 73
Time 1050
F 0
= MR/HR
= C/M/FT^2
Taken on grating

SPHERE PRIMARY DRUM COMPARTMENT
(PLAN VIEW)

R # 1283.
ne 1050
te 30 Nc- 73
T 0
. = mr/hr
) = C/M/ft^2

ΔT 2 + 05
βγ 9.6×10^{-10}
α 4.8×10^{-11}
ΔT
βγ
α

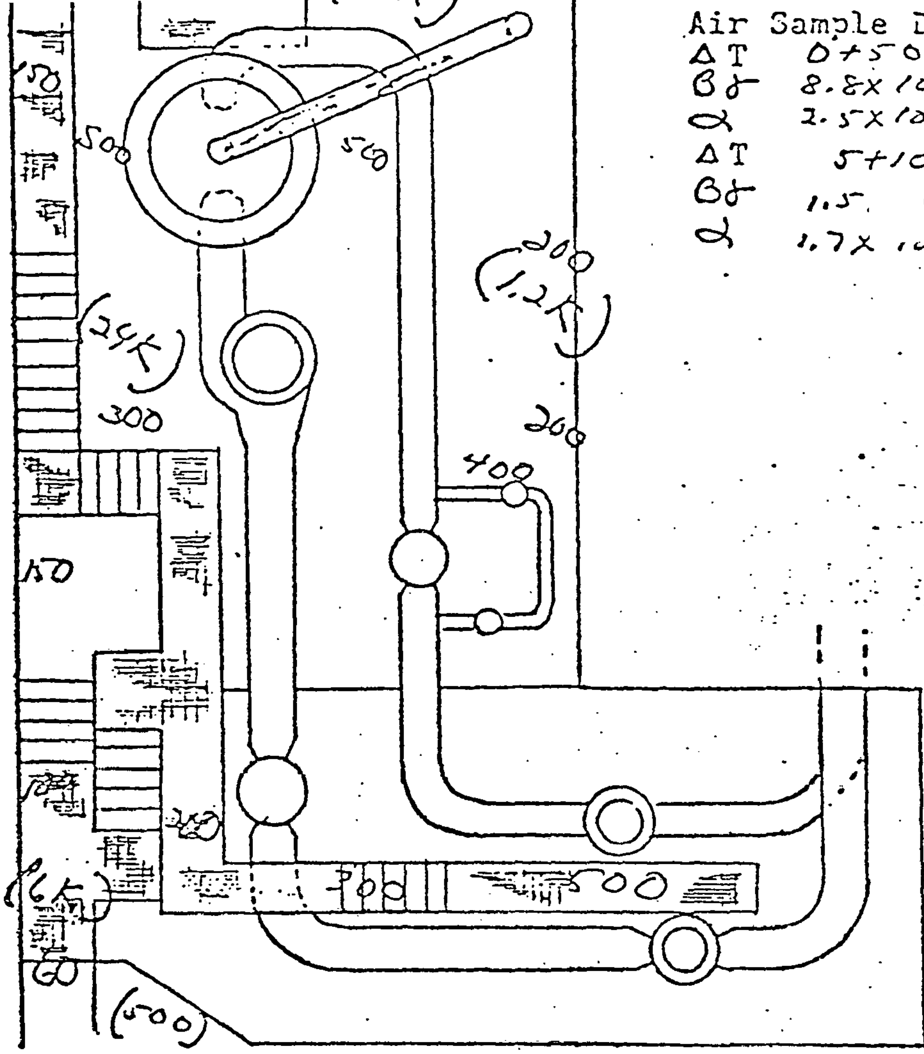


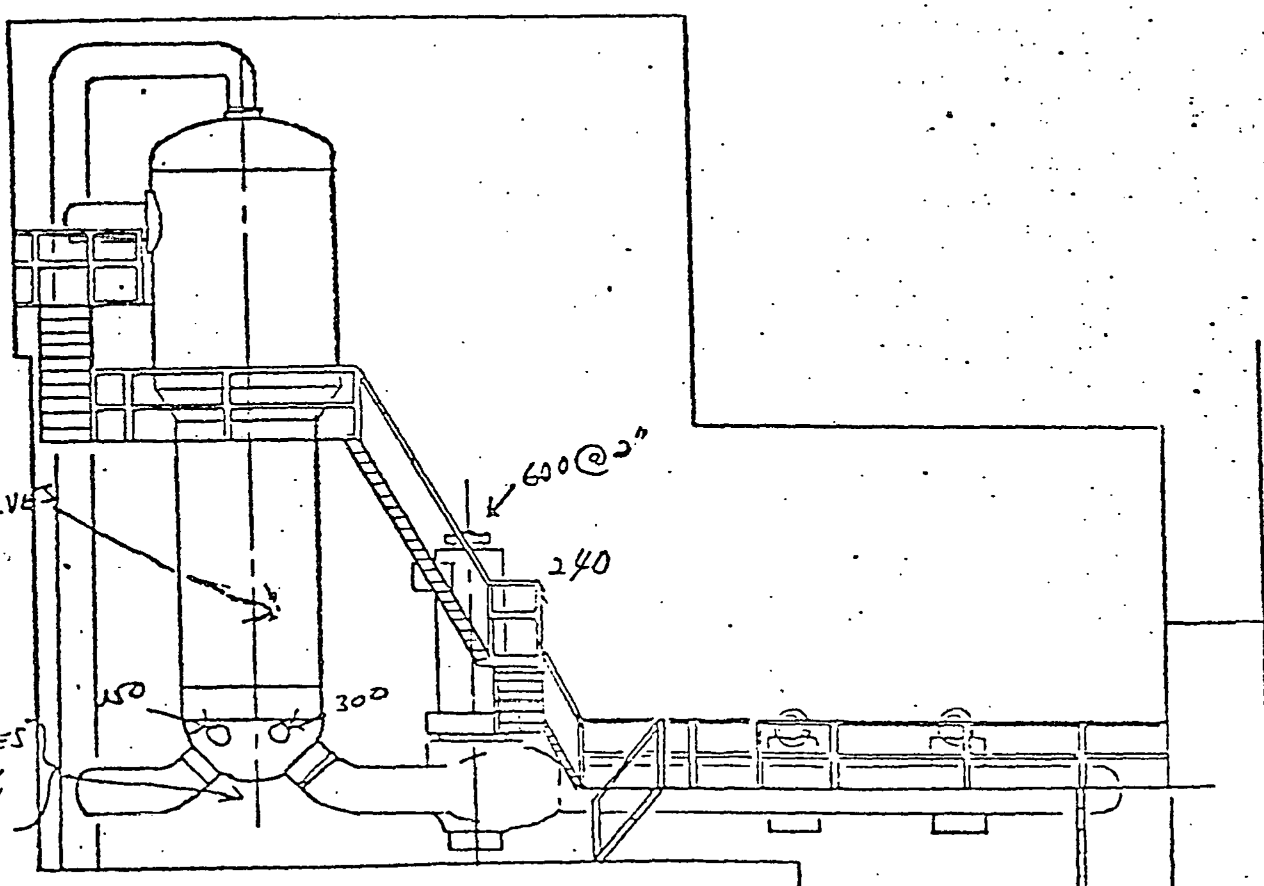
SPHERE EMERGENCY CONDENSER AND PRIMARY DRUM COMPARTMENT
(ELEVATI VIEW)

RSR # 1C0?C
Date 30 NOV 73
Time 1410
MWE 0
No. = MR/HR
( ) = C/M/FT^2

DIS SWP #2

Air Sample Dat
ΔT 0+50
βγ 8.8x10
α 2.5x10
ΔT 5+10
βγ 1.5
α 1.7x10





SPHERE "SW" AND "D" SECONDARY STEAM GENERATOR COMPARTMENT (PLAN AND ELEVATION VIEWS)

RADIATION SURVEY FOR SECONDARY STEAM GENERATOR ROOM "D"

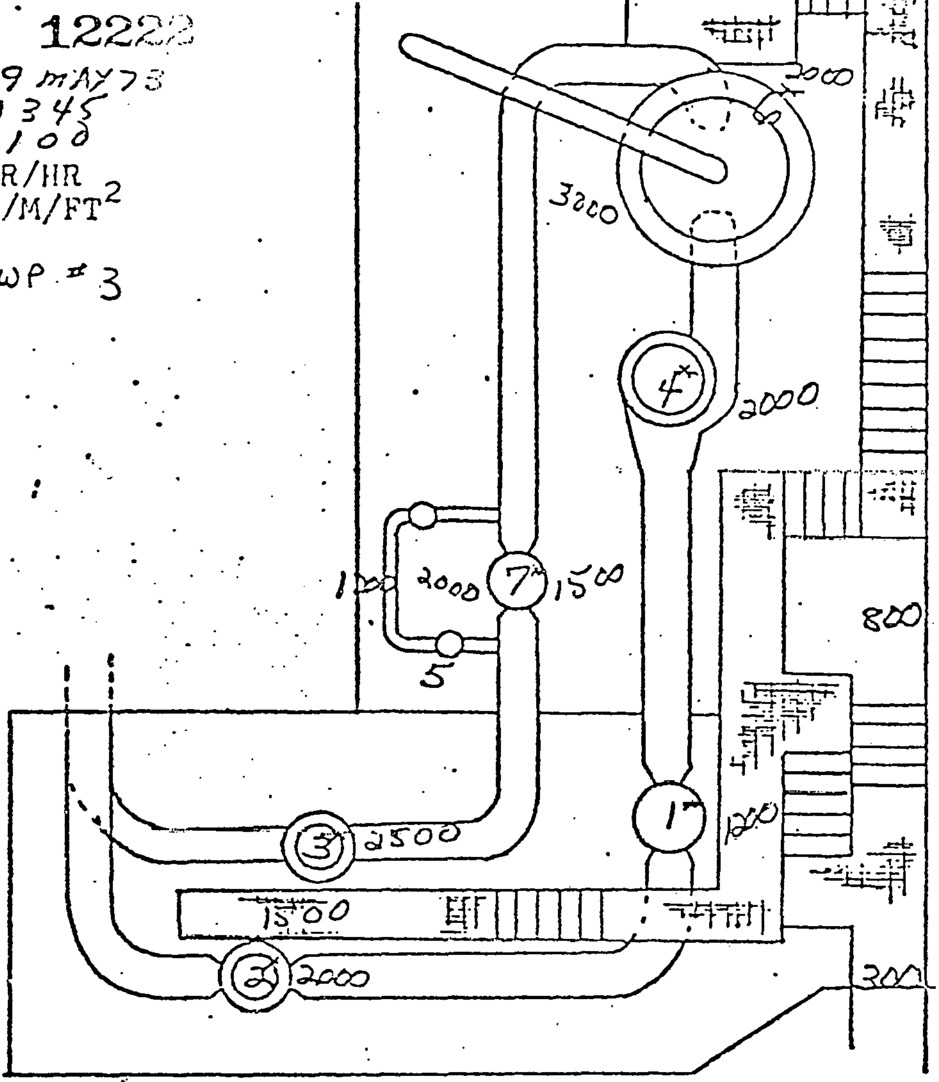
Description/Location of Measurement*	Measurement (mr/hr at 2")
1. Hand hole cover (right).	300
2. Hand hole cover (left).	100
3. Bottom drain (right).	3000
4. Bottom drain (left).	1500
5. Primary side vent (right).	3000
6. Primary side vent (left).	3000
7. To left of primary side vent, 1" pipe capped off (secondary side drain).	600
8. Pump top at vent.	600
9. Valve to right of pump top at vent.	300
10. Decon flange.	400
11. Suction side, decon flange.	600

*DIS Work Permit #2.

RSR # 12222
Date 9 MAY 73
Time 1345
MWE 100
No. = MR/HR
( ) = C/M/FT^2

DIS SWP # 3

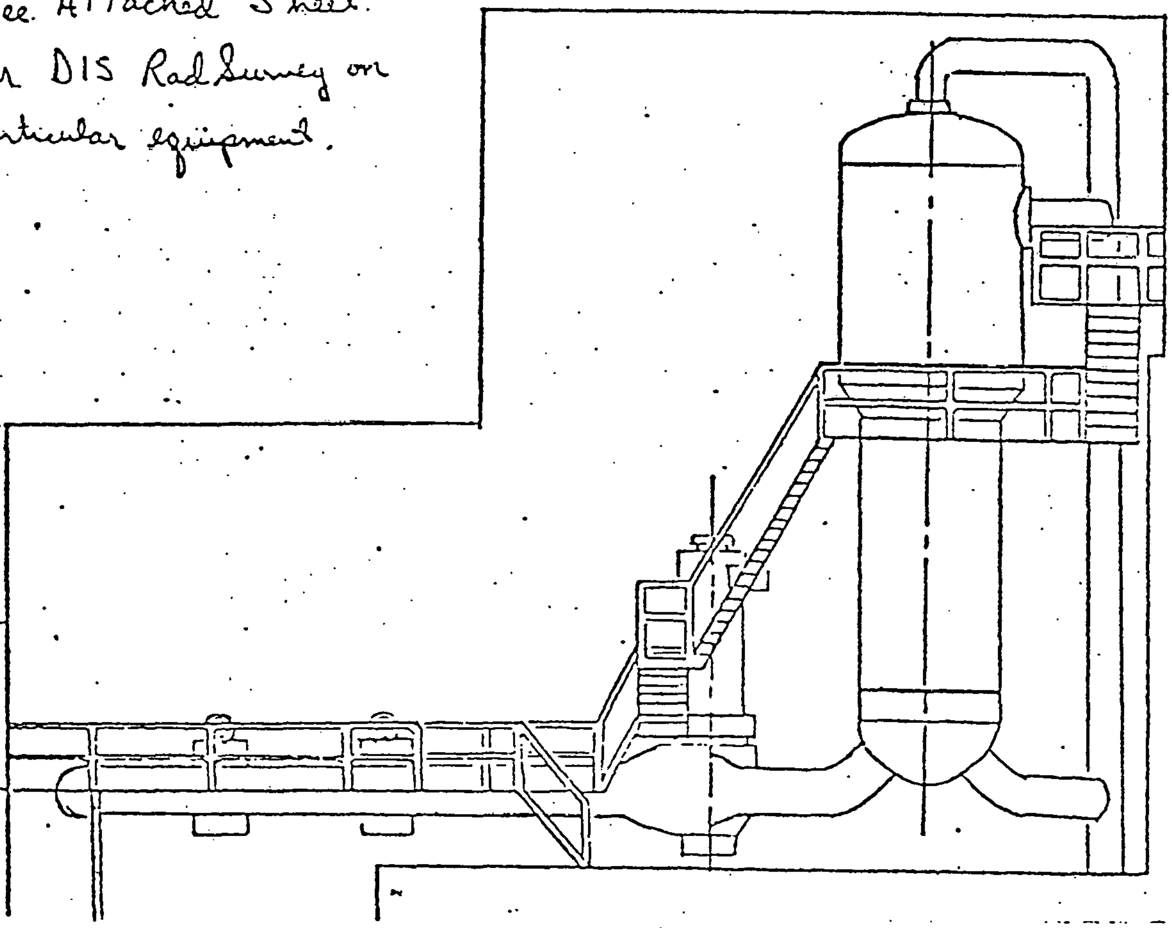
Air Sample Data
ΔT
$\beta\gamma$
α
ΔT
$\beta\gamma$
α



* Smears of s

1. (4 000)
2. (250 000)
3. (10000)
4. (5 000) Pum
5. (40 000)
6. (100 000) manh
7 (10 000)

*See Attached Sheet for DIS Rad Survey on particular equipment.



SPHERE "C" SECONDARY STEAM GENERATOR COMPARTMENT (PLAN AND ELEV[illegible]ON VIEWS)

RADIATION SURVEY FOR SECONDARY STEAM GENERATOR "C"

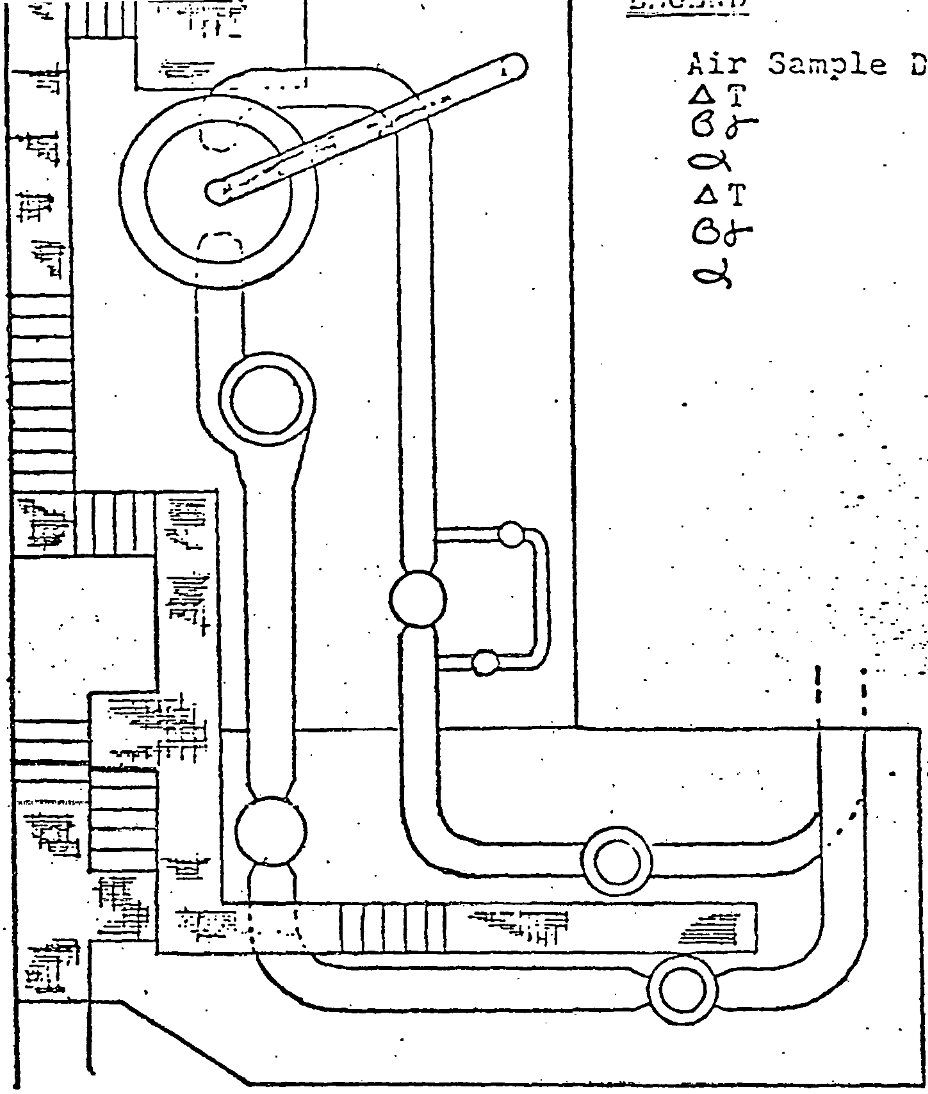
Description/Location of Measurement*	Measurement (mr/hr at 2")
1. Hand hole cover (right).	1600
2. Hand hole cover (left).	900
3. Bottom drain (right).	2500-3000
4. Bottom drain (left).	2500-3000
5. Primary side vent (right).	1300
6. Primary side vent (left).	1100
7. Secondary side drain.	850
8. Pump vent on top.	500
9. Pump valve to right of vent.	300
10. Decon flange (insulation absent).	650

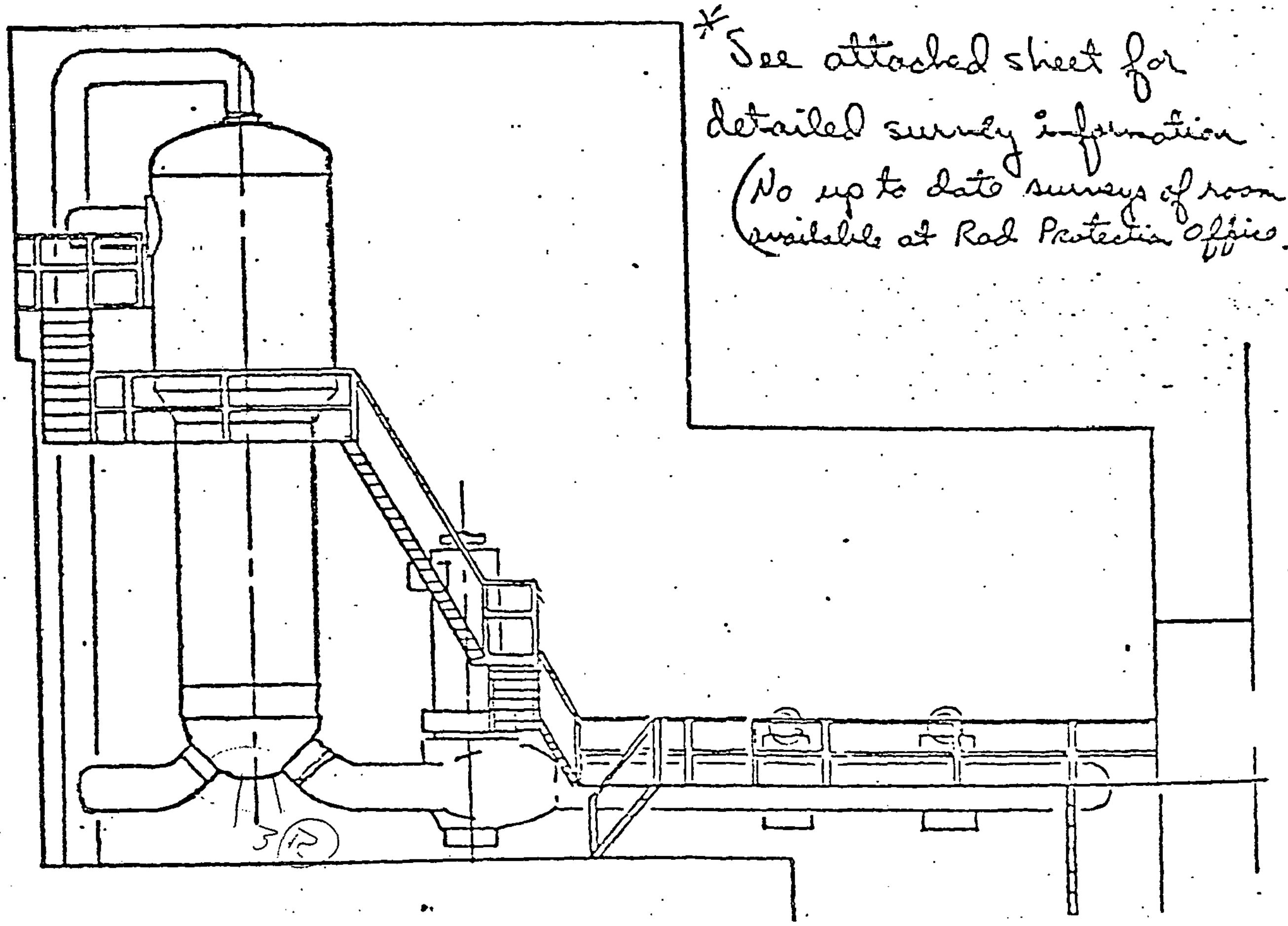
*DIS Work Permit #3.

RSR #A
Date 12/3/73
Time
MWE
No. = MR/HR
( ) = $C/M/FT^2$
DIS Work Permit # 4

LEGEND

Air Sample Dat
Δ T
$\beta\gamma$
α
Δ T
$\beta\gamma$
α





*See attached sheet for detailed survey information (No up to date surveys of room available at Rad Protection Office.

SPHERE "A" AND "D" SECONDARY STEAM GENERATOR COMPARTMENT
(PLAN AND ELEVATION VIEWS)

RADIATION SURVEY FOR SECOND STEAM GENERATOR "A"

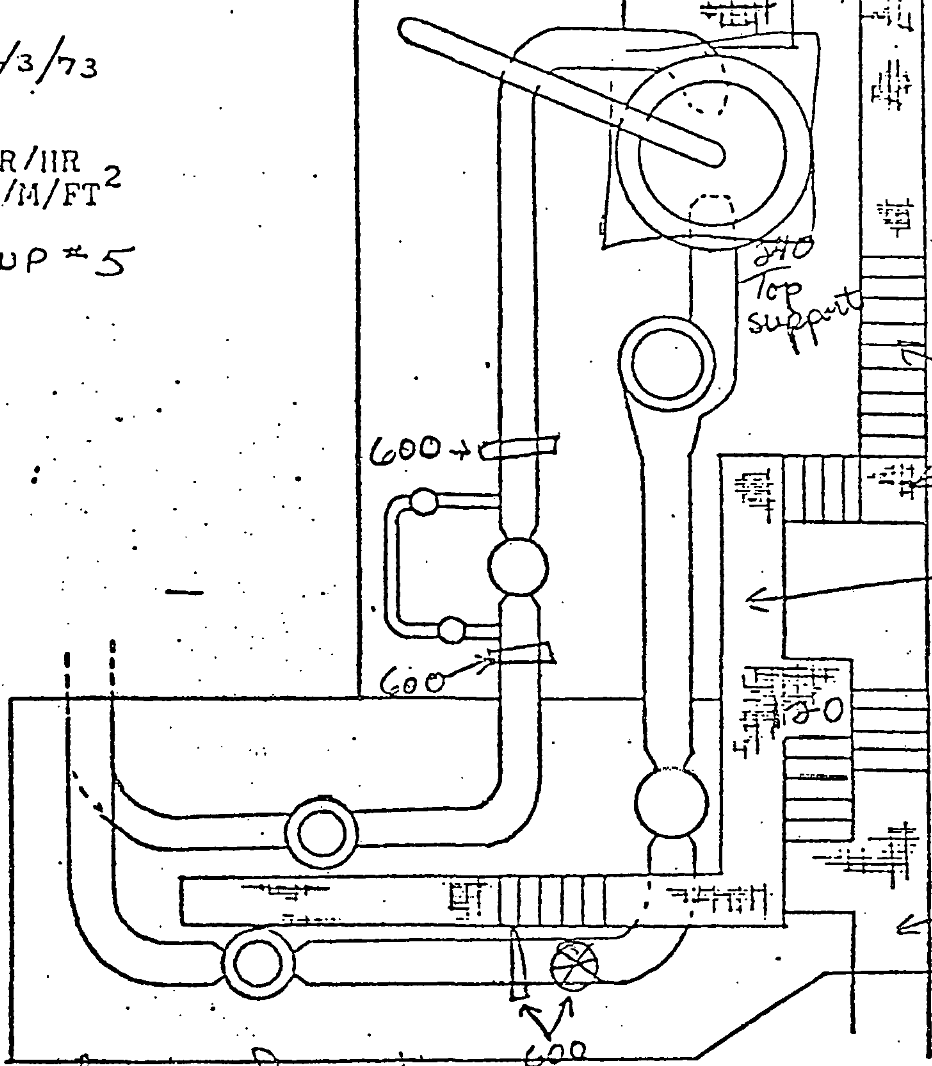
Description/Location of Measurement*	Measurement (mr/hr at 2")
1. Hand hole cover (right).	22
2. Hand hole cover (left).	110
3. Bottom drain.	3900
4. Bottom drain (left).	3000
5. Primary side vent (right).	1200
6. Primary side vent (left).	1600
7. Secondary side drain.	600
8. Pump vent on top.	20
9. Pump valve to right of vent.	(missed)
10. Decon flange in bypass loop.	800

*DIS Worksheet #4.

RSR #
Date 12/3/73
Time
MWE
No. = MR/HR
) = C/M/FT^2

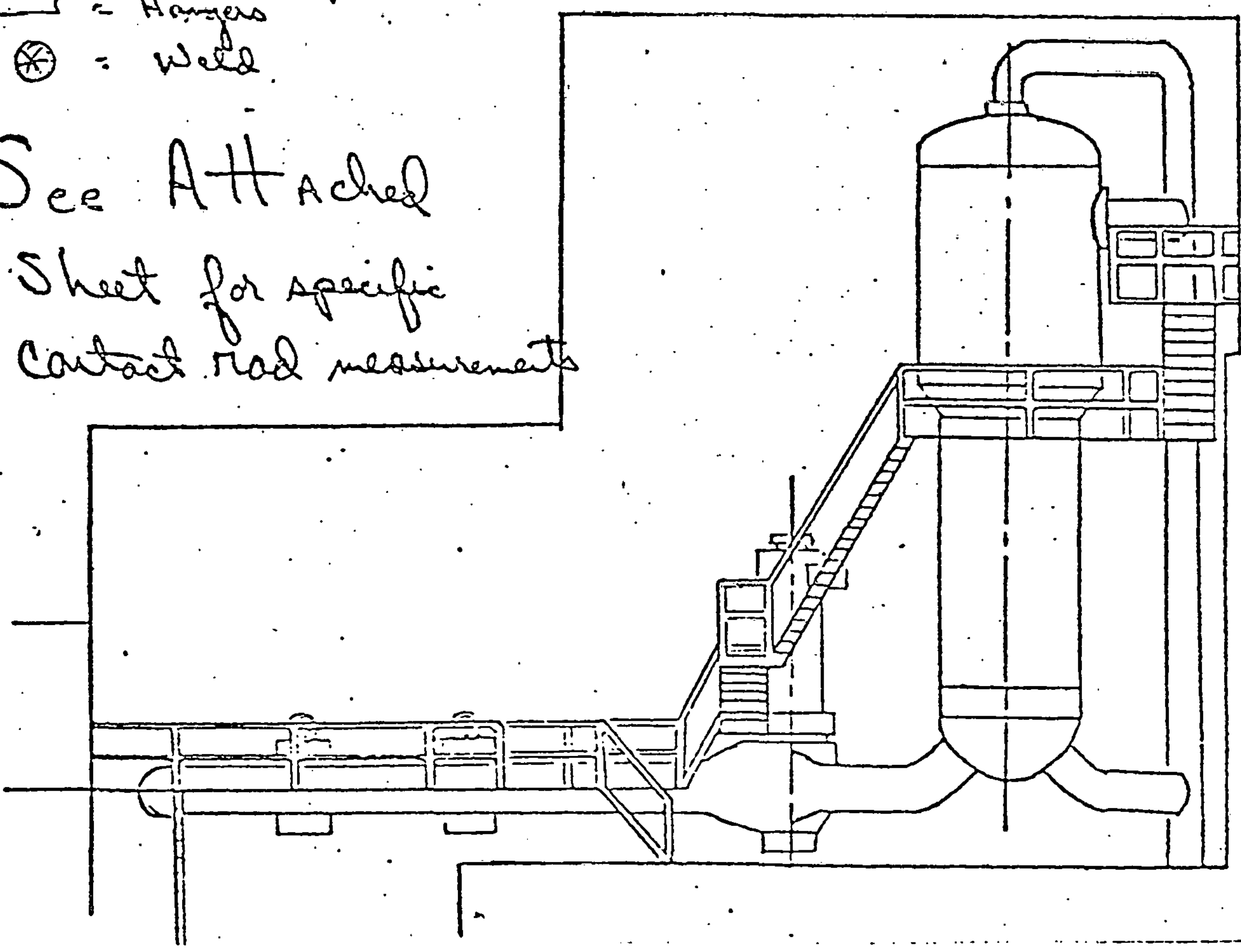
DIS SWP #5

Air Sample Data
ΔT
$\beta\gamma$
α
ΔT
$\beta\gamma$
α



In Service Inspection
= Hangers
⊗ = Weld

See Attached Sheet for specific Contact rad measurements

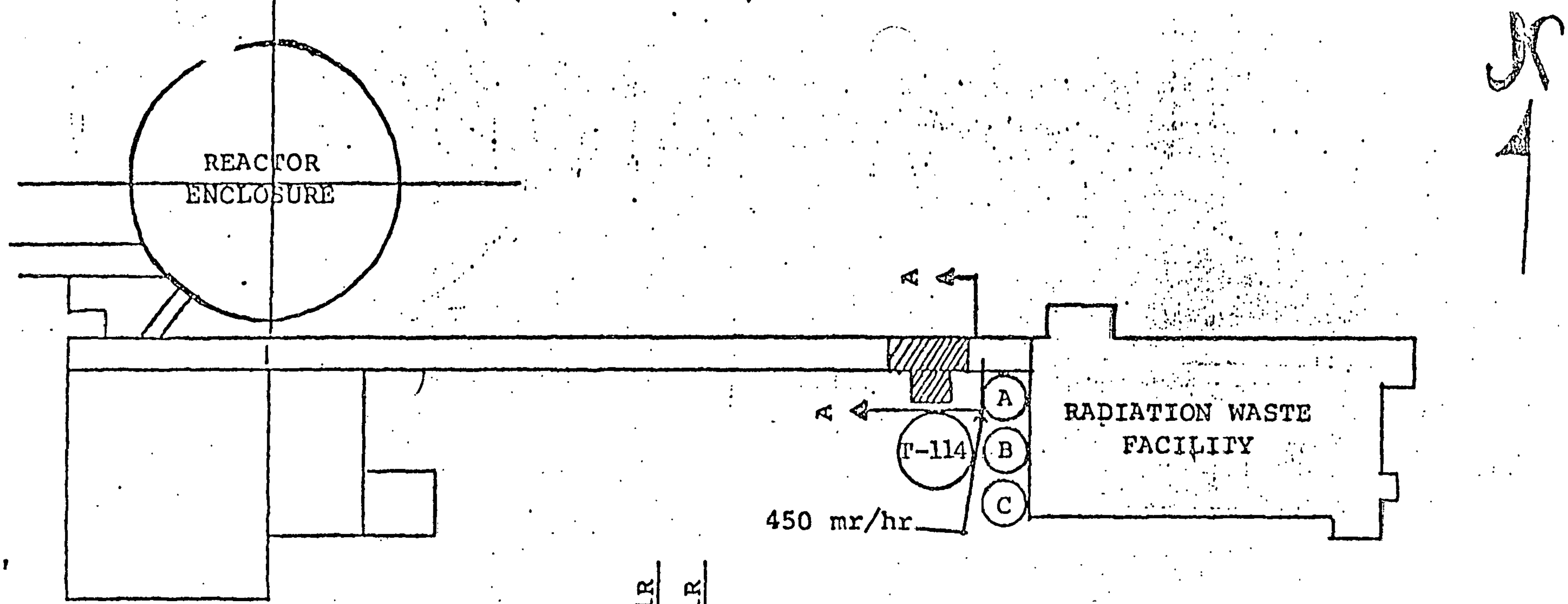


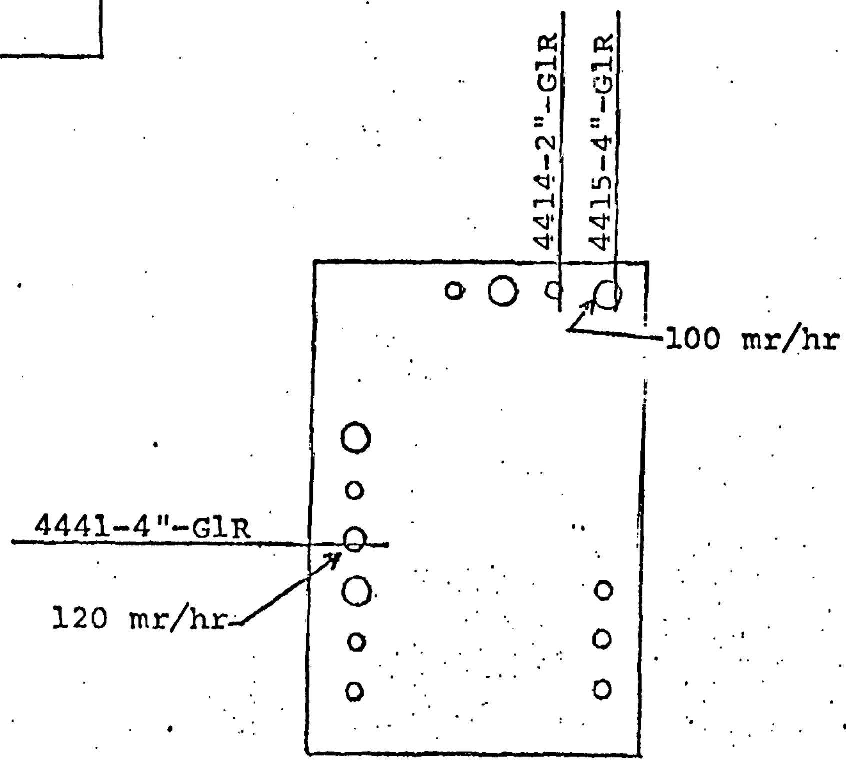
SPHERE "B" AND "C" SECONDARY STEAM GENERATOR COMPARTMENT
(PLAN AND ELEV-TON VIEWS)

RADIATION SURVEY FOR SECONDARY STEAM GENERATOR "B"

Description/Location of Measurement*	Measurement (mr/hr at 2")
1. Hand hole cover (right).	900
2. Hand hole cover (left).	800
3. Bottom drain (right).	3500
4. Bottom drain (left).	3500
5. Primary side vent (right).	500
6. Primary side vent (left).	550
7. Secondary side drain.	800
8. Pump vent on top.	250
9. Pump valve to right of vent on top.	150
10. Decon flange in bypass loop.	(None There)
11. Primary feed drain line.	1200

*DIS Worksheet #5.



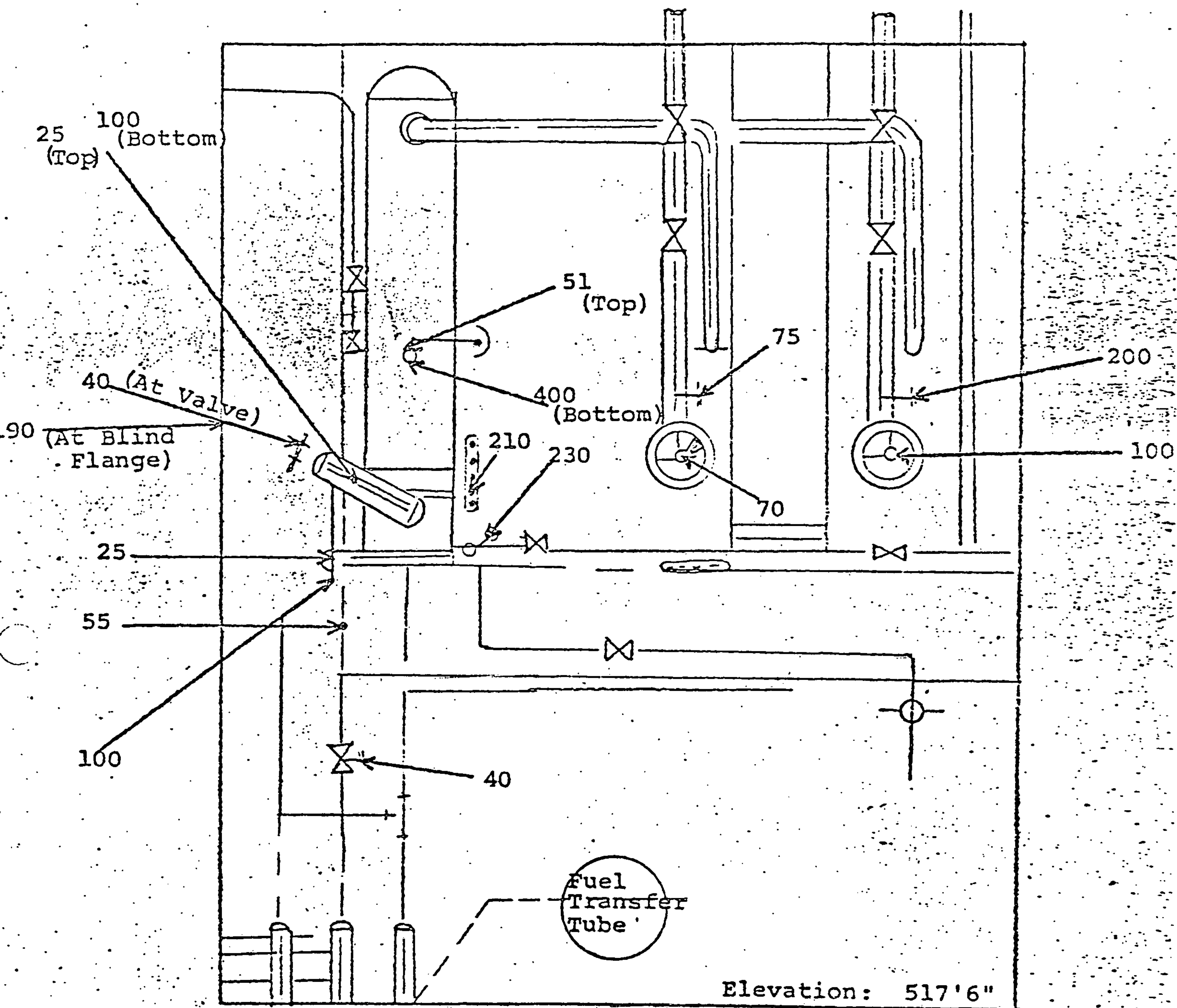


SECTION AA

RADIATION READINGS -
RADIATION WASTE TUNNEL

DIS SWP # 19

REACTOR UNLOADING ROOM



All readings are in mr/hr.

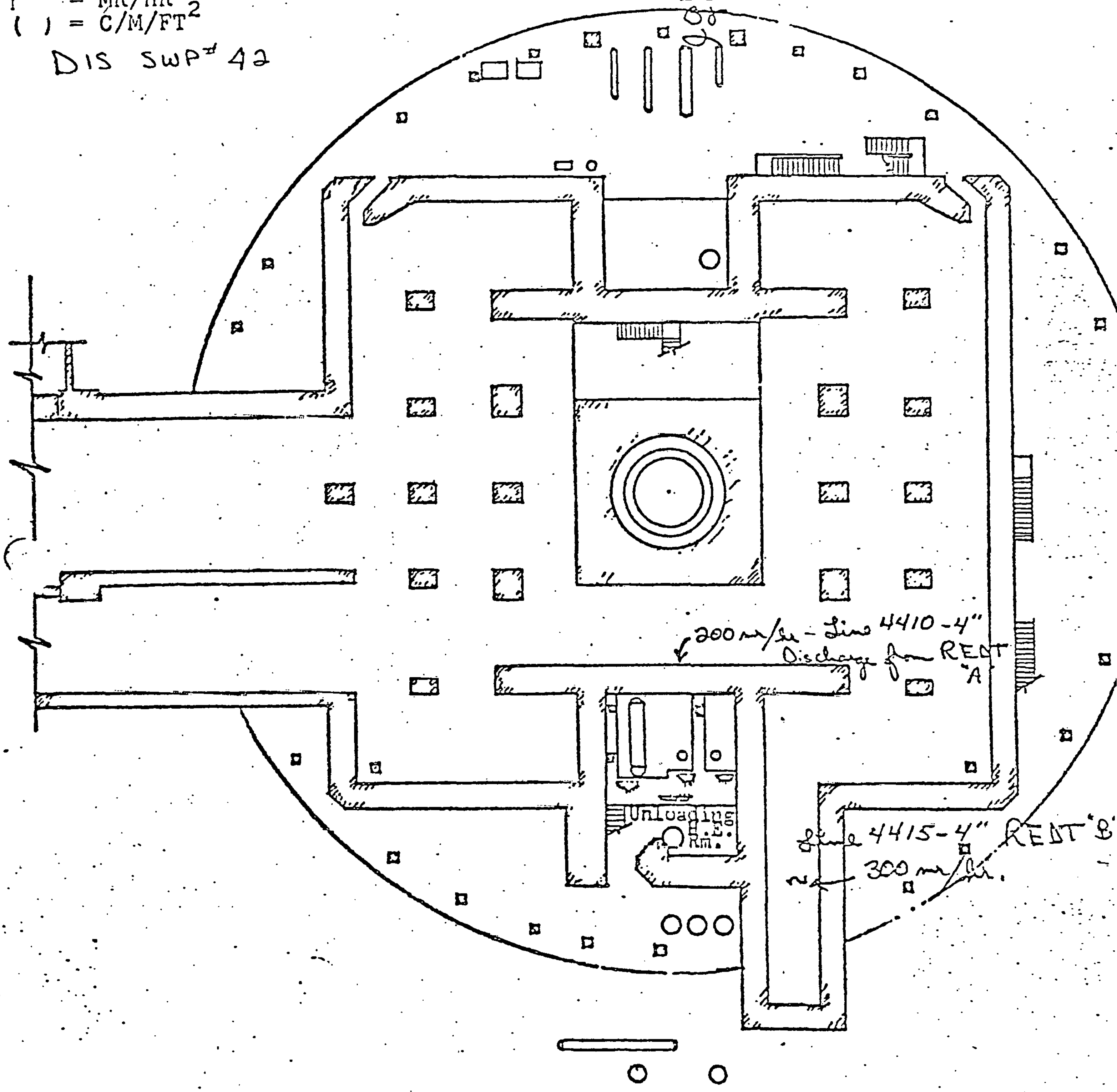
DIS SWP # 37

LEGEND

RSR #
Date 1/8/74
me
RWE
! = MR/HR
( ) = $C/M/FT^2$

DIS SWP# 42

Air Sample Data
Δ T
$\beta\gamma$
γ
Δ T
$\beta\gamma$
γ



Reactor Enclosure
Plan Elev. 517'-6"

D-1 REACTOR CRUDE SAMPLES*

Sample Holder No.	At Contact (mr/hr)**
1	650
2	2000
3	2400
4	1500
5	500

*Collected on November 4, 1973; stored in decontamination room "B" until January 9, 1974.

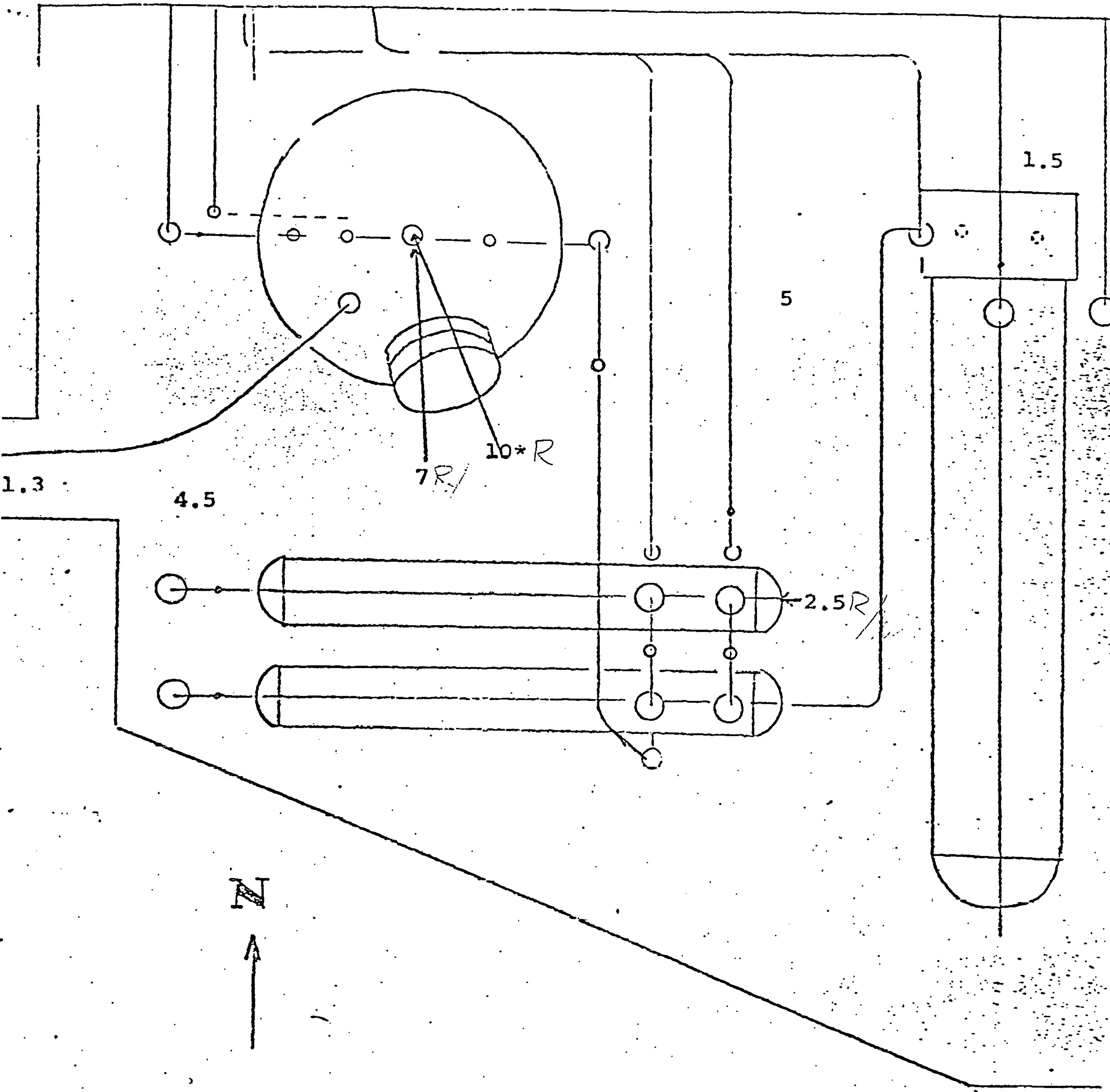
**Highest reading at contact with sample cup.

Note: With 3/8" of lead shielding wrapped around a 6" steel pipe:

Reading (mr/hr)	Location
75	Surface of 55 gallon drum.
8	Three feet from surface of 55 gallon drum.

DIS SWP* 43

DEMINERALIZER ROOM "B"



All readings are in mr/hr.
*Below floor 2'.
DIS SWP # 45

R & D REPORT

DOW CHEMICAL U.S.A.

RESTRICTED: for use within The Dow Chemical Company only.

6640

LABORATORY REPORT CODE: NBH2.13-22-3(1)

DATE ISSUED: December 5, 1973

DEPARTMENT: ealth & Environmental Research

ACCOUNT NO.: 998

PROBLEM NUMBER: 0039116

TITLE: RADIATION SAFETY COVERAGE OF SLUDGE SAMPLING OPERATION AT DRESDEN "I" NUCLEAR POWER STATION - (OWNED AND OPERATED BY THE COMMONWEALTH EDISON COMPANY, CHICAGO, ILLINOIS)

AUTHOR(S) SIGNATURE(S): G. W. Engdahl Gordon Engdahl 12/11/73

REVIEWER'S SIGNATURE: L. G. Silverstein L G Silverstein 3-26-74

DESCRIPTIVE SUMMARY WITH CONCLUSIONS: (Include in this space references to data books, and to earlier related reports, patents and publications.)

The Industrial Hygiene Section of Health and Environmental Research group, as requested by Dow Industrial Services, supplied health physics coverage of the sludge sampling operation at the Dresden "I" nuclear power reactor. During this operation, five samples were collected and there was no radiation overexposure to anyone involved in this operation. The sludge samples collected will confirm the solvent make-up to be used in decontamination of the primary system of the Dresden "I" Nuclear Plant.

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Central Report Index (5), 566 Building

dda

FORM C-40340

PURPOSE

Dow Industrial Services requested health physics assistance in the planning and execution of a sludge sampling procedure at the Dresden "I" nuclear power reactor.

RESULTS

1. Five sludge samples were collected with no undue personnel exposure and no unanticipated contamination problems.

2. Minor mishaps during the dismantling and cleanup phase at the end emphasized the need for tighter HP control until an entire procedure is finished.

3. All survey measurements and a personnel exposure summary are included.

DISCUSSION AND DATA

The Industrial Hygiene Section of Health and Environmental Research Department was requested by Dow Industrial Services (DIS) to provide radiation safety (health physics, HP) coverage of the sludge sampling operation at the Dresden "I" nuclear power station. DIS planned to collect samples from the floor of the reactor pool and the bottom of the reactor vessel. The radiation safety group's purpose in this project was two-fold. One, to provide input into the planning of the operation and write-up of the procedure. This would insure all radiation hazards had been considered and the operation was planned, safely, from a radiation exposure standpoint. Secondly, to maintain a close surveillance of the sampling from start to finish and to document radiation survey information. The fulfillment of these purposes would insure no undue human exposure to radiation and would document radiation exposure to those people involved in this project.

The nature of nuclear reactors and radioactive fuels used leads to activation of any trace elements which come into contact with the primary system. During refueling operations there is also a chance for portions of the fuel and fuel rods to escape into the primary system. This project involved collecting samples of the debris which had built up on the bottom of the reactor vessel. These samples were to be removed hydraulically and would help to confirm the adequacy of the solvent make-up to be used in decontamination of the primary system of the Dresden "I" plant. The samples collected could contain potentially dangerous quantities of radioactive material. Precautions were placed in the operating procedure to insure a radiologically safe sampling operation. These precautions included:

1. Emergency action in the case of collecting a highly radioactive sample.

2. Limitations on the number of people allowed near the reactor vessel sludge sampling device (RVSSD) while sampling.

3. Guidelines for communication between the pump operator of the RVSSD and radmen monitoring radiation levels in the sample collection system.

4. Limiting all samples collected to 1 R/hr (roentgens/hour) or less at the surface of the sample container.

5. Planned radiation survey of work areas before, during and after sludge sampling.

6. Special clothing, film badges and dosimeters were to be worn as designated by the Dresden Radiation Protection Office. Dow film badge and two pencil dosimeters were added to the required equipment for entry into the containment vessel.

During the actual sampling operation radiation levels were measured in all work areas where Southern Nuclear or DIS people were involved. The RVSSD was placed on the 584 foot level at

11:30 a.m. on November 3, 1973. During the afternoon the RVSSD system was assembled with piping and pump in appropirate positions as illustrated in Figure 1. An initial radiation survey was taken by the radiation safety group. The radiation levels are given in Figure 1.

The radiation measurements were made with a Victoreen Cutie Pie and a radgun with remote chamber. Both of these instruments are ionization chambers and were calibrated before use. One calibration was performed at the Midland location of The Dow Chemical Company and a second at the Dresden "I" calibration facility. The results of the Dresden calibration are included in the appendix of this report. The instruments were found to be in agreement with known dose rates. The radgun remote chamber was placed in a polyethylene bag and mounted between the legs of the filter mechanism of the RVSSD at a location of about 8 inches from the ball valve. This made it possible to monitor the build up of activity in the sampling cup from a distance of 70 feet. The remote cord from the chamber extended from inside the lead pig out to the catwalk as illustrated in Figure 1.

During the afternoon of November 3, 1973, the RVSSD system was checked for leaks and the flow control valve was calibrated.

Radiation safety coverage was carried out as planned in this dry run to check planned procedures. The filter mechanism and sample holder were contained in a lead pig two inches thick and as tall as the entire filter mechanism. This provided a 20 fold reduction factor in any radiation coming from the filters. Absorbent paper was placed wherever leakage from the RVSSD was possible to limit surface contamination in the work areas. Lead bricks were used to construct a small storage cave for the sample holders. The cave was lined with a polyethylene sheet and absorbent paper. After observing Southern Nuclear Engineering (SNE) personnel handle the sample cup holder with remote handling equipment, it was decided by the radiation safety group that it would not be necessary to use shielding during removal and capping of the sample. Radiation survey data gathered during the dry run are shown in Table 1 below.

TABLE 1. Radiation Levels During Assembly and Testing of Southern Nuclear Reactor Vessel Sludge Sampler, 11-3-73.

Description	Radiation Intensity mr/hr
Assembly table on 566 foot level	1
584 foot level:	
Entrance at top of stairway	2
At railing 5 feet from entrance	2.5
Radgun within shielding, before startup	0.2
Contact with filter, within shielding, before startup	3
After running pump for 5 minutes:	
Radgun	0.4
Contact with filter	25
After final testing and flow calibration, contact with filter	95

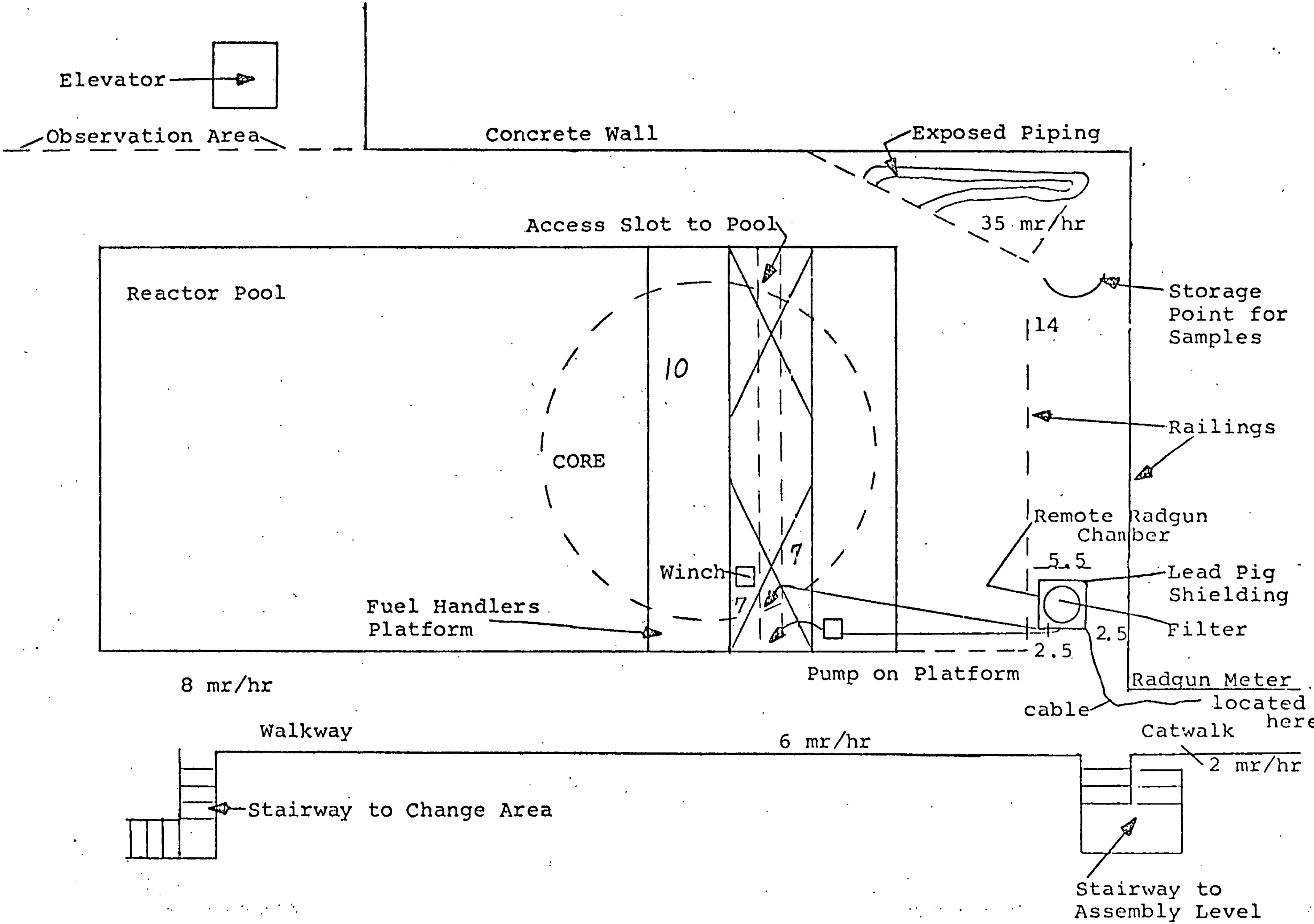

FIGURE 1: Initial Radiation Survey on 584' Level - 11-3-73

From this test run it was apparent that some activity was suspended in the pool water and could be expected to collect on the 5μ filters of the filter assembly during the sampling on the next day.

On November 4, 1973, DIS and Southern Nuclear personnel arrived at the 584 foot level in SWP clothing at 9:00 a.m. The Dresden radman and other personnel were also on location to help with the sampling operation. The Dresden fuel handlers operated the TV camera, lighting and the fuel handlers platform. The radiation safety group measured radiation levels in the work area on the 584 foot level and found the values to agree with those from the previous day as given in Figure 1. The filter assembly as measured with the Cutie Pie at contact had dropped to 55 mr/hr. The sample holder as measured with the radgun remote chamber read 1 mr/hr and external to the lead pig radiation levels were 2 mr/hr. The four area alarms were checked at 9:55 a.m. Three were found to work and the alarm level was set at 40 mr/hr. The radiation survey log during actual sampling was as follows:

NOTE: all readings were made with Victoreen Cutie Pie, unless stated otherwise.

1130 - Pump started; 65 mr/hr at surface of filter.

1200-1300 - Fuel handlers on break.

1325 - Radgun at sample holder #1 filled with sample measured 50 mr/hr.

The filter measures 175 mr/hr at the surface.

1330 - Sample holder #1 removed, measured 150 mr/hr two inches from the open top of the water and crud sample.

1330 - Measured 45 mr/hr approximately 10 inches from sample holder #1 with the cover in place.

1425 - The second sample holder is in place and filter assembly now measures 550 mr/hr at contact.

1425 - Sample holder #2 with cover on measures 700 mr/hr at contact.

1505 - Sample holder #3 has been filled with a sample from a third location. Sample holder #3 with cover on measures 75 mr/hr 10 inches from the surface.

1507 - Sample holder #3 with cover on measures 750 mr/hr at contact.

1510 - The filter measures 800 mr/hr at contact.

1510 - The flex hose (intake side) on the fuel handlers platform is measuring 50 mr/hr, background included, at contact.

1510 - Sample holder #4 in place and ready to be filled; measures 12 mr/hr with the radgun, (background with remote monitor).

1635 - Sample holder #4 filled capped, measured 600 mr/hr at contact.

1635 - Sample holder #4 with sample enclosed measured 60 mr/hr at approximately 10 inches from the surface.

1640 - Sample holder #5 is placed into position and lead door replaced. Background on the remote radgun measures 17 mr/hr before sampling from the floor of the reactor

1718 - Sample holder #5 filled and capped measures 40 mr/hr at 10 inches from the surface.

1718 - The filter after final sample (#5) had been collected measured 2500 mr/hr at contact.

1725 - Background radiation level inside the lead pig as measured with the remote chamber radgun was 23 mr/hr with no sample holder present.

After the sludge sampling was completed all the samples were stored in the lead cave. The Southern Nuclear personnel then started to disassemble the RVSSD system. All equipment brought to the 584 foot level had been logged in on November 3, 1973, and removal of equipment to the 566 foot level was now logged out. The log-in and log-out insured that all equipment was accounted for.

It became apparent that the safety awareness of the group diminished during the disassembly of the RVSSD system. One example of this was that the final filter on the discharge end of the RVSSD was pulled from the reactor pool and it set off the area alarm. Second, a contaminated section of flexible hose was removed to the 566 foot level before bagging. This was noticed quickly and corrected. The hose was placed in a polyethylene bag and stored with the filter assembly at the 584 foot level. The stainless steel pipe sections were wiped dry and removed to the Southern Nuclear work area on the 566 foot level with the exception of the bottom section. The bottom section was contaminated in the area of the flexible sampling head. The head was bagged and placed along the railing near the lead pig on the 584 level. The sample holders were all placed inside the lead pig and the door bolted on. All personnel working on this sampling project left the containment for the day. A representative of the Southern Nuclear Engineering Company assisted the Dresden crane operator the following day (11-5-73) in the removal of the lead pig and other contaminated equipment to a storage room at a lower level.

The radiation levels on the samples collected during this project are summarized in Table 2. Radiation exposure data for the personnel involved in the RVSSD project is summarized in Table 3. All exposures were well below permissible limits.

TABLE 2

Data on Samples Collected at Dresden "I", November 4, 1973

Sample Holder Number	mr/hr at 10 Inches
1	45
2	28*
3	75**
4	60
5	40

*Based on calculation, 700 mr/hr at 2 inches measured.

**Based on calculation, 750 mr/hr at 2 inches measured.

TABLE 3

Exposure During Sludge Sampling For
November 3, 1973 & November 4, 1973

Name	Dresden Badge Number	Estimated Pen Readings*	Film Badge Results
J. Sterling	5551	51 mr	Defective
L. Silverstein	5552	62 mr	70
G. Engdahl	5553	48 mr	50
M. Snyder	5578	26 mr	Minimal
F. Frauson	5579	17 mr	Minimal
W. Mulitt**	5571	68 mr	110
L. Lohkamp**	5572	68 mr	110
T. Sawyer**	5573	62 mr	120
R. Boyle	5574	33 mr	50
O. Anders	5575	39 mr	40

*Based on Dresden Pencil Dosimeters.

**Southern Nuclear Engineering Personnel.

Calibration at Dresden Station Calibration Facility

Victoreen "Cutie Pie"

Field (mr/hr)	Reading (mr/hr)
250	250
450	450
1000	1000
5000	6000

Victoreen "Radgun" remote chamber attachment

Field (mr/hr)	Reading (mr/hr)
5000	5000
1000	1000
450	550
250	325
100	125

APPENDIX XIII

TOXICOLOGICAL PROPERTIES AND INDUSTRIAL HANDLING HAZARDS OF DOW SOLVENT NS-1

TOXICOLOGY RESEARCH LABORATORY

DOW CHEMICAL U.S.A.

FILE
HET DR-0160-5001-(1)

SUBMITTED BY	CHARGE	DATE	K NUMBER
T. Boyce	580-0120022	12-11-75	DR-0160-5001

ICOLOGICAL PROPERTIES AND INDUSTRIAL HANDLING HAZARDS OF:

DOW SOLVENT NS-1

REPORTED BY: J. Henck, C. Vaughn, P. A. Keeler CHECKED BY: K. J. Olson

INFORMATIVE SUMMARY WITH CONCLUSIONS BASED ON THE SAMPLE RECEIVED. ADDITIONAL INFORMATION INCLUDING THE EFFECTS OF REPEATED EXPOSURE MAY BE REQUIRED AS SPECIFIC USES AND FORMULATIONS ARE DEVELOPED OR IF PROCESS CHANGES OCCUR.

A sample of Dow Solvent NS-1, a clear colorless liquid bearing lot no. 29, was submitted to the Toxicology Research Laboratory for toxicological hazards. This formulation will be used as a solvent for the decontamination of radioactive corrosion scales in nuclear reactors.

This formulation has a low acute oral toxicity and should pose no problem from ingestion incidental to industrial handling. The LD50 in rats is greater than 5 g/kg.

The test formulation is essentially non-irritating to the eye. Minimal eye protection, safety glasses, is recommended for industrial handling whenever the likelihood of eye contact exists.

Skin irritation studies indicate that the test material is essentially non-irritating to either intact or abraded skin and appears not to be absorbed through the skin in acutely toxic amounts. Reasonable care and cleanliness should avert skin irritation problems incidental to industrial handling.

No problem is expected from a single short-term exposure to the vapors of the test material. Six rats were exposed for seven hours to the saturated vapors (20 mg/l of air) generated at room temperature. During the exposure and the subsequent period following the exposure, no adverse effects were observed. Effects of chronic or repeated exposures were not evaluated at this time.

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APPENDIX XIV

EXPECTED RADIATION DOSE RATES IN THE NEW DRESDEN STATION RADIOACTIVE WASTE PROCESSING BUILDING

R & D REPORT

DOW CHEMICAL U.S.A.

CRI NUMBER

LABORATORY REPORT CODE: HEH2.13-22-3(4)

DATE ISSUED: April 20, 1977

DEPARTMENT: HER Industrial Hygiene Laboratory

ACCOUNT NO.:

PROBLEM NUMBER: 580-0560023

TITLE

EXPECTED RADIATION DOSE RATES IN THE DRESDEN STATION RADIOACTIVE WASTE PROCESSING BUILDING

AUTHOR(S) SIGNATURE(S)
S. K. Norwood S. K. Norwood 4-28-77

REVIEWER'S SIGNATURE
R. A. Olson R. A. Olson 4-28-77

DESCRIPTIVE SUMMARY WITH CONCLUSIONS: (Include in this space references to data books, and to earlier related reports, patents and publications.)

Dow Nuclear Service, Functional Products & Systems Department, Dow USA, plans to chemically clean the interior of the primary cooling loop of the Dresden-1 Nuclear Power Station and solidify the radioactive liquid waste that is generated. Expected radiation dose rates in the proposed Dresden Station radioactive waste processing building were calculated to determine if construction specifications for the building were adequate for shielding purposes. It was determined that the shielding walls will be adequate for the expected activity of cobalt-60.

bjd

DISTRIBUTION:

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Central Report Index (5), 566 Building

CRI NUMBER

PROBLEM

The Industrial Hygiene Laboratory was requested to follow construction specifications for a radioactive waste processing building (radwaste building) to be used for solidification of liquid radioactive waste generated by the chemical cleaning of the interior of the primary cooling loop of the Dresden-1 Nuclear Power Station. Specifically, Industrial Hygiene reviewed construction specifications to ensure that there was adequate shielding to reduce radiation levels "as low as reasonably achievable."

RESULTS AND CONCLUSIONS

1. Maximum expected dose rates were calculated for various areas of the proposed radwaste building.

2. Catalytic, Incorporated, provided a summary of expected dose rates calculated independently from the Industrial Hygiene Laboratory.

3. The calculated expected dose rates indicated that the construction specifications were in keeping with the as low as reasonably achievable philosophy.

4. The expected dose rate, from radioactive wastes in the radwaste building, at the nearest perimeter fence was acceptable.

DISCUSSION

The interior of the primary cooling loop of the Dresden-1 Nuclear Power Station is to be chemically cleaned by Dow Nuclear Service. Cleaning chemicals will be circulated through the primary cooling system of the reactor. The chemicals (radioactive waste) will then be pumped from the reactor containment vessel to the radwaste building to be held in the radioactive waste (radwaste) receiving tanks. The radioactive waste will then be reduced in volume by evaporation and pumped to evaporator bottoms storage tanks. The radioactive waste will then be pumped to the solidification area and solidified in 55 gallon drums. The drums of solidified waste will be stored in the solidified drum storage area.

Catalytic, Incorporated, was contracted to design the radwaste building. The design of the radwaste building was to include appropriate shielding to reduce radiation levels as low as reasonably achievable. A summary of Catalytic, Incorporated's, calculations of expected dose rates can be found in Appendix A.

Industrial Hygiene calculated expected dose rates at the nearest perimeter fence and also at various locations within the radwaste building. Standard shielding calculations were used to calculate expected dose rates[1]. Beside actual dose rate calculations, many locations were spot-checked using transmission curves for Co-60 gamma rays passing through concrete and iron[2]. Generally, the calculated dose rates from Catalytic, Incorporated, agreed with calculations made by Industrial Hygiene.

The dose rate estimates from piping were not considered for most locations. However, Catalytic, Incorporated's, specifications call for piping to be adequately shielded so that general radiation levels do not increase above those specified in Appendix A.

The nomenclature and equations used for the dose rate calculations can be found in Appendix B. Appendix B also contains the calculations of the flux to dose rate conversion factor for Co-60 and the macroscopic cross sections used in the dose rate calculations.

The calculations of the expected dose rate at the nearest perimeter fence, due to radioactive wastes in the radwaste building, can be found in Appendix C. The dose rate can be expected to be less than 0.01 millirem per hour.

The radwaste pipe trench contains the pipe running from the reactor to the radwaste building. The calculations of the expected dose rates above the radwaste pipe trench and at the nearest perimeter fence while the solvent is flowing through the pipe can be found in Appendix D. The expected dose rate above the radwaste pipe trench was 0.5 millirem per hour. The expected dose rate at the nearest perimeter fence due to radioactive wastes in the radwaste pipe trench was 6×10^{-12} millirem per hour.

The calculations of the expected dose rates through the shielding walls to the side and above the radwaste receiving tanks can be found in Appendix E. The maximum expected dose rate was 0.04 millirem per hour. However, the dose rate around the roof hatch may be significantly higher.

The calculations of the expected dose rates through the shielding walls to the side and above the evaporator bottoms storage tanks can be found in Appendix F. The maximum expected dose rate was 50 millirem per hour.

Expected dose rates at the side of a 55 gallon drum of solidified waste were calculated for distances of two inches and three feet, see Appendix G. The expected dose rates for one curie of Co-60 in a solidified waste drum were 9.5 rem per hour at two inches and 0.65 rem per hour at three feet.

The calculations of the expected dose rates on the roof and truck bay of the solidified drum storage area can be found in Appendix H. The expected dose rates on the roof and in the truck bay were 2,100 millirem per hour and 0.2 millirem per hour, respectively. This assumes that all the drums were stored in a small area at one end of the solidified drum storage area.

REFERENCES

[1] Theodore Rockwell III, *Reactor Shielding Design Manual*, (Navel Reactors Branch, Division of Reactor Development, United States Atomic Energy Commission)

[2] *Radiological Health Handbook*, US Department of Health, Education, and Welfare, Revised Edition, January, 1970

APPENDICES

Appendix A

Catalytic, Incorporated, Dose Calculations

Drawings showing Dose Rates in the Radwaste Building

Appendix B

Nomenclature

Equations

Calculated Terms

Appendix C

Dose Rate at the Fence Due to Radioactive Wastes in the Radwaste Building

Building Contribution

Valve Pit Contribution

Appendix D

Dose Rates Due to Radioactive Wastes in the Radwaste Pipe Trench

Dose Rates Above the Trench

Dose Rate at the Fence

Appendix E

Dose Rates Due to Radioactive Wastes in the Radwaste Receiving Tanks

Dose Rate at the Side of the Tank

Dose Rate on the Roof

Appendix F

Dose Rates Due to Radioactive Wastes in the Evaporator Bottoms Storage Tanks

Dose Rate at the Side of Tank

Dose Rate Above the Tank

Appendix G

Dose Rates Due to Solidified Radwaste in a 55 Gallon Drum

Dose Rate at Two Inches

Dose Rate at Three Feet

Appendix H

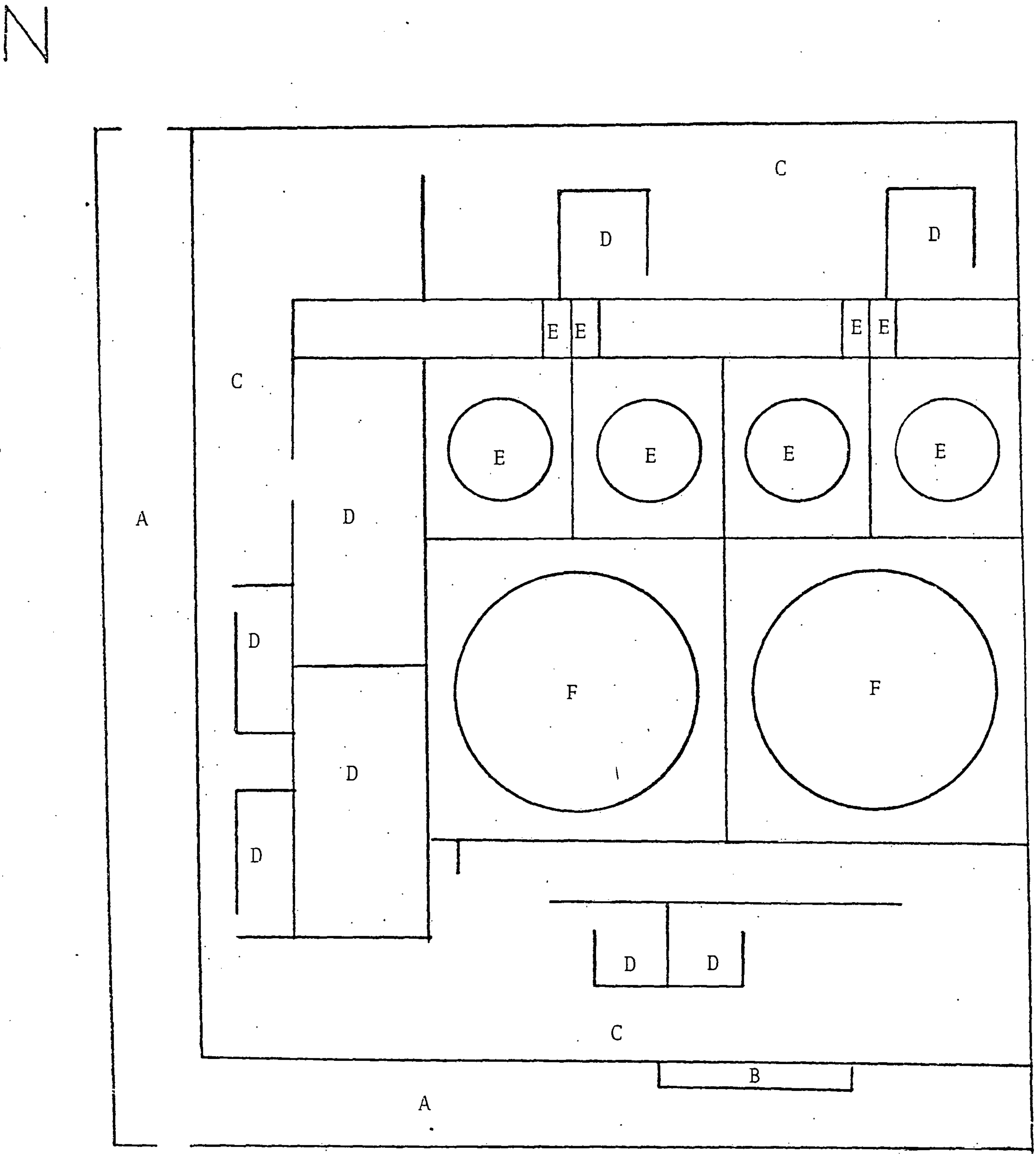
Dose Rates Due to Solidified Radwaste in the Solidified Radwaste Drum Storage Area

Dose Rate on Roof

Dose Rate in Truck Bay

RADWASTE BUILDING PROCESS AREA

1ST LEVEL PLAN



See plan notes on next page

Notes: Radwaste Building Process Area, 1st Level Plan

A. The walkway <2.5 mR/hr.

B. The valve pit >100 mR/hr.

C. The general area just inside the liquid containment perimeter <15 mR/hr.

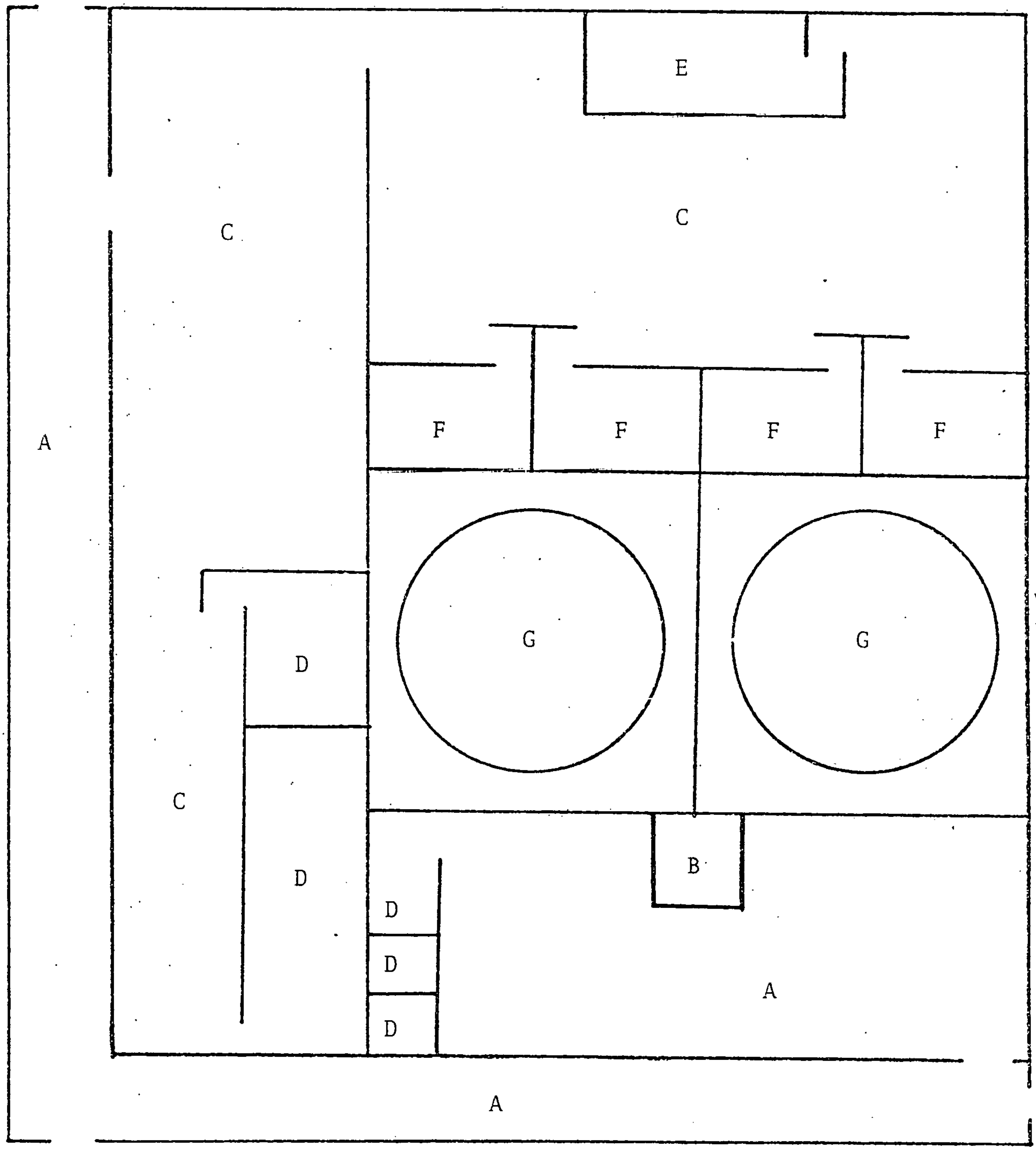
D. The evaporator and support equipment <30 mR/hr in any shielded compartment if there is no radioactive material in that compartment.

E. The evaporator bottoms storage area >100 mR/hr in any shielded compartment if there is no radioactive material in that compartment and the next compartment contains evaporator bottoms.

F. The radwaste receiving tanks <100 mR/hr in any shielded compartment if the receiving tank is empty and the adjacent receiving tank is full.

RADWASTE BUILDING PROCESS AREA

2ND LEVEL PLAN



See plan notes on next page

Notes: Radwaste Building Process Area, 2nd Level Plan

A. The walkway <2.5 mR/hr.

B. The pipe chase >100 mR/hr.

C. The general area <15 mR/hr.

D. The evaporator and support equipment <30 mR/hr in any shielded compartment if there is no radioactive material in that compartment.

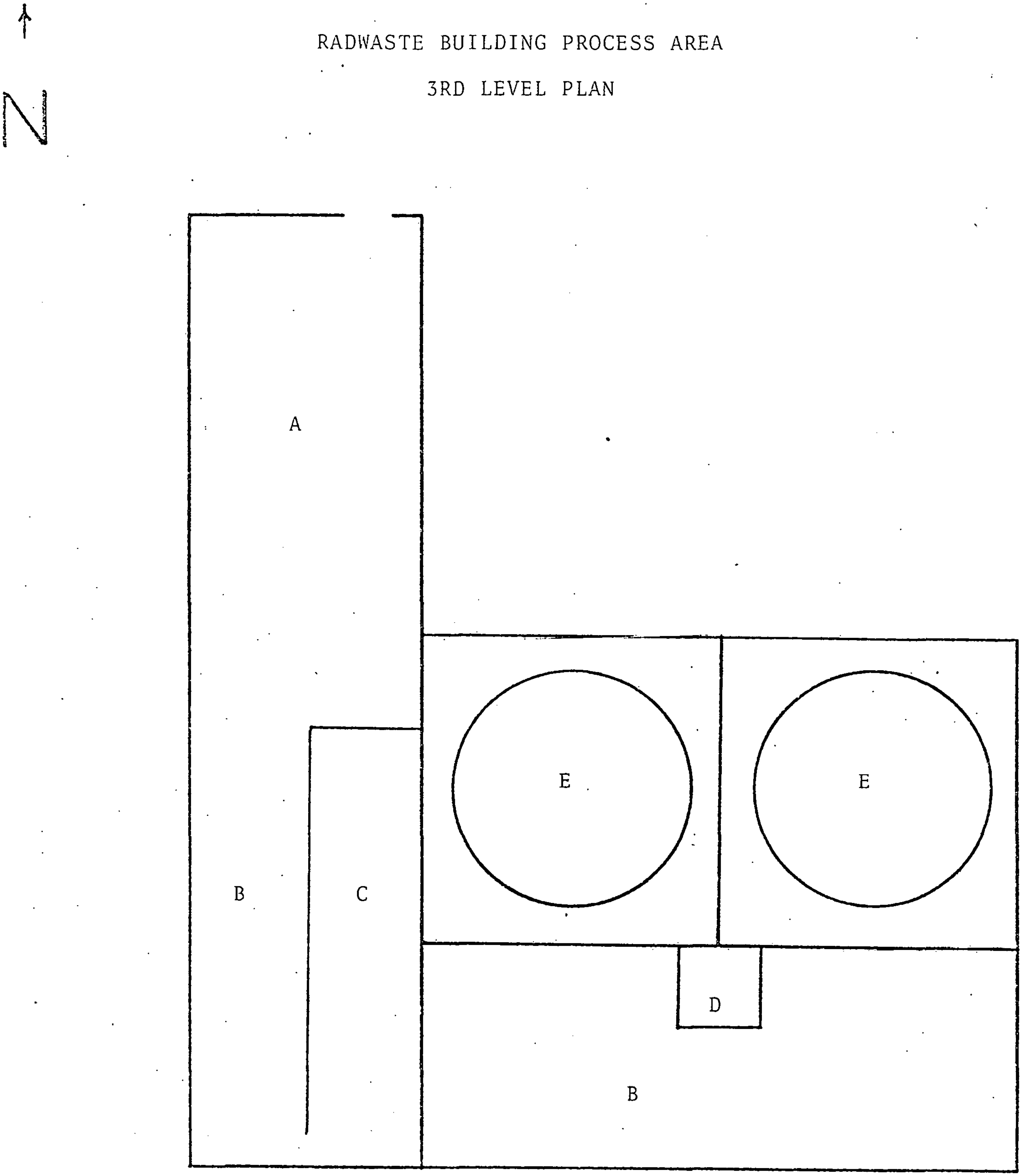
E. The metering tank <30 mR/hr.

F. The general area <30 mR/hr.

G. The radwaste receiving tank <100 mR/hr in any shielded compartment if the receiving tank is empty and the adjacent tank is empty and the adjacent receiving tank is full.

RADWASTE BUILDING PROCESS AREA

3RD LEVEL PLAN



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Notes: Radwaste Building Process Area, 3rd Level Plan

A. The general area <2.5 mR/hr.

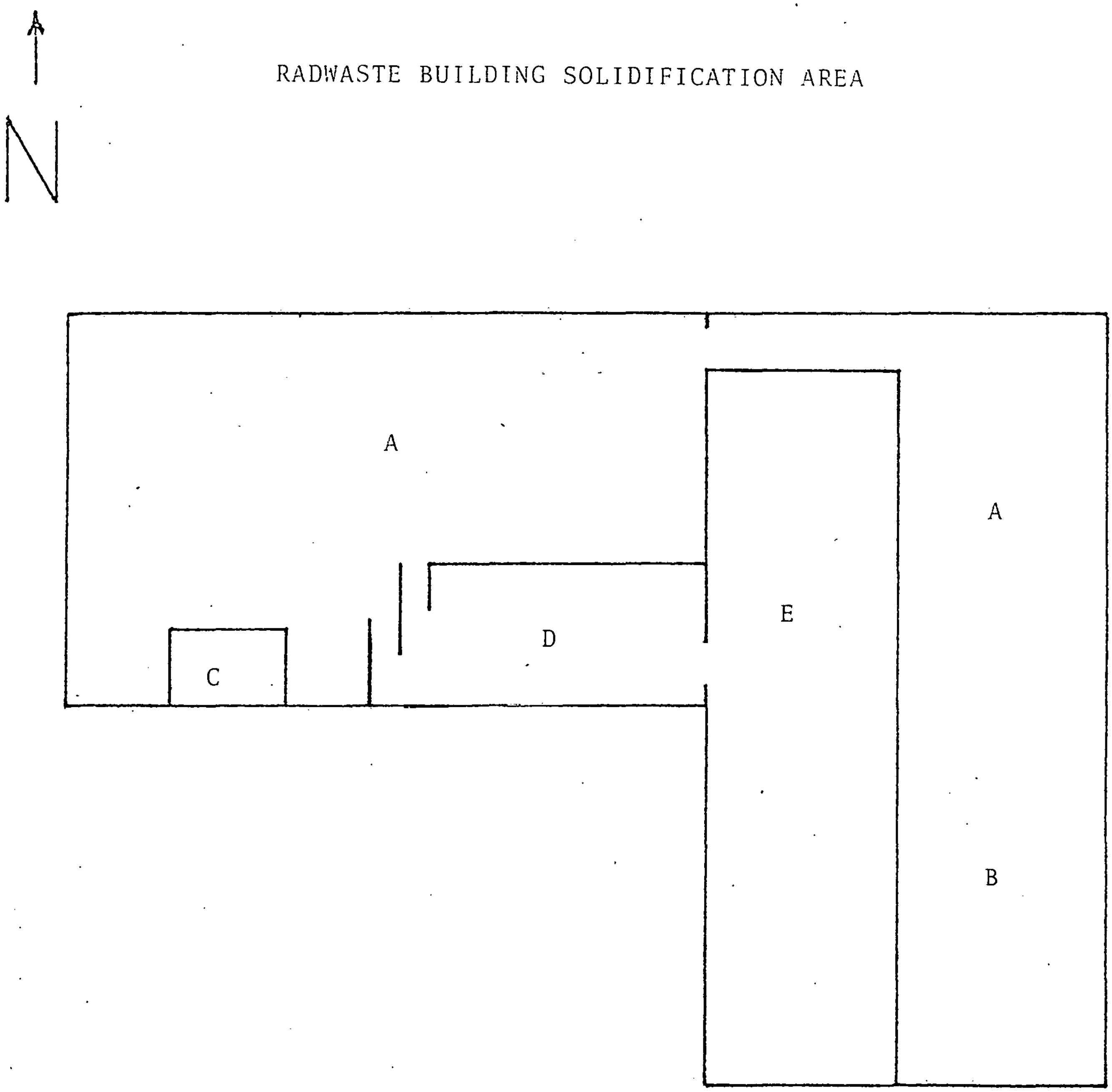
B. The general area <15 mR/hr.

C. The general area <30 mR/hr.

D. The pipe chase >100 mR/hr.

E. The radwaste receiving tank <100 mR/hr in any shielded compartment if the receiving tank is empty and the adjacent receiving tank is full.

RADWASTE BUILDING SOLIDIFICATION AREA



See plan notes on next page

Notes: Radwaste Building Solidification Area

A. The general area <1 mR/hr

B. The area opposite the portion of removable block <2.5 mR/hr.

C. The elevator <15 mR/hr when it is above the solidification area shielding walls.

D. The solidification area <100 mR/hr when it contains no radioactive material and the solidification drum storage area contains radioactive material.

E. The solidified drum storage area <100 mR/hr when it contains no radioactive material and the solidification area contains radioactive material.

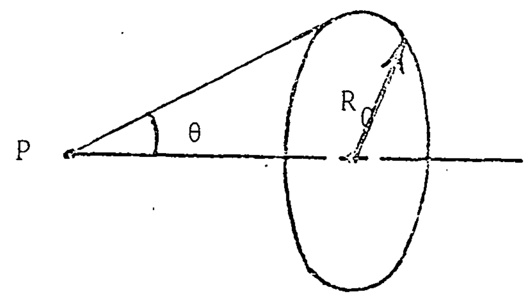
NOMENCLATURE

ϕ	Scalar flux ($cm^{-2}sec^{-1}$)
S_A	Source strength of plane source ($cm^{-2}sec^{-1}$)
S_V	Source strength of volume source ($cm^{-3}sec^{-1}$)
μ_s	Macroscopic cross section of source material (cm^{-1})
$\mu_1, \mu_2, \ldots \mu_n$	Macroscopic cross sections of shields 1, 2, . . . n (cm^{-1})
t_i	Thickness of ith shield (cm)
b_1	$\sum_i^n \mu_i t_i$
b_2	$b_1 + \mu_s Z$
Z	Effective self-attenuation distance (cm)
R_0	Radius of disk, cylinder, or sphere (cm)
B	Symbolic build-up factor
$E_n(b)$	$b^{n-1} \int_b^\infty \frac{e^{-t}}{t^n} dt \quad n \geq 0 \quad E_0(b) = \frac{e^{-b}}{b}$
$F(\theta, b)$	$\int_0^\theta e^{-b} \sec\theta' \, d\theta'$
a	Distance from cylinder
h	Height of cylinder
p	Point of calculated dose rate

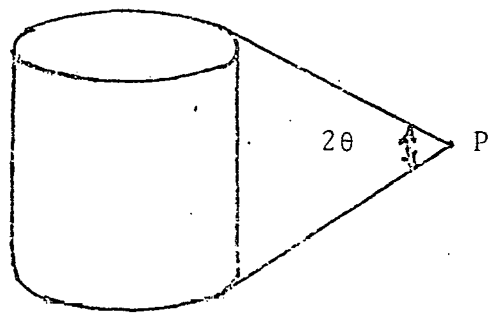
(Reference 1, Page 346)

NOMENCLATURE

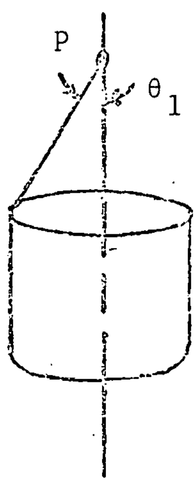
θ, Disk source



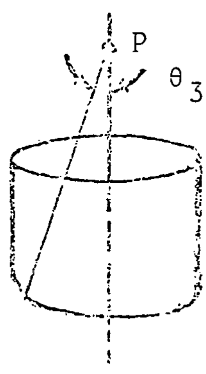
θ, Side of cylinder
P is at h/2



θ_1, Top of cylinder



θ_3, Top of cylinder



EQUATIONS

All equations were taken from reference 1. The exact page number is indicated next to each equation.

Equation 1 (Disk Source):

$$\phi = \frac{B\ S_A}{2}\left[E_1(b_1) - E_1(b,\ \sec\theta)\right] \qquad \text{(Page 348)}$$

Equation 2 (Side of cylinder):

$$\phi = \frac{B\ S_v\ R_0^{\ 2}}{2(a + z)}\ F\ (\theta,\ b_2) \qquad \text{(Page 360)}$$

Equation 3 (Top of cylinder):

$$\phi\ \text{(upper limit)} = \frac{B\ S_v}{2\ \mu_s}\left[E_2(b_1) - \frac{E_2\ (b_1\ \sec\ \theta_1)}{\sec\ \theta_1}\right] \qquad \text{(Page 364)}$$

Equation 4 (Top of cylinder):

$$\phi\ \text{(lower limit)} = \frac{B\ S_v}{2\ \mu_s}\left[E_2\ (b_1) - \frac{E_2\ (b_1\ \sec\ \theta_3)}{\sec\ \theta_3}\right] \qquad \text{(Page 364)}$$

ϕ can be converted to mRem/hr for Co-60 by dividing by 450.

CALCULATED TERMS

Flux to dose rate conversion factor (Co-60)

$\frac{4.5 \times 10^2 \text{ photons cm}^{-2} \text{ sec}^{-1}}{\text{mR/hr}}$ can be calculated from the specific ionization of Co-60 photons in tissue.

Quick check of this value

Specific gamma ray constant = 1.32×10^3 mR/hr

from one curie of Co-60 at one meter (Reference 2, Page 131)

Flux equation for a point source =

$$\phi = B \frac{S_0}{4 \pi a^2} e^{-b_1} \qquad \text{(Reference 1, Page 347)}$$

$B = 1$

$S_0 = (3.7 \times 10^{10} \text{ dis/sec})(2 \text{ photons/dis}) = 7.4 \times 10^{10} \text{ sec}^{-1}$
(1 Ci = Co-60)

$a = 100$ cm

μ air/ρ air (1.25 MeV γ) = 0.06065 (Reference 2, Page 139)

ρ air = 0.001293 gm/cm^3 (Reference 2, Page 66)

μ air = 0.0000784204 cm^{-1}

t air = 100 cm

$$\text{Dose rate} = \left[\frac{(1)(7.4 \times 10^{10})}{4 \pi (100)^2} e^{-(0.0000784204)(100)}\right] \Big/ 4.5 \times 10^2$$

$= 1.298 \times 10^3$ mR/hr

Dose conversion factor checks within two significant figures.

MACROSCOPIC CROSS SECTIONS

All μ's are calculated for Co-60 using a 1.25 MeV gamma ray.

μ/ρ air = 0.06065 (Reference 2, Page 139)

ρ air = 0.001293 g/cm^3 (Reference 2, Page 66)

μ air = 0.000078 cm^{-1}

μ/ρ concrete = 0.06075 (Reference 2, Page 139)

ρ concrete = 2.3 g/cm^3 (Reference 2, Page 66)

μ concrete = 0.1397 cm^{-1}

μ/ρ_s (liquid radwaste - water) = 0.0674 (Reference 2, Page 138)

ρ_s = 1 g/cm^3 (Reference 2, Page 66)

μ_s = 0.0674 cm^{-1}

μ/ρ iron = 0.057125 (Reference 2, Page 138)

ρ iron = 7.86 g/cm^3 (Reference 2, Page 65)

μ iron = 0.449 cm^{-1}

μ/ρ sand = 0.06065 (Reference 2, Page 139)

ρ sand = 1.54 g/cm^3 (Reference 2, Page 66)

μ sand = 0.0934 cm^{-1}

μ_s (solidified waste) = 0.09 cm^{-1} (determined empirically)

μ/ρ insulation = 0.0655 (lucite) (Reference 2, Page 139)

ρ insulation = 1.54 g/cm^3 (plaster) (Reference 2, Page 66)

μ insulation = 0.1 cm^{-1}

APPENDIX C

<u>Dose rate at fence due to radioactive wastes in the radwaste building</u>

The maximum dose rate just inside the liquid containment perimeter is 15 mR/hr. The valve pit is the major source of radiation outside of the liquid containment perimeter. It will be shielded to below 2.5 mR/hr. The outside of the radwaste building will be less than 2.5 mR/hr except at some roof locations. See Appendix A, (1st Level Plan, Sections A, B, and C).

The distance to the fence = 170 feet, see Drawing A-21001, Revision No. 1.

Maximum wall length = 120 feet

Distance from outside wall to inside concrete wall = 11 feet

Valve pit = 22 feet

Minimum wall thickness = 2 feet of concrete

See Drawing A-103, Revision 2 for all of the above.

<u>General building contribution</u>

Assume 120 feet diameter disk source, 15 mR/hr at the surface.

Using equation 1:

Dose rate = 15 mR/hr $B \left[E_1(b_1) - E_1(b_1 \text{ Sec } \theta) \right]$

$b_1 = \mu$ air t air + μ concrete t concrete

μ air = 0.0000784 cm^{-1}

μ concrete = 0.139725 cm^{-1}

$b_1 = (0.0000784204)(5516.88) + (0.139725)(60.96) = 8.95$

$B = \left[B(\text{iron}) + B(\text{al})\right]/2 = (16.5 + 16.075)/2 = 16$

Point source B was used. This is more conservative than a plane source B (Reference 2, Page 145)

$\sec\theta = \frac{hyp}{adj} = \frac{217'}{181'} = 1.2$

$E_1(b_1) = E_1(8.95) = 1.3 \times 10^{-5}$ Reference 1, Page 375

$E_1(b_1 \sec\theta) = E_1(10.74) = 1.85 \times 10^{-6}$ Reference 1, Page 375

$$\text{Dose Rate} = (15\ \text{mR/hr})(16)\left[1.3 \times 10^{-5} - 1.85 \times 10^{-6}\right]$$
$$= 3 \times 10^{-3}\ \text{mR/hr}$$

Contribution from valve pit

Assume 22' diameter disk: surface reading 2.5 mR/hr

t air = 5181.6 cm

μ air = 0.0000784204

$b_1 = 0.41$

$B = 1$

$\sec\theta = \frac{hyp}{adj} = \frac{171.4'}{170'} = 1.008$

$E_1(b_1) = E(0.41) = 0.685$ Reference 1, Page 372

$E_1(b_1 \sec\theta) = E(0.413) = 0.68$ Reference 1, Page 372

Using equation 1:

$$\text{Dose rate} = (2.5\ \text{mR/hr})(1)\left[0.685 - 0.68\right] = 0.01\ \text{mR/hr}$$

This is <u>very</u> conservative as the valve pit is below the floor.

APPENDIX D

Dose Rate Above Radwaste Pipe Trench

Parameters from drawing A-13201, Revision B

The diameter of the pipe is 12 inches.

The thickness of the concrete above the pipe is 12 inches.

The thickness of the dirt above the pipe is 18 inches.

The pipe centerline is 65 inches below the surface.

Assumptions

There are 3,000 curies of Co-60 in 85,000 gallons of cleaning solvent.

The expected dose rate is at 3 feet above the ground.

$S_v = 6.9 \times 10^5 \text{ cm}^{-3} \text{ sec}^{-1}$

$R_0 = 15.24$

$a = 241.3$ cm

$a/R_0 = 15.8$

$\mu_s R_0 = 1.03$

$\mu_s Z = 0.55$ Reference 1, Page 361

$Z = 8.2$

$b_1 = \mu$ concrete t concrete + μ sand t sand

$= 4.26 + 4.27$

$= 8.53$

$\theta = 90°$ (infinite pipe)

$b_2 = 0.55 + 8.53 = 9.08$

$F(\theta, b_2) = 4.5 \times 10^{-5}$ Reference 1, Page 386

B (For 24 inches concrete) = $\left[B(iron) + B(al)\right]/2$, for a point source

This is conservative for a pipe.

B = 15 Reference 2, Page 145

Using equation 2:

$$\text{Dose rate} = \frac{(15)(6.9 \times 10^5)(15.24)^2}{(2)(241.3 + 8.2)(450)} \left[4.5 \times 10^{-5}\right]$$

$$= 0.5 \text{ mR/hr}$$

Attenuation by the pipe was not considered.

<u>Dose Rate at 100 feet from the Radwaste Pipe Trench</u>

$S_v = 6.9 \times 10^5 \text{ cm}^{-3} \text{ sec}^{-1}$

$R_0 = 15.24$

For ease of calculation the radwaste pipe centerline was assumed to be 3 feet below the surface.

a = 3053.5 cm

$a/R_0 = 200.4$

$b_1 = \mu$ concrete t concrete + μ sand t sand

$= 4.26 + 142.6$

$= 147$

$\mu_s R_0 = 1.027$

$\mu_s Z = 0.54$

Z = 8.01

$b_2 = 147.9$

$\theta = 90°$ (infinite pipe)

$F(\theta, b_2) < 2 \times 10^{-15}$ (Reference 1, Page 390)

B>50 (Reference 2, Page 145)

Using equation 2:

$$\text{Dose rate at 100 feet} = \frac{(50)(6.9 \times 10^{5})(15.24)^{2}}{(2)(3061.5)(450)} \quad (2 \times 10^{-15})$$

$$= 6 \times 10^{-12} \text{ mRem/hr}$$

APPENDIX E

Radwaste Receiving Tanks

Volume: 150,000 gallons

Height: 31 feet

R_0 = 13.5' = 411.48 cm

μ concrete = 0.1397 cm^{-1}

μ iron = 0.449 cm^{-1}

μ_s = 0.0674 cm^{-1}

Dose Rate at Side of Tank

a = 5' = 152.4 cm

a = R_0 = 0.37

$\mu_s(a + R_0)$ = 38

m = 2.0 (Reference 1, Page 362)

b_1 = μ concrete t concrete + μ iron t iron

t = concrete = 3' = 91.44 cm

t iron = 3/16" = 0.48 cm

S_v = 1 Ci(Co-60)/gal = 1.955 x 10^7 cm^{-3} sec^{-1}

b_1 = 13

$(\frac{1}{m})\ \mu_s$ z = 2.84 (Reference 1, Page 363)

z = 84

B = $\left[\text{B iron + B al}\right]$ = 30

Conservative as assumed point source but did not consider B in tank (Reference 2, Page 145)

$\theta = 72°$

$b_2 = 18.7$

$F(\theta, b_2) = 2.0 \times 10^{-9}$ (Reference 1, Page 388)

Using Equation 2:

$$\text{Dose Rate} = \frac{(30)(1.955 \times 10^7)(411.48)^2(2.0 \times 10^{-9})}{(2)(236.4)(450)}$$

$$= 0.93 \text{ mRem/hr per Ci(Co-60)/gallon}$$

Maximum Dose Rate at the Side of the Tank

Assume 3,000 Ci(Co-60) in 85,000 gallons

$$\left(\frac{3{,}000 \text{ Ci}}{85{,}000 \text{ gal}}\right) \frac{(0.93 \text{ mRem/hr})}{(\text{Ci/gal})} = 0.03 \text{ mRem/hr}$$

Dose Rate on Roof

t concrete = 2'3" = 68.58

t iron = 0.476"

$b_1 = 9.8$

$B = \left[B(\text{iron}) + B(\text{al})\right]/2 = 17$ (Reference 2, Page 145)

$\sec \theta_1 = 2.552$

$E_2(b_1) = 4.6 \times 10^{-6}$ (Reference 1, Page 375)

$E_2(b_1 \sec \theta_1) = 5 \times 10^{-13}$ (Reference 1, Page 378)

Using Equation 3:

$$\text{Dose Rate} = \frac{(17)(1.955 \times 10^7)}{(2)(0.0674)(450)} \left[4.6 \times 10^{-6} - \frac{5 \times 10^{-13}}{2.552}\right]$$

$$= 25 \text{ mRem/hr per Ci(Co-60)/gal}$$

Maximum Dose Rate on the Roof

Assume 3,000 Ci(Co-60) in 85,000 gallons

$$\left(\frac{3{,}000\ \text{Ci}}{85{,}000\ \text{gal}}\right)\left(\frac{25\ \text{mRem/hr}}{\text{Ci/gal}}\right) = 0.04\ \text{mRem/hr}$$

The dose rates around the hatch above the tanks may be significantly higher.

APPENDIX F

Evaporator Bottoms Storage Tanks

Volume = 4,800 gallons

Assume tank is a cylinder 12'9" in diameter and 5'8" tall.

$\mu_s = 0.0674\ cm^{-1}$ Conservative: will be denser than water

$S_v = 1\ Ci(Co\text{-}60)/gal = 1.955 \times 10^7\ cm^{-3}\ sec^{-1}$

$R_0 = 194.31$ cm

a = 133.35 cm

Dose Rate at Side of Tank - 2' Concrete

μ concrete = $0.1397\ cm^{-1}$

t concrete = 2' = 60.96 cm

$b_1 = 8.52$ (shielding by tank wall not considered)

$a/R_0 = 0.69$

$\mu_s(a + R_0) = 22.1$

m = 1.72 (Reference 1, Page 362)

$(\frac{1}{m})\mu_s z = 2.16$ (Reference 1, Page 363)

z = 55

$b_2 = 12.24$

Sin $\theta = 0.544$ $\theta = 33°$

$F(\theta, b_2) = 1.4 \times 10^{-6}$ (Reference 1, Page 387)

B = (B iron + B al)/2 for point source, more conservative

= 14 (Reference 2, Page 145)

Using Equation 2:

$$\text{Dose Rate} = \frac{(14)(1.955 \times 10^{7})(194.31)^{2}(1.4 \times 10^{-6})}{(2)(188.35)(450)}$$

$$= 8.535 \times 10^{1} \text{ mR/hr per Ci(Co-60)/gal, through 2' concrete}$$

<u>Dose Rate at Side of Tank - Maximum Dose Through Minimum Shielding</u>

Assume: 3,000 Ci/4,800 gal = 0.625 Ci/gal

$$\left(\frac{85.35 \text{ mR/hr}}{\text{Ci/gal}}\right) (0.625 \text{ Ci/gal}) = 53 \text{ mR/hr}$$

In normal access areas there is 3' of concrete instead of the 2' of concrete in this example. Shielding by the tank wall was also neglected.

<u>Dose Rate Above Evaporator Bottoms Tank</u>

h = 172.7 cm

μ_s = 0.0674 (water, conservative as bottoms will have larger ρ)

S_v - 1 Ci(Co-60)/gal = 1.955×10^{7} cm^{-3} sec^{-1}

R_0 = 194.31 cm

a = 175.26 cm

μ concrete = 0.139725 cm^{-1}

t concrete = 2'3" = 68.6 cm

b_1 = 9.58 (shielding by tank wall not included)

$B = \left[B(\text{iron}) + B(\text{al})\right]/2 = 18$ (Reference 2, Page 145)

$\sec\theta_1$ = 2.0779

$\sec\theta_3$ = 1.218

$E_2(b_1) = 6 \times 10^{-6}$ (Reference 1, Page 375)

$E_2(b_1 \sec \theta_1) = 9.4 \times 10^{-11}$ (Reference 1, Page 377)

$E_2(b_1 \sec \theta_3) = 6.2 \times 10^{-7}$ (Reference 1, Page 375)

Using Equation 3:

$$\text{Dose Rate (upper limit)} = \frac{(18)(1.955 \times 10^7)}{(2)(0.0674)(450)} \left[6 \times 10^{-6} - \frac{9.4 \times 10^{-11}}{2.0779}\right]$$

= 35 mRem/hr per Ci/gal

Using Equation 4:

$$\text{Dose Rate (lower limit)} = \frac{(18)(1.955 \times 10^7)}{(2)(0.0674)(450)} \left[6 \times 10^{-6} - \frac{6.2 \times 10^{-7}}{1.218}\right]$$

= 32 mRem/hr per Ci/gal

<u>Maximum Dose Rate Through Floor Above Evaporator Bottoms Tank</u>

Assume 3,000 Ci/4,800 gallons

$$\left(\frac{3{,}000 \text{ Ci}}{4{,}800 \text{ Gal}}\right)\left(\frac{35 \text{ mRem/hr}}{\text{Ci/gal}}\right) = 22 \text{ mRem/hr}$$

Tank wall attenuation was not included.

APPENDIX G

<u>Dose Rate at 2" for one Curie Co-60</u>

$S_v = 4.26822 \times 10^5$

R_0 = 28 cm (as measured from solidified block)

a = 5.1 cm

h = 77.5 cm (as measured from solidified block)

16 gauge steel drum = 0.06 inch iron = 0.1524 cm iron

μ iron = 0.449 cm^{-1}

$b_1 = \mu$ iron t iron = 0.068

$\mu_s = 0.09$

$a/R_0 = 0.18$

$\mu_s(a + R_0) = 2.98$

m = 0.705 (Reference 1, Page 362)

$\left(\frac{1}{m}\right)(\mu_s z) = 2$ (Reference 1, Page 363)

z = 15.7

Sin θ = 0.9916 $\theta = 83°$

$b_2 = 1.48$

B = B(28 cm water) = 3.08 (Reference 2, Page 147)

$F(\theta, b_2) = 1.75 \times 10^{-1}$ (Reference 1, Page 385)

Using Equation 2:

$$\text{Dose Rate} = \frac{(3.08)(4.26822 \times 10^5)(28)^2(0.175)}{(2)(20.8)(450)}$$

$$= 9.635 \times 10^3 \text{ mR/hr per Ci(Co-60)/drum}$$

Dose Rate at one Meter for one Curie Co-60

$S_v = 4.26822 \times 10^5$

$R_0 = 28$ cm

$a = 100$ cm

$h = 77.5$

$b_1 = 0.068$

$\mu_s = 0.09$

$a/R_0 = 3.57$

$\mu_s(a + R_0) = 11.5$

$m = 1.58$ (Reference 1, Page 362)

$\frac{1}{m}(\mu_s z) = 0.99$ (Reference 1, Page 363)

$z = 17.38$

$B = 3.08$

$\sin\theta = 0.36$ $\theta = 21°$

$b_2 = 1.63$

$F(\theta, b_2) = 6.6 \times 10^{-2}$ (Reference 1, Page 385)

Using Equation 2:

$$\text{Dose Rate} = \frac{(3.08)(4.26822 \times 10^5)(28)^2(6.6 \times 10^{-2})}{(2)(117.38)(450)}$$

$$= 6.44 \times 10^2 \text{ mR/hr per Ci(Co-60)/drum}$$

APPENDIX H

Solidified Drum Storage Area Roof

Assume 3,000 Ci (Co-60) in a 14' diameter cylinder 2.5' tall

Volume of cylinder = $\pi r^2 h = \pi (213.36)^2 (76.2)$

$= 1.09 \times 10^7 \text{ cm}^3$

$S_v = (3{,}000 \text{ Ci}/1.09 \times 10^7 \text{ cm}^3)(3.7 \times 10^{10} \text{ dis/sec/Ci})(2\ \gamma/\text{dis})$

$= 2.04 \times 10^7 \text{ cm}^{-3} \text{ sec}^{-1}$

μ concrete = 0.1397 cm^{-1}

μ iron = 0.449 cm^{-1}

μ insulation = 0.1 cm^{-1}

t concrete = 3" = 7.62

t iron = 1/16" = 0.159

t insulation = 3" = 7.62

$b_1 = 1.065 + 0.04 + 0.762 = 1.87$

B = 3 (Reference 1, Page 145)

$\mu_s = 0.09$

h = 76.2

$\sec \theta_1 = 1058/1036 = 1.021$

$\sec \theta_3 = 1148/1128 = 1.018$

$E_2(b_1) = 4.4 \times 10^{-2}$ (Reference 1, Page 373)

$E_2(b_1 \sec \theta_1) = 4.2 \times 10^{-2}$ (Reference 1, Page 373)

$E_2(b_1 \sec \theta_3) = 4.2 \times 10^{-2}$ (Reference 1, Page 373)

Using Equation 3:

$$\text{Upper Dose Rate} = \frac{(3)(2.04 \times 10^{7})}{(2)(0.09)(450)} \left[4.4 \times 10^{-2} - \frac{4.2 \times 10^{-2}}{1.021} \right]$$

$$= 2.2 \times 10^{3} \text{ mR/hr}$$

Using Equation 4:

$$\text{Lower Dose Rate} = \frac{(3)(2.04 \times 10^{7})}{(2)(0.09)(450)} \left[4.4 \times 10^{-2} - \frac{4.2 \times 10^{-2}}{1.018} \right]$$

$$= 2.1 \times 10^{3} \text{ mR/hr}$$

The dose rate will drop significantly as one moves out from the cylinder centerline or as the material is spread out in the truck bay.

<u>Solidified Drum Storage Area Truck Bay</u>

t concrete = 3'3" = 99.1 cm

μ concrete = 0.139725 cm^{-1}

b_1 = 13.85

$B = \left[B(\text{iron}) + B(\text{al}) \right] / 2 = 27$

a = 3'3" = 99.1 cm

R_0 = 7' = 213.4

a/R_0 = 0.46

$\mu_s(a + R_0)$ = 28

m = 2.0 (Reference 1, Page 362)

$(\frac{1}{m})(\mu_s z)$ = 2.74 (Reference 1, Page 363)

z = 60.9

b_2 = 19.3

$\sin\theta = 38.1/106.2 = 0.36$

$\theta = 21°$

$F(\theta, b_2) = 1.15 \times 10^{-9}$ (Reference 1, Page 388)

Using Equation 2:

$$\text{Dose Rate} = \frac{(27)(2.04 \times 10^{7})(213.4)^{2}(1.15 \times 10^{-9})}{(2)(160)(450)}$$

$$= 0.2 \text{ mR/hr (sky-shine neglected)}$$