U.S. ARMY RESEARCH LABORATORY

ADELPHI LABORATORY CENTER
RISK MANAGEMENT BRANCH

FAX COVER SHEET

DATE: 16 Apr 97

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Number of Pages, Including Cover: 13

REMARKS:

As discussed. Final with enclosures and transmitting memorandum will be coming shortly.

M

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DRAFT

DORF EXPOSURE ROOM SURVEY

1. Background

1.1 During the period of 1961 through 1977, an Army Research Laboratory predecessor organization, Harry Diamond Laboratories, owned and operated a TRIGA nuclear research reactor in a dedicated facility known as the Diamond Ordnance Radiation Facility (DORF). During the period of 1977 through 1979, the DORF was decommissioned, and released for unrestricted use. The DORF building has since been used as a radioactive waste storage area by the Walter Reed Army Medical Center (WRAMC). The building is now owned and operated by the Walter Reed Army Medical Center.

1.2 During 1996, the Army Reactor Office (ARO) at the U.S. Army Nuclear and Chemical Agency began a review of documents regarding all Army reactors, past and present. Of concern was whether DORF was decommissioned in a way that complies with today's decommissioning standards. ARO's review of DORF final survey documentation indicated that although the gamma radiation levels complied with the decommissioning criteria of 1980, they might not comply with today's standard. The request for survey at Encl 1 was therefore sent to ARL.

1.3 ARL therefore planned and conducted this survey. To minimize the cost to the Army, all survey planning, area gridding, and measurements were conducted by Mr. Michael Borisky, the ALR Health Physicist. The ARL Radiation Control Committee was also used as a planning resource. A review of Rockwell's DORF Decommissioning Program Final Report (Encl 2) and associated surveys was first conducted to determine what types of surveys were conducted in the past, and what type was needed now. The results of the review, attached as Encl 3, indicated that past survey efforts are sufficient to demonstrate compliance with all of today's requirements except the criteria requiring that the gamma dose rate at one meter from surfaces not exceed background by more than 5 urem/hr. Since it was possible that the gamma survey would demonstrate compliance with the gamma levels, the survey was approached and conducted so that it might eventually serve as a final survey.

1.4 Although this survey encompasses only a small portion of the decommissioning and survey effort, an attempt will be made to follow the final report format recommended by the Nuclear Regulatory Commission in NUREG 5849. In this way, it is hoped that this report will be more complete and informative. If the reader seeks a greater level of detail on past information, it is requested the reader study the referenced and enclosed past documents and reports.

2. Site Description

2.1 Review of the decommissioning report and associated survey indicates that the only area of concern is the exposure room. The exposure room is a concrete structure surrounded by soil, with a tunnel leading into the exposure room from the south side. The tunnel had previously been equipped with a rolling concrete plug door that was inserted into the tunnel during reactor operation.

2.2 Outside the tunnel is a general area where WRAMC short half-life radioactive waste is now held for decay. Also located outside the tunnel is a shielding pig that houses a Cs-137 calibration source, identification number 137-Cs-002. Inside the exposure room, located along the west side of the exposure room, is a cold room used to hold animal carcasses contaminated with short-lived radionuclides. Directly above the exposure room, separated by approximately 10 feet of earth/concrete shielding, is a low-level radioactive waste staging and packaging area. And at one end of the staging and packaging area is a fume hood where old liquid radionuclide sources are stored awaiting disposal. Measurements of all these sources indicated that due to geometry, shielding, activity, and decay, none of these sources were capable of significantly interfering with the measurement of gamma levels inside the exposure room.

2.3 The interior surfaces of the exposure room are mostly concrete, in some places covered with a thin layer of phenolic impregnated paper. On certain portions of the walls and ceiling, the concrete had been surface excavated during decommissioning removing 0 to approximately 12 inches of severely neutron activated concrete. The portion of the east wall, ceiling, and floor that intersected with the reactor pool had been completely removed and replaced with new concrete. The portion of the reactor pool volume behind the east wall and above the ceiling had been backfilled with concrete rubble from the demolition of the reactor pool parapet.

2.4 As mentioned earlier, a cold room was present along the west wall of the exposure room. The cold room rendered the concrete surfaces of the walls, floor, and ceiling along the west side of the exposure room inaccessible. Because the cold room was constructed of very low density materials (aluminum sheet metal filled with styrofoam insulation), and because the cold room walls, ceiling, and floor were within 1 meter of the test cell concrete surfaces, measurement through the cold room walls ceilings and floor was possible and considered technical feasible.

3. Operating History

3.1 As previously mentioned, the DORF building housed a TRIGA Mark F reactor which operated from 1961 until 1977. During that period, the reactor was used to generate neutron environments in the exposure room where the effects of nuclear radiation on electrical, chemical, and biological systems were studied. In the process, the neutron environment activated the concrete and possibly the reinforcement bar in the exposure room and reactor pool concrete.

3.2 Decommissioning began in 1979 and was completed in 1980. As part of the decommissioning effort, highly neutron activated concrete in the exposure room and reactor pool was either removed, or surface excavated. Core samples of concrete were taken for isotopic analysis. There is no indication that samples of reinforcement bar were taken for isotopic analysis. Also as part of the decommissioning effort, a final survey of building surfaces, water, soil, and vegetation was conducted.

3.3 In 1980, the U.S. Army Environmental Hygiene Agency conducted a close-out survey of the facility, and found that the facility conformed to the requirement of Nuclear Regulatory Commission Regulatory Guide 1.86 for unrestricted release of the facility. The license for the reactor was therefore terminated.

3.4 Shortly following the license termination, the facility was transferred to the control and ownership of WRAMC. The WRAMC Health Physics Group has been using the facility as a low level radioactive waste holding, staging, and packaging area ever since. These rad waste operations have been covered by WRAMC's NRC licenses 08-01738-02 and DARA 08-01-97. In the exposure room, which is the subject of concern for this survey, rad waste operations have been limited to holding animal carcasses and radiotherapy wastes in cold storage for radioactive decay or shipment. Radioisotopes involved included primarily H-3, C-14, I-125, I-131, Cr-51, Ce-141, and Sc 46. Radioactive wastes held in the cold room or test cell were generally contained or enclosed in plastic bags, or containerized in 55 gallon drums. Weekly surveys by WRAMC Health Physics verifies the absence of radiological contamination in the exposure room from these rad waste operations.

4. Decommissioning Activities.

4.1 Previous decommissioning activities are described in Rockwell's DORF Decommissioning Program Final Report (Encl 2). This effort is limited to a gamma radiation survey of the exposure room. The objective of this survey is to determine whether the gamma radiation levels in the exposure room meet today's standards.

4.2 Surveys conducted by Rockwell after decommissioning included removable and fixed contamination measurement on concrete both inside and outside the exposure room, as well as neutron activation sampling of exposure room concrete. Air monitoring was conducted in the exposure room and high bay area during decommissioning activities. Radiation levels were found to be within limits for unrestricted release. As stated in Encl 3, the fixed and removable surface contamination clean-up limits that were applied meet today's limits. The water, soil, and vegetation were found to be contamination free. Interior surfaces were found to be free of removable contamination, with any fixed contamination present within limits.

4.3 Rockwell's pre and post clean-up contamination surveys did indicate significant neutron activation of the exposure room concrete and reactor pool concrete. The post clean-up levels at 1 cm from the surfaces indicated about 100 urad/hr over relatively large areas. Core samples indicated the presence of Co-60, Eu-152, and Eu-154 in the concrete, at concentrations of about 10-100 pCi/gm. Again, activation in the reinforcement bar was apparently not assayed.

4.4 The AEHA survey at Encl 4 also included surface wipe tests, concrete, water, soil, and vegetation analysis. The conclusion the of survey was that the facility conformed to the requirement of NRC Regulatory Guide 1.86 for unrestricted release. As pointed out by ARO in Encl 1, the AEHA survey also found radiation levels as high as 400 microroentgens per hour in the exposure room. ARO further pointed out that even allowing for normal decay of the radioactive contamination, this level indicates that the DORF exposure room might remain above the 5 urem/hr (above background at one meter) current standard for decommissioned facilities.

5. Survey Procedure

5.1 To address the concerns of the ARO, the exposure room was surveyed for compliance with the new 5 urem/hr decommissioning criteria. The survey was planned and conducted to meet the requirements of NUREG 5849 for a final survey, so that if the criteria was met, no further surveying would be necessary. As surveying of the exposure room began, it became evident that the 5 urem/hr criteria would not be met, and that the survey would therefore not serve as a final survey, but rather a scoping or characterization survey. It was therefore unnecessary to survey every originally planned location.

5.2 Microrem/hr measurements were made at approximately 36 inches from the floor, ceiling, and wall surfaces in the exposure room. Given that the activation of the concrete created a planar source of radioactivity (as evidenced by the concrete surface excavations), the difference between 36 inches and one meter was not considered a significant variable to gamma levels expected. The use of 36 inches instead of one meter aided in systematically gridding the exposure room The identity of radionuclides presenting the gamma radiation levels were not considered important for the purposes of this survey.

5.3 Because it was know that radionuclides were present in the exposure room concrete as a result of reactor operation in the past, the exposure room was considered an "affected area", and surveyed accordingly.

5.4 Grids were established on the walls, ceiling, and floor. Grid sizes were less than or equal to 36 inches in size. Where possible, grid corners were marked permanently on the surfaces with paint, for future reference if necessary. Grid locations were transferred to the inside of the cold room as described earlier. Reference Encl 5 for a scale drawing of measurement locations, which were actually the grid corners. Measurement were taken at approximately 36 inches from surfaces so that in corners, a single measurement would serve as a measurement point for more than one surface. For this reason, a single measurement may be associated with 2 or 3 measurement location designators.

5.5 Two areas were chosen as reference areas to establish background gamma radiation levels.

5.5.1 The rolling door alcove, located outside and opposite the exposure room entrance tunnel, was chosen for two reasons. First, like the exposure room, it is subterranean, with 2 and 3 plane corners. Secondly, it was most probably constructed of with the same building materials as the exposure room. The disadvantage of door alcove is that being located within the facility, and being associated with present and past WRAMC rad waste operations, it might be subject to some concern as to whether it would serve as a suitable reference area. See Encl 6 for a scale drawing of the door alcove, and originally planned measurement locations. Like the test cell, the area was gridded. To account for any effect that 2 and 3 plane corners might have on background levels, measurement locations were chosen so that the percentage of measurement in 2 and 3 plane corners would be approximately the same as in the exposure room.

5.5.2 An additional reference area was chosen outside the facility, but as close as possible to the facility. At the end of the DORF entrance road, a new parking garage has recently been constructed to serve as a bus stop. Like the exposure room and door alcove, the lower level is subterranean, with concrete walls and floors backed by soil. This areas was also gridded, and again, measurement locations were chosen so that the percentage of measurement in 2 and 3 plane corners would be approximately the same as in the exposure room. See Encl 7 for a scale drawing of the garage reference area, and measurement locations.

5.5 A BAIRD microrem/hr portable survey meter was chosen to perform the measurements. The meter procured was also equipped with a low energy window, which allows measurement of photons down to 17 keV in energy. The low energy window was not considered necessary for the survey, but was selected for other uses at ARL in the future. The low energy window was not expected to detract in any way from the validity of the survey.

5.5.1 Final survey gamma measurements are typically conducted using a uR/hr meter cross calibrated to a pressurized ion chamber (PIC). The cross calibration is necessary because of the severe difference in relative response between NaI and the pressurized gas with variations in gamma energy. It is then assumed that each microroentgens per hour measured with the pressurized gas in a particular location would result in one microrem per hour of effective dose equivalent in that location. In an area where much scattering is occurring, which is certainly the case with the concrete exposure room, the energy spectrum of photons could vary greatly from location to location, and might not match that present at the location of cross calibration. For these reasons, ARL decided to employ a relatively new technology offered by the BAIRD Microrem/hr meter.

5.5.2 The BAIRD uses a tissue equivalent scintillation detector. The detector is as sensitive as the NaI uR/hr meter, but offers the advantage that it's tissue equivalent detector allows direct measurement of effective dose equivalent rates, regardless of the photon energy spectrum present at the location of measurement. It is also lightweight and portable, and eliminates the need for cross calibration with the PIC, which is bulky .

5.5.3 Upon receipt, it was found that on the lowest scale, the fluctuation of the BAIRD meter needle at background level was such that discerning a reliable reading would not be possible. For example, the background reading would typically fluctuate between 0 and 6 urem/hr. To address this problem, those at Oakridge Associated Universities reportedly cover the meter face with a hand, quickly lifting the hand and recording the meter needle position at that moment. ARL decided to employ another approach.

5.5.4 Coordinating with BAIRD design engineers, ARL proposed increasing the RC time constant of the lowest scale so that at background levels, the meter reading would be more stable and discernible, approximately +/- 20%. A 0.068 microfarad capacitor was recommended by BAIRD, and wired into the lowest scale RC circuit by ARL engineers. ARL then tested the meter stability at background, and found it was discernible, with fluctuations of about +/- 20%. The time required to hold the instrument in a particular location was increased, but for ARL purposes, it was considered a worthwhile tradeoff. The instrument was then returned to BAIRD for re-certification and calibration, to ensure the modification did not introduce any adverse impacts. BAIRD also installed a switch so that on the lowest scale, the 0.068 microfarad capacitor could be switched in for static measurements, and out for scanning where fast response is needed. BAIRD empirically determined that in the slow response position on the lowest scale, 90 seconds was now required following meter reset to reach 95 maximum reading.

5.5.5 To ensure the proper operation of the instrument on each day that it was used, both before and after measurements were taken, it was battery checked, high voltage checked, background checked, and response checked as described in Encl 8. The response check included placing the meter in a 5 urem/hr above background field using a Cs-137 source, and ensuring a net value of 5 urem/hr was indicated.

5.5.6 All measurements were taken between 18 and 28 Mar 1997, well within 6 months of the 16 Jan 97 calibration date.

5.6 The standard written procedure at Encl 8 was followed. Quality assurance/quality control procedures employed included recording data on a standard form with ink, daily pre and post instrument checks as described, and the use of scale drawings of measurement locations.

5.7 The survey planning, layout, measurement, and reporting was conducted by Michael Borisky. Michael Borisky has been an operational Health Physicist with the Army Research Laboratory since 1981, during which period he provided technical oversight over the massive decommissioning at ARL's Materials Technology Laboratory from 1991 to 1996. Michael Borisky holds a M.H.S. in Radiation Health from the Johns Hopkins

School of Hygiene and Public Health. Michael Borisky has attended various NRC workshops on decommissioning, and in Nov 96, attended a 2 day NRC sponsored workshop on Radiological Surveys in Support of Decommissioning, conducted by ORISE. It was during this workshop that Michael Borisky learned of the BAIRD tissue equivalent Microrem/hr meter.

6. Survey Findings

6.1 As mentioned earlier, there was radioactive waste and a calibration source present during the survey that were not considered a significant contributor to the gamma radiation levels measured. In the event that the net gamma levels measured inside the exposure room were critically close to the 5 urem/hr criteria, an attempt would have been necessary to remove all sources during the survey. This was not necessary because of the relatively high gamma levels measured throughout the exposure room. The following is the rationale used for assessing the impact of the radioactive waste and calibration source on the survey results:

6.1.1 The reference alcove outside the exposure room tunnel is an area where short half-life radioactive waste was being held for decay. On the day that reference measurements were made in the alcove, gamma levels on contact with the waste were not significantly above background when checked with the BAIRD. Nonetheless, the waste was moved approximately 10-15 feet out of the alcove while the reference measurement were being made. As can be seen in Encl 10, the average level measured in the reference alcove was still less than that measured in the reference parking garage, and certainly well below the average level measured in the exposure room. It is therefore inconceivable that the low level rad waste outside the exposure room could have contributed to the gamma radiation levels measured inside the exposure room.

6.1.2 Also located outside the tunnel was a shielding pig containing a Cs-137 calibration source, identification number 137-Cs-002. The gross gamma level measured with the BAIRD approximately 3 feet from the pig was approximately 4.5 urem/hr. As a point source, geometry alone would reduce any gamma levels inside the exposure room many meters from the calibration source to insignificant levels.

6.1.3 The BAIRD was used to measure levels on contact with the few contaminated carcasses and therapy waste in the cold room. No discernible increase was detected in close proximity to the carcasses or waste. Furthermore, the gamma measurements were made with the carcasses and waste in the cold room, and only location F4, where 41 urem/hr was detected, appeared significantly higher that the levels generally measured in the cold room. The remainder of the measurements inside the cold room were generally lower than in the remainder of the exposure room. This was perhaps because the test cell is located along the wall farthest from where the reactor was located, resulting in a decreased level of neutron activation in that portion of the exposure room.

6.1.4 Directly above the exposure room, separated by approximately 10 feet of earth/concrete shielding, is a low-level radioactive waste staging and packaging area. At one end of the staging and packaging area is a fume hood where old liquid radioisotope sources are stored awaiting disposal. The BAIRD was used to make measurement in this area. The gamma level measured in the area was generally about 10 urem/hr, with about 20 urem/hr on the fume hood, and 70 urem/hr on a consolidation drum. Given that the exposure room is shielded from these levels by 10 feet of concrete and/or earth, and given that the fume hood and drum would appear as point sources to the exposure room, shielding and geometry would reduce the contribution of these sources to the levels measured in the exposure room to insignificant levels.

6.2 The raw field data for the surveys is attached as Encl 9. A re-tabulation of the data appears as Encl 10. Gamma radiation levels measured can be summarized as follows:

Area	Range (urem/hr)	mean	standard deviation
reference garage	2.6 - 4.2	3.2	0.4
reference alcove	1.9 - 3.2	2.3	0.3
exposure room	18 - 41	28.9	5

6.3 NUREG 5849, paragraph 8.5.3 recommends that exposure rates be compared directly with the guideline value. The present exposure rate guideline typically applied is 5 urem/hr above background at one meter. Adding 5 urem/hr to the mean and upper range value for the reference garage yields values of 8.2 urem/hr and 9.2 urem/hr respectively. Adding 5 urem/hr to the mean and upper range value for the reference alcove yields values of 7.3 and 8.2 respectively. These values are easily exceeded by both the mean level and lowest level measured in the exposure room. It therefore appears that the exposure room concrete still contains residual activation, and that the activation is relatively uniformly distributed throughout the exposure room concrete.

6.4 ARL will be making arrangements for a gamma spectrometry survey of the gamma radiation in the exposure room. In this way, it will be possible to determine what radionuclides are present in the concrete. After identifying the radionuclides, it will be possible to determine how long it will take for radioactive decay to reduce the exposure room levels to the 5 urem/hr criteria. This will determine whether delay and decay can be used as a decommissioning strategy, as opposed to the large expense of removing the concrete and disposing of it as radioactive waste.

7.0 Summary

7.1 ARO requested a survey of gamma radiation levels in the exposure room. ARL's review of decommissioning records indicated that past survey efforts are sufficient to demonstrate compliance with all of today's requirements except the criteria requiring that the gamma dose rate at one meter from surfaces not exceed background by more than 5 urem/hr. A survey was therefore conducted to determine whether the gamma radiation levels in the exposure room meets this criteria.

7.2 Pre and post clean-up surveys conducted at the time of decommissioning indicated significant neutron activation of the exposure room concrete and reactor pool concrete. Concrete core samples indicated the presence of Co-60, Eu-152, and Eu-154 in the concrete. Activation in the reinforcement bar was apparently not assayed.

7.3 Microrem/hr measurements were made at approximately 36 inches from the floor, ceiling, and wall surfaces in the exposure room. Two areas were chosen as reference areas to establish background gamma radiation levels. A BAIRD microrem/hr portable survey meter with a tissue equivalent scintillator material was chosen for the measurements. The gamma radiation levels measured at 36 inches from the exposure room concrete easily exceeded background plus 5 urem/hr. The exposure room concrete still contains residual radioactivity. The activation is relatively uniformly distributed throughout the exposure room concrete.

7.4 ARL will make arrangements to conduct a gamma spectrometry survey in the exposure room to identify the radionuclide(s) present in the concrete. It will then be possible to determine from the associated half life whether delay and decay will offer a feasible decommissioning strategy.

ENCLOSURE 10

SURVEY DATA

1. Garage Reference Area

Location	Gross Reading (urem/hr)	Location	Gross Reading (urem/hr)
R1	3.0	R2	3.2
R3	3.7	R4	3.2
R5	2.9	R6	3.2
R7	2.9	R8	3.0
R9	3.3	R10	3.5
R11	3.1	R12	4.2
R13	4.1	R14	3.0
R15	2.6	R16	3.1
R17	3.7	R18	3.4
R19	3.9	R20	2.9
R21	3.3	R22	3.0
R23	2.9	R24	3.4
R25	3.0	R26	3.0
R27	3.2	R28	2.8
R29	2.9	R30	3.1

$\overline{X}$ = 3.22 urem/hr SD = 0.38

2. Alcove Reference Area

Location*	Gross Reading (urem/hr)	Location*	Gross Reading (urem/hr)
E1, F13	2.4	E2, F17	2.2
E3, C13	2.6	E4, C17	2.7
S2, F18	2.3	S3, F19	2.0
S6, C18	3.2	S7, C19	2.4
W1, F20	2.8	W2, F16	2.1
W3, C20	2.7	W4, C16	2.1
F14,	2.5	F15	2.2
F6	1.9	F7	2.2

* more than one location designator may be listed for a measurement as some measurement locations coincide with more than one location designator. See scale drawings.

$\overline{X}$ = 2.39 urem/hr SD = 0.34

ENCLOSURE 10 (cont)

3. Exposure Room

Location*	Gross Reading (urem/hr)	Location*	Gross Reading (urem/hr)
F1,N1,W7	26	F2,W6	26
F3,W5	31	F4,W4	41
F5,W3	31	F6,W2	25
F7,W1,S9	22	F8,N2	21
F9	25	F10	30
F11	28	F12	27
F13	30	F14,S8	28
F15,N3	25	F16	25
F17	23	F18	29
F19	28	F20	22
F21,S3	26	F22,N4	32
F23	34	F24	30
F25	32	F26	28
F27	26	F28	30
F29,N5	35	F30	31
F31	33	F32	30
F33	30	F34	32
F35	28	F36,N6	30
F37	28	F38	32
F39	31	F40	27
F41	33	F42,S5	33
F43,N7,E1	30	F44,E2	27
F45,E3	22	F46,E4	18
F47,E5	22	F48,E6	28
F49,E7,S4	28	E8,N14	30
E9	31	E10	25
E11	23	E12	30
E13	28	E14,S10	29
E15,N21,C43	33	E16,C44	33
E17,C45	28	E18,C46	26
E19,C47	30	E20,C48	31
E21,C49, S16	38	N8,W14	20
N9	28	N10	31
N11	31	N12	32
N13	34	N16,C8	not taken
N17,C15	35	N18,C22	36
N19,C29	38	N20,C36	35
S1	not taken	S7	23
S2	not taken	S3	not taken

3. Exposure Room (cont)

Location*	Gross Reading (urem/hr)	Location*	Gross Reading (urem/hr)
S11	35	S6	28
S12	25	S13	20
S14	27	S15,W8	27
S17,C42	34	S18	35
S19	28	S20,C14	26
W15,S21,C7	20	S22	31
S23	28	S24	30
W9	30	C37	36
W10	30	W11	30
W12	31	W13	25
W16,C6	26	W17,C5	25
W18,C4	25	W19,C3	25
W20,C2	22	W21,C1,N15	not taken
C38	35	C39	30
C40	25	C41	27
C30	35	C31	38
C32	30	C33	29
C34	30	C23	40
C24	35	C25	30
C26	28	C27	32
C16	30	C17	30
C18	30	C19	30
C20	28	C9	30
C10	30	C11	30
C12	30	C13	26

$\bar{X} = 28.9$ urem/hr SD = 5

* more than one location designator may be listed for a measurement as some measurement locations coincide with more than one location designator. See scale drawings

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RISK MANAGEMENT BRANCH

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AMSRL-CS-AL-RK (385-11p) 7 Nov 1996

MEMORANDUM FOR RECORD

SUBJECT: DORF Contamination Review

1. In preparation for the tasking coming from DA USANCA to conduct surveys of the decommissioned DORF facility, I have visited DORF, and reviewed ARL's old DORF files. From this review, I have ascertained the following important points:

a. Applied Clean-up Limits. The clean-up was conducted applying fixed and removable <u>surface</u> radiation levels that meet today's limits. Unfortunately, a new limit has since been added that specifies that the gamma radiation level at 1 meter <u>from the surface</u> can not exceed 5 uR/hr. It is unclear at this time whether the decontamination achieves compliance with this new limit. This is the basis of USANCA's concern.

b. The pre and post clean-up contamination surveys that were conducted were relatively extensive and thorough, and well documented in Rockwell's DORF Decommissioning Program Final Report.

c. The pre and post clean-up contamination surveys indicate there is no need for concern for non-fixed (removable) contamination. All surfaces, water, soil, and vegetation were found to be non-fixed contamination free, both pre and post clean-up.

d. The pre and post clean-up contamination surveys indicate that there was significant neutron activation of the exposure room concrete and pool (tank) concrete. Further details are as follows:

(1) Both pre and post clean-up, the activation appeared to be relatively uniform over relatively large areas of concrete. The post clean-up levels, measured at 1 cm from the concrete surface through 7 mg/cm^2 was about 0.1 mrad/hr. It is unclear whether a beta particle field was present to contribute to the measured levels. If so, the gamma component may have been in fact very small. If a beta field was not present, the uniform levels would indicate that the concrete is a planar source of gamma, and the radiation level would not be expected to diminish quickly with distance. Compliance with the 5 uR/hr limit would therefore have been doubtful.

(2) Pre and post clean-up core samples of the activated concrete were taken, and definitely demonstrated the presence of neutron activation in the concrete. The primary long-lived isotopes present were Co-60 (5.27 yr), Eu-152 (12.2 yr), and Eu-154 (16 yr). The post clean-up levels present were approximately 10-100 pCi/gm. These levels are either NRC license exempt concentrations, or in the 15 mrem/yr range that the NRC has recently accepted as suitable for unrestricted area release. There has of course

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been some decay in the meantime, but probably not enough to drop the radiation levels below 5 uR/hr at 1 meter if the levels measured were due solely to gamma from these radionuclides, without contribution from a beta field.

(3) Activation in the rebar in the concrete was not assayed pre or post clean-up. It is therefore unknown how much contribution, if any, other isotopes in the rebar might have contributed to the radiation levels measured at the surface of the concrete. If the radiation levels at concrete surface were partially due to medium-lived isotopes in the rebar, such as Fe-59 (46 day), the radiation levels today may be much smaller due to the decay of these medium-lived isotopes.

2. The simplest way to proceed at this time is to make uR/hr measurements at 1 meter from the exposed concrete surfaces. Since the pool was back filled with rubble, and then sealed off with concrete, the concrete surfaces in the pool are not accessible for survey. Measurements over the concrete surfaces in the exposure room could be considered representative of concrete in the pool, especially because they were equally neutron activated. This measurement method would account for gamma coming from Co-60, Eu-152, and Eu-154. It would also account for any previously unidentified gamma emitters present in the rebar.

3. It may prove difficult to discern an additional 5 uR/hr above a background that is generally about 20 uR/hr. It may also be difficult due to the presence of the rad waste in storage located in the building and exposure room. If this turns out to be the case, it may be necessary to use a GeLi MCA to measure selectively for Co-60, Eu-152, and Eu-154, and then calculate the exposure rate level present due to these radioisotopes. In this case, it will be necessary to use the MCE to ensure that other radioisotopes are not present in the rebar, contributing to the exposure rate at 1 meter. If isotopes are detected in the rebar, it would be necessary to include these in the calculation.

4. The above will be discussed with the radiation control committee, and a plan to proceed will be derived.

Michael Borisky
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Health Physicist

601-07

ESG-80-23

DIAMOND ORDNANCE RADIATION FACILITY

DECOMMISSIONING PROGRAM

FINAL REPORT

Prepared

for

Department of the Army

Contract Number DAAK 21-79-C-0136

Rockwell International

Energy Systems Group
8900 De Soto Avenue
Canoga Park, California 91304

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DIAMOND ORDNANCE RADIATION FACILITY

DECOMMISSIONING PROGRAM

FINAL REPORT

by

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Energy Systems Group
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CONTRACT: DAAK 21-79-C-0136
ISSUED: JULY 7, 1980

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ABSTRACT

The Atomics International (AI) Division of the Energy Systems Group (ESG) of Rockwell International was contracted by the Department of the Army to dismantle and decontaminate the Diamond Ordnance Radiation Facility (DORF) located at the Forest Glen Section of Walter Reed Army Medical Center in Silver Spring, Maryland. The contract was for a firm fixed price with a schedule duration of 8 months.

All the contracted terms specified in DAAK 21-79-C-0136 were fulfilled within the required schedule and budget. There was no significant radiation exposure to personnel or internal deposition of radioactive material as a result of decommissioning the Diamond Ordnance Radiation Facility.

I. INTRODUCTION

The objective for dismantlement and decontamination of radioactivity of the Diamond Ordnance Radiation Facility (DORF) was to make the facility acceptable for unrestricted use by removing radioactivity to levels below those requiring surveillance and licensing.

Dismantling the reactor and removing the radioactive components was the mode selected for decommissioning DORF. Specifically identified reactor components were dismantled, packaged, and shipped to Westinghouse's Hanford Engineering Development Laboratories (HEDL) in Richland, Washington. The pool tank, lead shield doors, lead shield hoist, exposure room wood lining, rolling shield door, and activated concrete were dismantled, removed from the facility, and disposed to clean salvage/disposal or to radioactive disposal.

The regulatory agency governing operations at DORF was the U.S. Army. The Army specified Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86 as the governing document for the decommissioning activity. This guide specifically requires decontamination to levels which are as low as reasonably achievable (ALARA), but in all cases to levels below those listed in Table 1. To show compliance with ALARA, Rockwell established the limits shown in Table 2 as a target. These limits are based on experience regarding levels that in most cases are reasonably achievable and can be effectively monitored.

Radioactive materials and components which exceeded Table 1 limits were removed from the facility. The limits shown in Table 2 were also met in all areas of the facility except in the exposure room where, due to room geometry and the accumulative properties of activation products, the activity ranged from 0.08 to 0.24 mrad/h as measured with a Technical Associates Mark III Cutie Pie - CP7M. The overall average was slightly higher than 0.1 mrad/h. Individual pieces of concrete from the higher activity areas, when removed from the exposure room, indicated levels below 0.1 mrad/h. These activity levels were deemed acceptable by the contracting officer's representative and by the United States Army Environmental Health Agency (USAEHA) radiation survey team.

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS FROM NRC REGULATORY GUIDE 1.86

Nuclide*	Average†§	Maximum†**	Removable†,††
U^{nat}, U^{235}, U^{238}, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra^{226}, Ra^{228} Th^{230}, Th^{228}, Pa^{231}, Ac^{227}, I^{125}, I^{129}	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th^{nat}, Th^{232}, Sr^{90}, Ra^{223} Ra^{224}, U^{232}, I^{126}, I^{131}, I^{133}	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr^{90} and others noted above.	5,000 dpm $\beta\gamma$/100 cm^2	15,000 $\beta\gamma$/100 cm^2	1,000 dpm $\beta\gamma$/100 cm^2

*Where surface contamination by both alpha- and beta-gamma-emitting nuclides exist, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

†As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

§Measurements of average contaminant should not be averaged over more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

**The maximum contamination level applies to an area of not more than 100 cm^2.

††The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

TABLE 2

CONTAMINATION LIMITS FOR DECONTAMINATION AND DISPOSAL OF DORF

	Total	Removable
Beta-Gamma Emitters	0.1 mrad/h average* and 0.3 mrad/h maximum† at 1 cm with 7 mg/cm^2 absorber	100 dpm/100 cm^2
Alpha Emitters	100 dpm/100 cm^2	20 dpm/100 cm^2

*Measurements of average contaminant should not be averaged over more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

†The maximum contamination level applies to an area of not more than 100 cm^2.

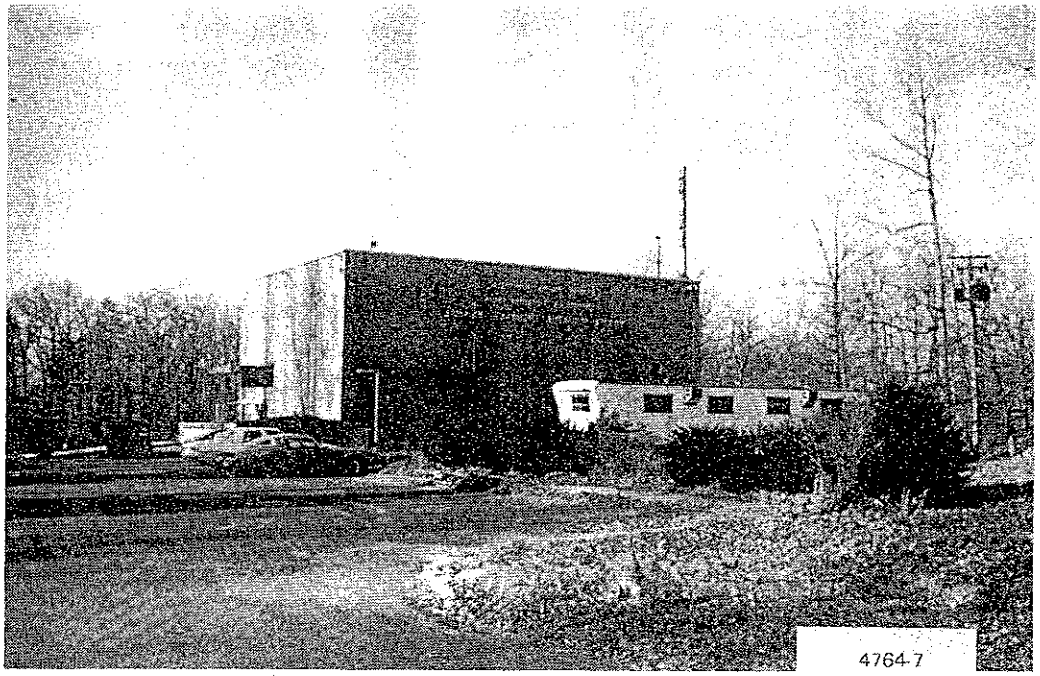


Figure 1. Exterior View of DORF

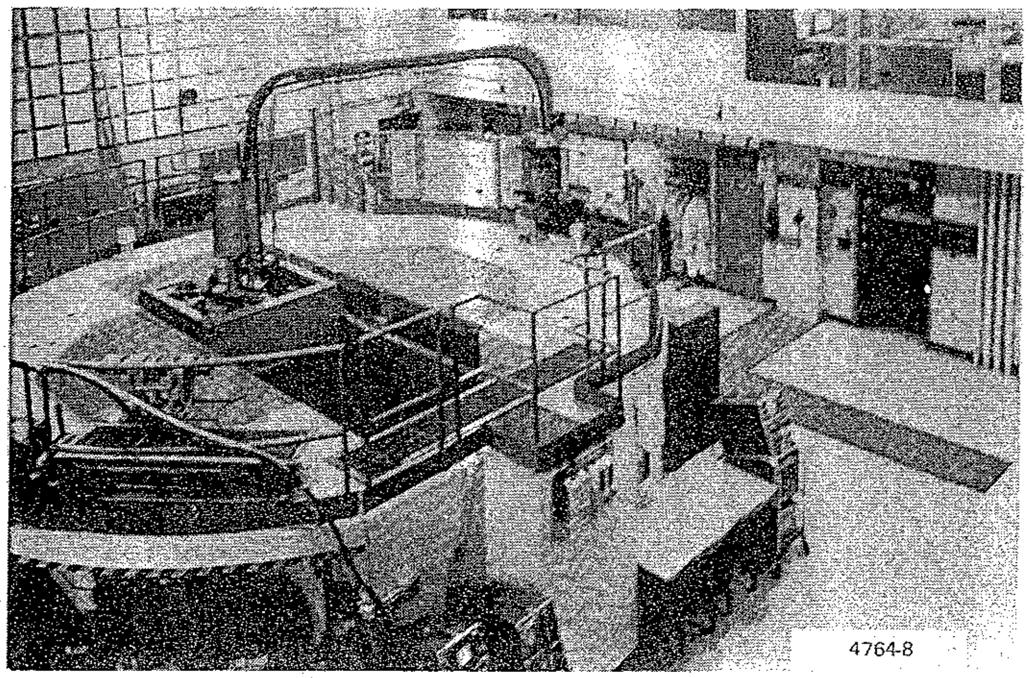


Figure 2. Interior View of DORF Showing TRIGA Reactor Carriage and Control Drive Housing Located on Parapet

II. FACILITY DESCRIPTION

The Diamond Ordnance Radiation Facility (DORF), Figure 1, was operated by the Department of the Army's Harry Diamond Laboratories (HDL). The facility housed a TRIGA Mark F Reactor, Figure 2, as the principal research tool in the study of neutron and gamma radiation effects on electrical and electronic components.

DORF is located within the metropolitan area of Washington, D.C. at the Forest Glen section of the Walter Reed Army Medical Center (WRAMC), which is 8 miles due north of the center of Washington, D.C. The building containing the reactor is 65 ft by 50 ft and 25 ft high. It is encircled by an exclusion fence with a radius of about 240 ft. Access to the 4.2-acre site is controlled at a single entrance gate.

The reactor was designed and built by Gulf General Atomics, San Diego, California. It was designed for both steady-state and pulsed operation with a design capability of:

1) Steady-state or square-wave operation up to 250 kW for a maximum power generation of 1 mW-h/day.

2) Pulsed operation resulting in a peak power of 2,000 MW with a pulse width of 9.5 ms at half maximum.

On September 18, 1961, the DORF-TRIGA Mark F reactor achieved criticality for the first time. The first core was aluminum clad, but it was replaced with a stainless steel clad core in 1964. This stainless steel clad core was operated from 1964 through September of 1977, when reactor operations at DORF were terminated. An estimate of the burnup on the core at the time of shutdown was 0.48% based on 242,451 kWh of operation.

In the spring of 1979, the core was removed from the reactor. It was dispositioned to several university programs and to the DOE-Hanford Engineering Development Laboratories in Richland, Washington.

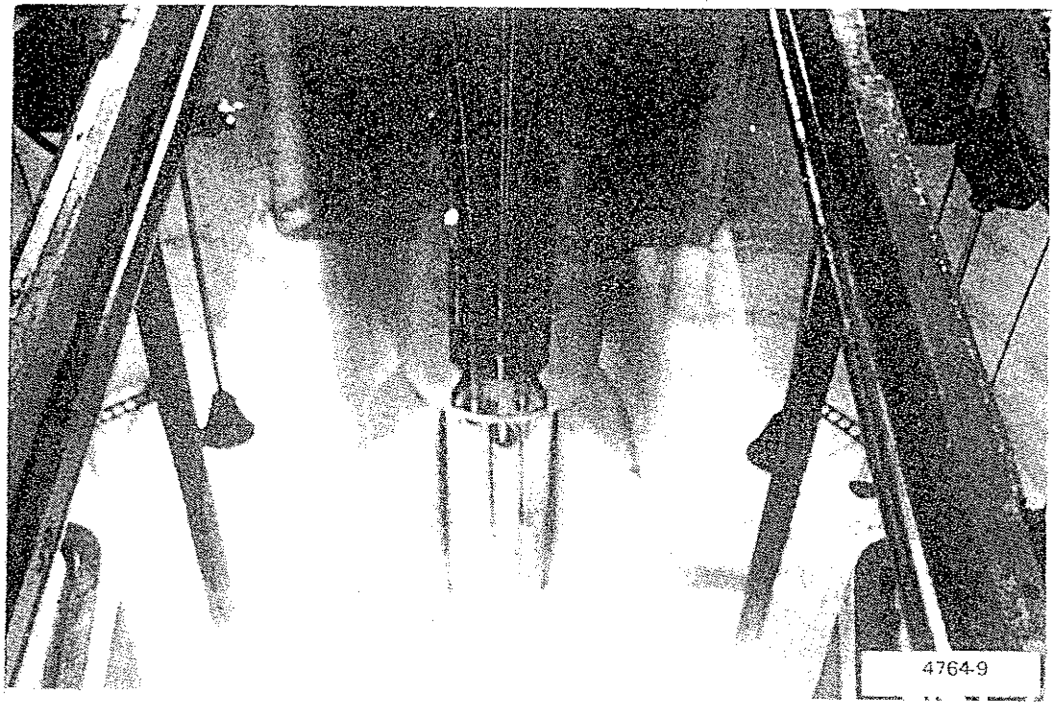


Figure 3. View of Reactor Core Housing and Support Structure

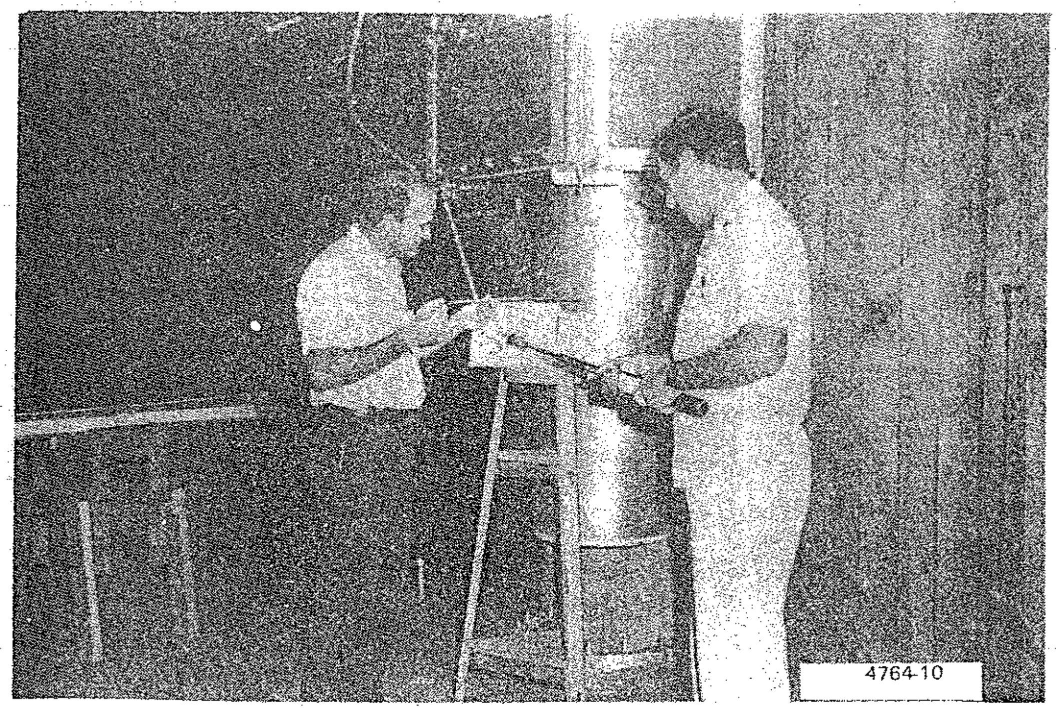


Figure 4. Test Setup in Exposure Room

The reactor core, Figure 3, was located near the bottom of a 15,000-gal aluminum tank which was about 13 ft in diameter and 20 ft deep. The core was suspended by a support structure from a motor driven carriage mounted on rails at the top of the tank. The carriage was capable of traversing the tank to enable the reactor to be positioned behind lead doors so that entry could be made into the exposure room immediately after a test. Figure 4 shows a typical test setup in the exposure room. Figure 5 is a diagram showing a cross-section view of the facility and the relative position of reactor to exposure room. With the lead doors open, the reactor could traverse the tank to a position by the lead shield.

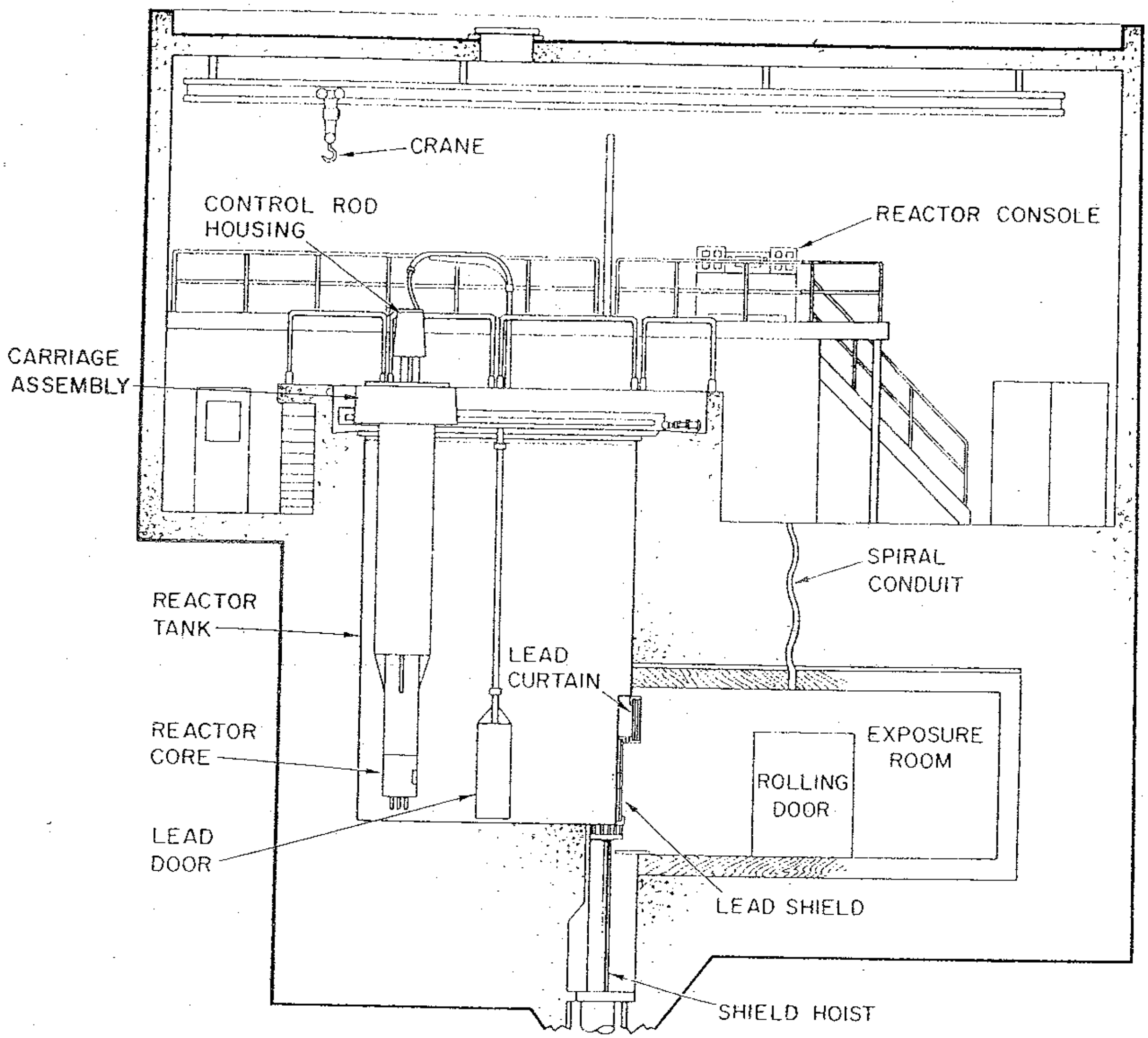


Figure 5. Vertical Section of DORF Reactor

III. SUMMARY OF DECOMMISSIONING ACTIVITIES

The activities which comprise the decommissioning of DORF were grouped into three phases. Phase I consisted of the planning, procurement, and staffing activities required to conduct Phases II and III. Phase I was conducted in Canoga Park, California. Phase II consisted of those activities required to remove and dispose of the radioactive and nonradioactive components and materials identified in the RFQ. Phase III consisted of the demolition of nonradioactive portions of the facility. Phases II and III were conducted in Silver Spring, Maryland.

A. PHASE I

Facilities Dismantling Plan for DORF N001-FDP-960-001 was prepared to delineate the activities necessary to achieve the stated objectives. These were categorized as: planning, monitoring, and control; radiological survey; dismantlement and disposal; and documentation. This dismantling plan was reviewed and approved by the Rockwell D&D Program Office, Health, Safety and Radiation Services, and by the Engineering Department. It was then reviewed and approved by the Army Reactor Committee for Health and Safety (ARCHS).

Activities concurrent with planning were: (1) the acquisition of equipment, tools, and material; (2) placement of service contracts; and (3) the recruitment and training of personnel. Phase I activities were initiated on September 17, 1979, and were completed on November 21, when ARCHS approved the dismantling plan.

B. PHASE II

Phase II was initiated on November 26, 1979, with the movement of personnel to the DORF site in Silver Spring, Maryland, and was completed on February 22, 1980, with the return and reassignment of personnel to other projects.

Phase II consisted of the following activities: (1) site preparation, (2) packaging and shipping reactor components to HEOL, (3) exposure room dismantlement, (4) pool tank removal, (5) concrete excavation, (6) site survey, and (7) waste disposal.

1. Site Preparation

Site preparation included those activities required to move the Rockwell staff and their equipment to the DORF site and to establish a base of operations in Maryland. A radiological survey of the nonradioactive portions of the site was conducted for documentation and an analysis of the pool water was performed to determine compliance with 10 CFR 20.303.

On November 26, 1979, a six-man team from Rockwell International's Energy Systems Group in Canoga Park, California, arrived in Maryland to begin the Phase II work outlined in the contract. A base of operations was established within the first week including a site radiological survey. An agreement was made with Holy Cross Hospital in Silver Spring wherein they would accept for treatment any radioactively contaminated person from DORF.

A radiological survey taken of one of the floor drains adjacent to the parapet near the main experimental area indicated activity in the range of 250 cpm $\beta\gamma$. The floor grating over the drain was removed and the radioactive residue was vacuumed into an approved radioactive waste container. Resurvey of this drain and all other areas of the facility outside the exposure room indicated levels of activity well below those listed in Tables 1 and 2.

Water samples from the pool tank were analyzed by Teledyne Isotopes, WRMAC, and Rockwell. The data are shown in Table 3. These data show the water to be well within the allowable limits given in 10 CFR 20, Appendix B, Table 1, Column 2. Walter Reed Hospital's Health and Safety Branch granted Rockwell permission to drain the water through their sanitary sewer system.

2. Packaging and Shipping Reactor Components to HEDL

The TRIGA reactor and its components were disassembled, packaged, and shipped to DOE-Hanford Engineering Development Laboratories (HEDL), Richland,

TABLE 3

ANALYSIS OF POOL WATER

	μCi/ml
Rockwell	4.4×10^{-9} βγ 6.85×10^{-10} α
Teledyne Isotopes	$<1 \times 10^{-9}$ gross β 1.41×10^{-6} H-3
WRAMC	<Detectable gross β 5×10^{-7} H-3

Note: 10 CFR 20 limits were interpreted to be as follows: 4×10^{-7} μCi/ml βγ; 4×10^{-7} μCi/ml α; 3×10^{-3} μCi/ml H-3

Washington. The packages and shipment conformed to Department of Transportation (DOT) specification, Title 49, Code of Federal Regulations (49 CFR).

The reactor and components were disassembled to the degree necessary to permit packaging. All of the items listed in Table 4 were removed, packed into weatherproof containers, and transported to HEDL. Figures 6 and 7 show reactor and component disassembly. Figures 8 and 9 show packaging activities. Figure 10 shows packages loaded into a truck for shipment.

Each package was monitored by the Health Physicist to determine its radioactive content. Only one of the containers had significant detectable radiation at the surface. It was Container No. 158, a DOT-type A-7A drum containing the 10 Ci americium-beryllium neutron source. Its radiation measured 120 mrad/h neutron-beta-gamma at the surface and 4 mrad/h at 1 m. All of the other containers were <10 mrad/h at the surface and near background at 1 m. Table 5 is a list of containers, their volumes, weights, and contents.

TABLE 4

LIST OF REACTOR COMPONENTS SHIPPED TO HEDL

Item No.	Description	Unit	Quantity
1	Core Support Structure, Upper Section	Each	1
2	Core Support Structure, Lower Section	Each	1
3	Top and Bottom Grid Plates	Each	1
4	Connecting Rods for Control Rods	Set	1
5	Control Rods	Set	1
6	Carriage Drive Motor	Each	1
7	Water Pump: 1.5 hp	Each	1
8	Incore Experiment Tube	Each	1
9	Ion Chamber Support and Ion Chambers	Set	3
10	Carriage Support Rails	Set	1
11	Lead Shield Door Drives and Linkage	Set	1
12	Pool Cover Plates	Set	1
13	Fuel Storage Racks, Underwater	Each	8
14	Fuel Measurement Tool with Dial Micrometer	Each	1
15	Aluminum Water System Piping	Each	1
16	Water Pumps	Each	3
17	Demineralizers, 3 ft^3	Each	4
18	Flowmeters, 25 gpm	Each	2
19	Neutron Source, 10 Ci, Am-Be	Each	1
20	Neutron Source Holder	Each	1
21	Pool Lights	Set	1
22	Carriage Positioning Potentiometer	Each	1
23	Carriage Umbilical Arm	Each	1
24	Fuel Element Location Diagram	Each	1
25	Water Box, 1 ft^3 Capacity	Each	1
26	Charcoal Filter, 1 ft^3 Capacity	Each	1

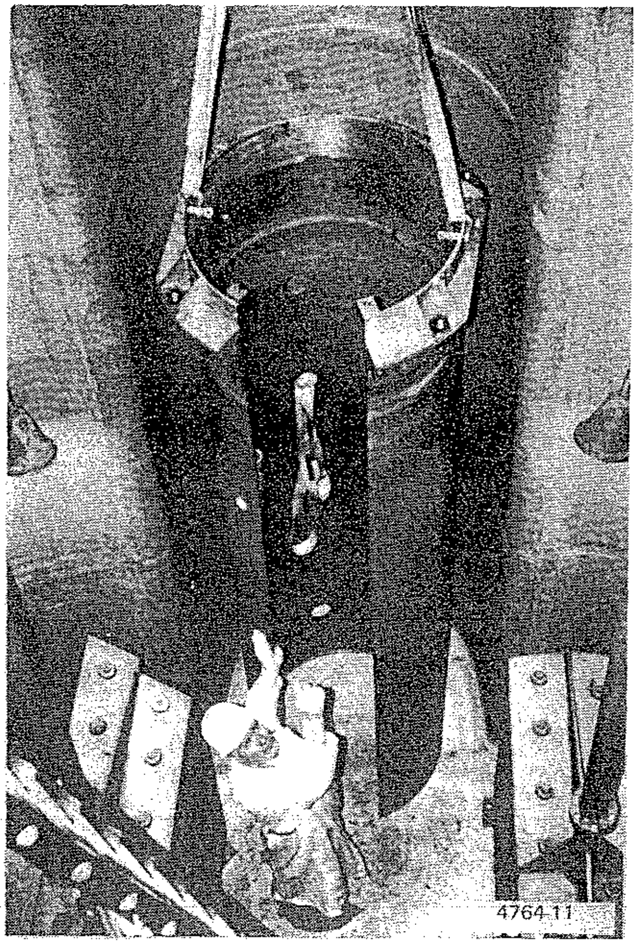


Figure 6. Removal of Reactor Support Structure From Pool Tank

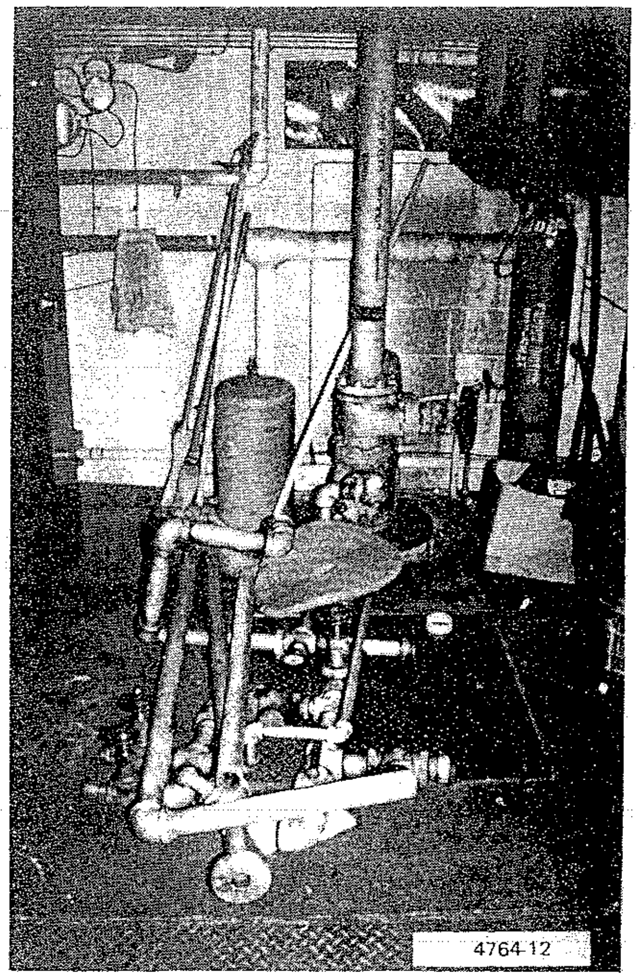


Figure 7. Reactor Cooling System Piping After Disassembly but Prior to Packaging

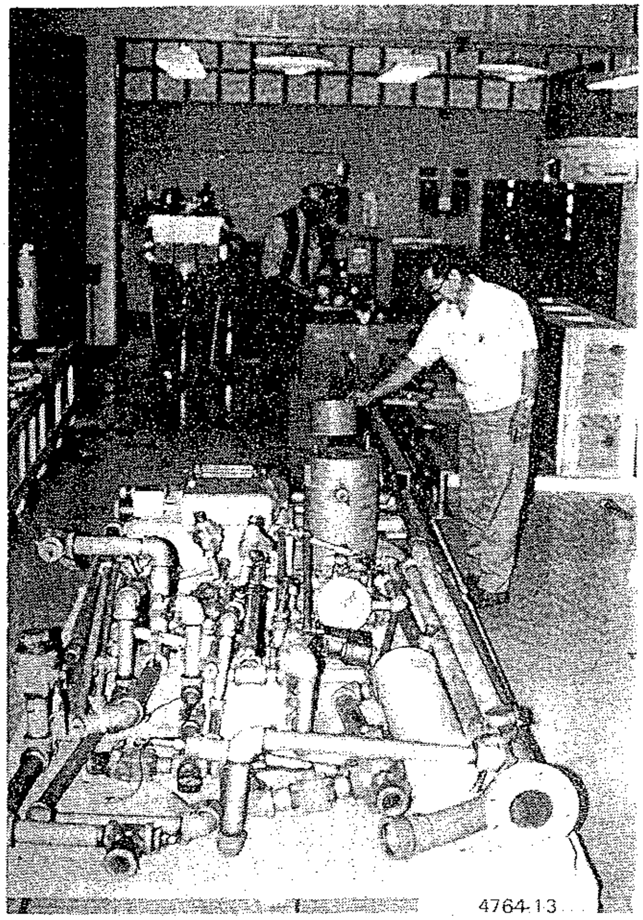


Figure 8. Reactor Cooling System Piping Partially Packaged (Box No. 156)

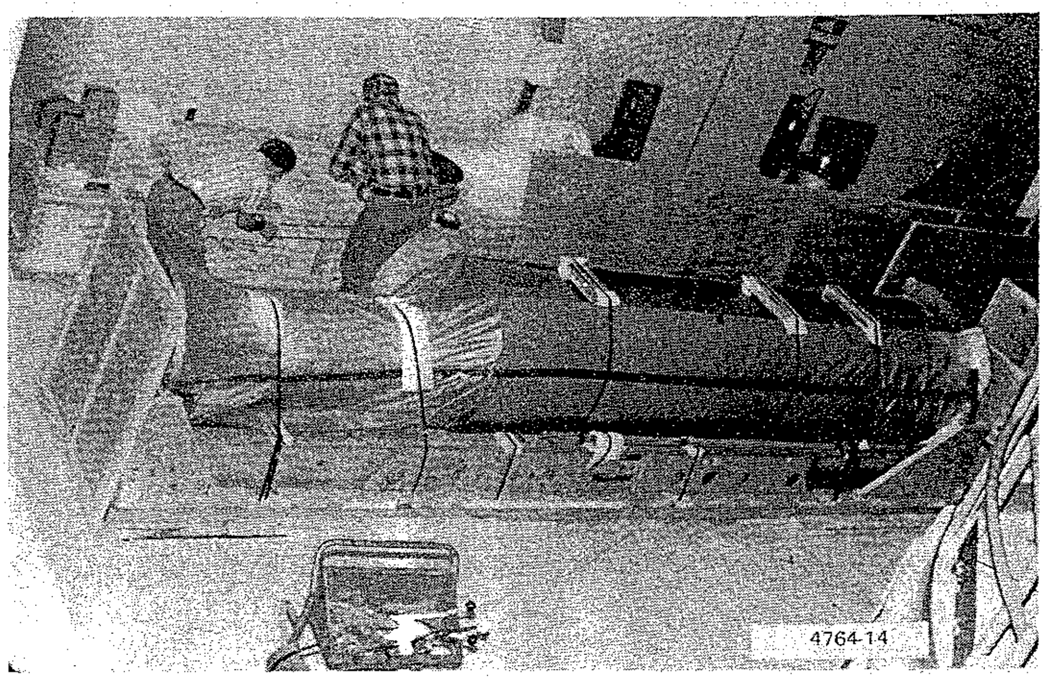


Figure 9. Partially Packaged Reactor Core Support Structure (Box No. 154)

TABLE 5

CONTAINER PACKING LIST FOR HEDL SHIPMENT
(Sheet 1 of 2)

		Quantity
Box No. 151 (38 ft^3, 1100 lb)		
No. 11	Lead Shield Door Drive and Linkage	
	Motor and Clutch	1
	Transmission Tee	1
	Right Angle Transmission	2
	Door Transmission	2
	Short Shaft	2
	Long Shaft	2
No. 16	Water Pump	3
	Seals	4 boxes
	Carriage Drive Motor (Spare)	1
Box No. 152 (112 ft^3, 700 lb)		
No. 9	Ion Chamber Supports	4 sets
No. 4	Connecting Rods for Control Rods	7
No. 5	Control Rods (2), 1 graphite	3
No. 14	Fuel Measurement Tool and Dial Micrometer	1
	Dip Leg (Water Diffuser Pump), 1 long, 1 short	2
	Standard Control Rod FFCR	2
	Ion Chamber Guide	2
	Control Rod Guide	2
	Core Thimble Guide	4
No. 20	Neutron Source Holder	1
Box No. 154 (333 ft^3, 4500 lb)		
No. 1	Core Support Structure, Upper Section	1
No. 2	Core Support Structure, Lower Section	1
No. 12	Pool Cover Plates	4
No. 13	Fuel Storage Racks	16
	Connecting Rods and Bolts	
No. 21	Pool Lights	4
No. 10	Carriage Support Rails	2
No. 24	Fuel Element Location Diagrams (Picture Frames)	3
No. 5	Control Rod and Connecting Rod	2
Box No. 155 (159 ft^3, 3200 lb)		
No. 6	Carriage	1
No. 7	Water Pump	
No. 22	Potentiometer	
No. 17	Demineralizer Tanks	4

TABLE 5

CONTAINER PACKING LIST FOR HEDL SHIPMENT
(Sheet 2 of 2)

	Quantity
Box No. 156 (192 ft^3, 1700 lb)	
No. 15 Aluminum Piping - Water System	1
No. 18 Flowmeter (NK 398-00150)	2
No. 23 Carriage Umbilical Arm	1
Post	1
No. 25 Water Box	1
No. 26 Charcoal Filter	1
Barrel Assembly (Spare)	1
Connecting Rod	4
Drum No. 157 (17H Drum) (7.5 ft^3, 200 lb)	
No. 3 Lower Core Assembly	1
Top and Bottom Grid Plates	
Drum No. 158 (17H Drum) (7.5 ft^3, 550 lb)	
No. 19 Americium-Beryllium Neutron Source	1

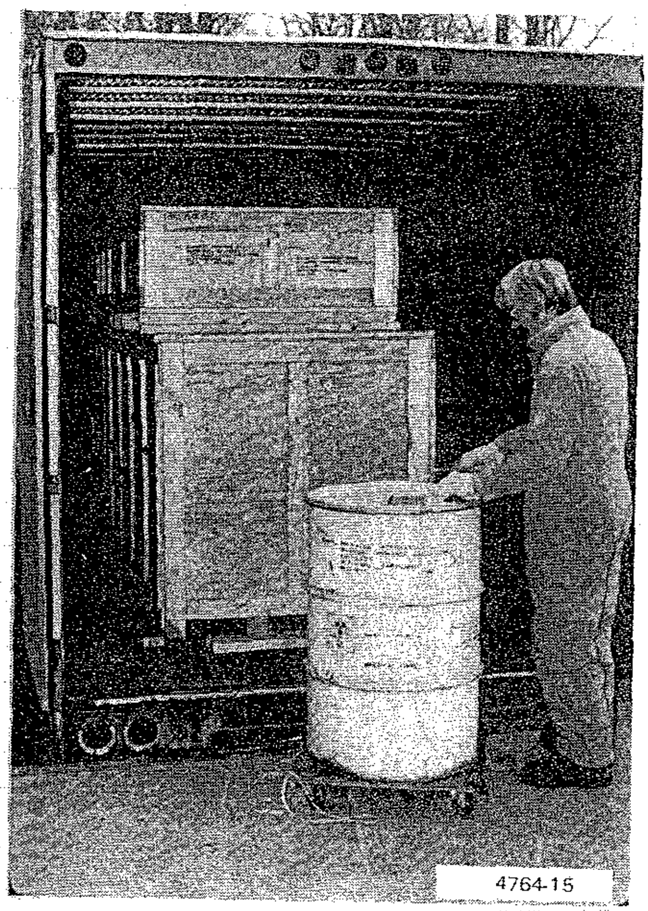


Figure 10. Truck Being Loaded with Boxes and Drums of Reactor Components for Shipment to HEDL

3. Exposure Room Dismantlement

The exposure room was stripped of its wood lining, lead shields, lead shield hoist, and other removable components. The material was separated and dispositioned based on radiological analysis.

The three floor drains were temporarily plugged to prevent transporting radioactive materials into the facilities holdup tanks. The aluminum tracks on the ceiling and the masonite covering the wood lining were removed from the exposure room. Radiation survey analysis determined that about two-thirds of the wood lining could be disposed of as clean wood, the remaining one-third was packaged and disposed of as radioactive waste. The clean wood was removed from the exposure room, put into a large dumpster, transported to a local dump site, and buried to prevent its reuse. The wood was structurally damaged as a result of neutron irradiation and might have been tempting for use as structural material if left unburied. Figures 11 through 15 show these activities. The concrete wall of the exposure room (Figure 16) was covered with the phenolic-coated tar paper listed in the RFQ as being attached to the aluminum pool tank. Two lead shields were removed from the wall adjacent to the pool tank. These were coated on one side with the phenolic-coated paper from the wall which was activated. The coating was scraped from the lead and the lead was recovered as clean scrap. Figure 17 shows the lead removal task.

Six 1-in. thick pieces of lead were removed from the top portion of the exposed pool section of the tank in the exposure room. The aluminum frame was removed from the lead and disposed of as radioactive waste. The lead was analyzed and determined to be acceptable as clean scrap. Figures 18 and 19 show these operations.

The lead shield was removed from the lead shield hoist as shown in Figure 20. This shield was activated slightly and was packaged as radioactive waste. The lead shield hoist was drained of hydraulic oil. The oil was analyzed and was found to be nonradioactive. This oil was picked up by a "reclaimed oil" processor at no cost to the program. The lead shield hoist was removed by first

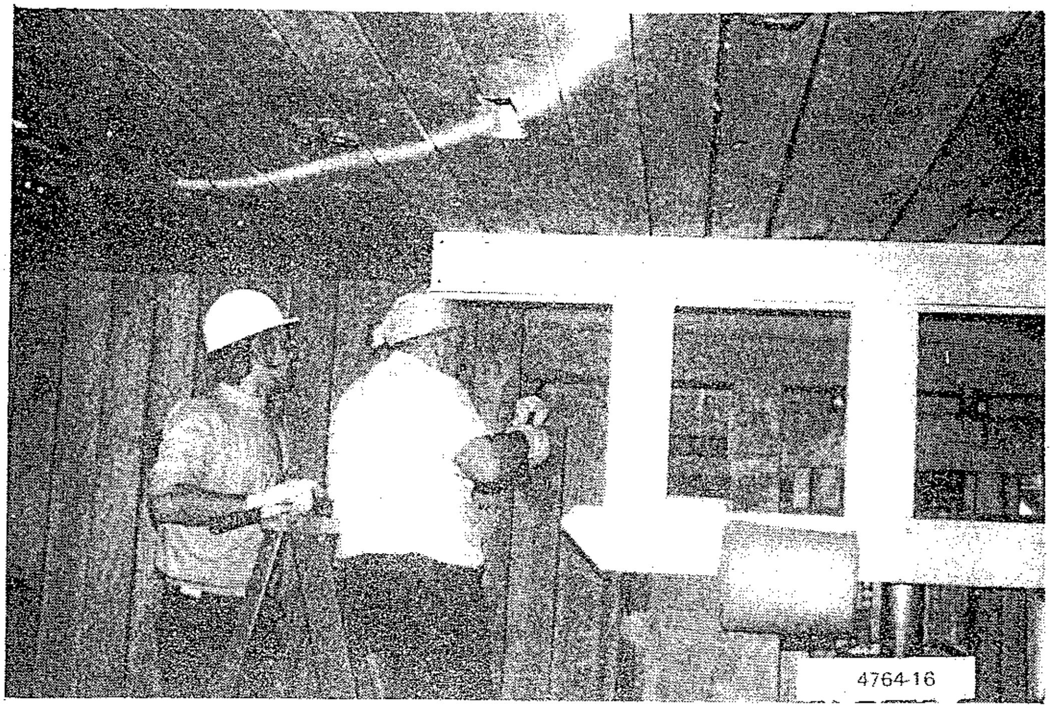


Figure 11. Start of Wood Removal From Exposure Room by First Removing Load Bearing Beam From Over Doorway While Lifting Ceiling With Cribbing

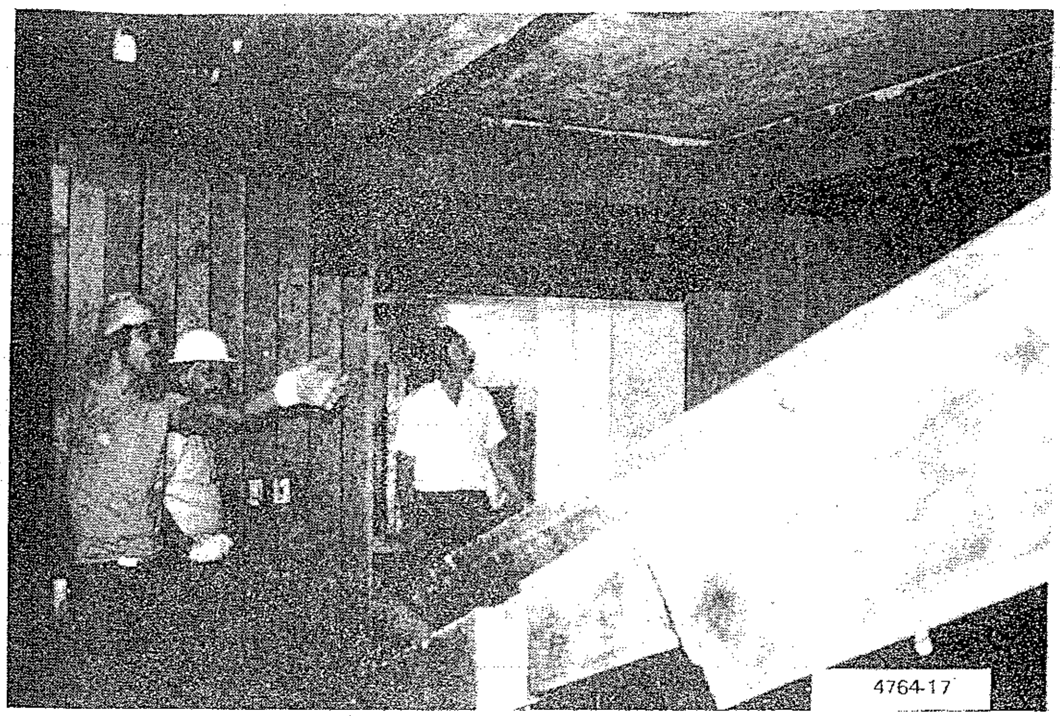


Figure 12. CeilingTimbers Partially Removed After Lowering Cribbing With Fork Lift

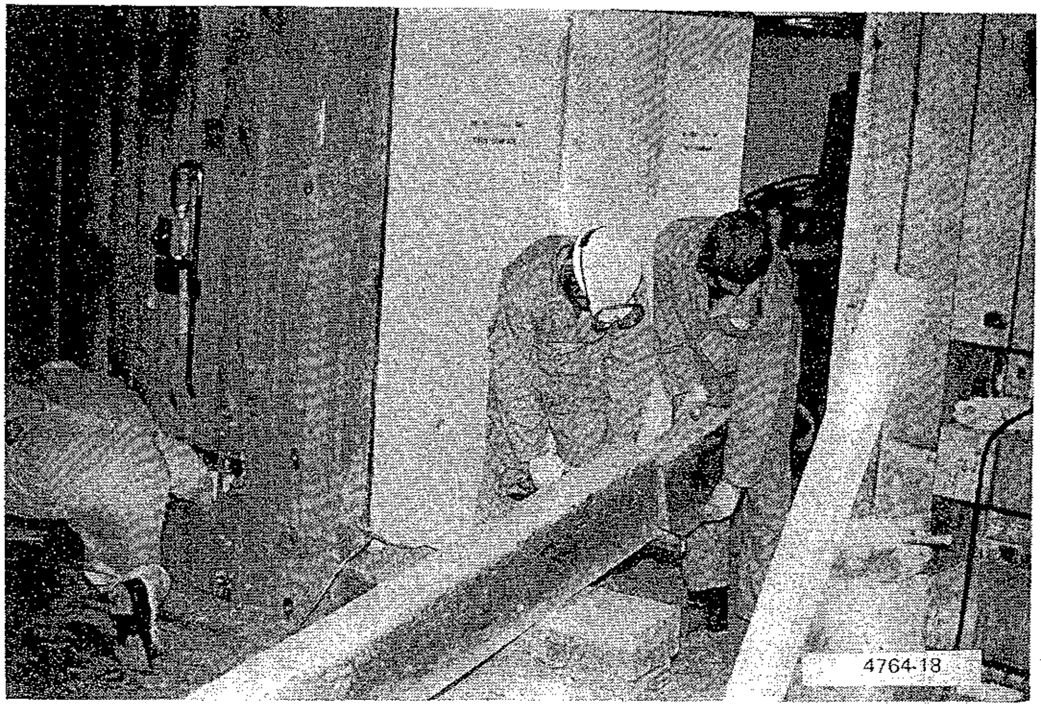


Figure 13. Using Chain Saw to Cut Wood Into Disposable Pieces

Figure 14. Clean Wood Being Loaded Into Dumpster for Disposal

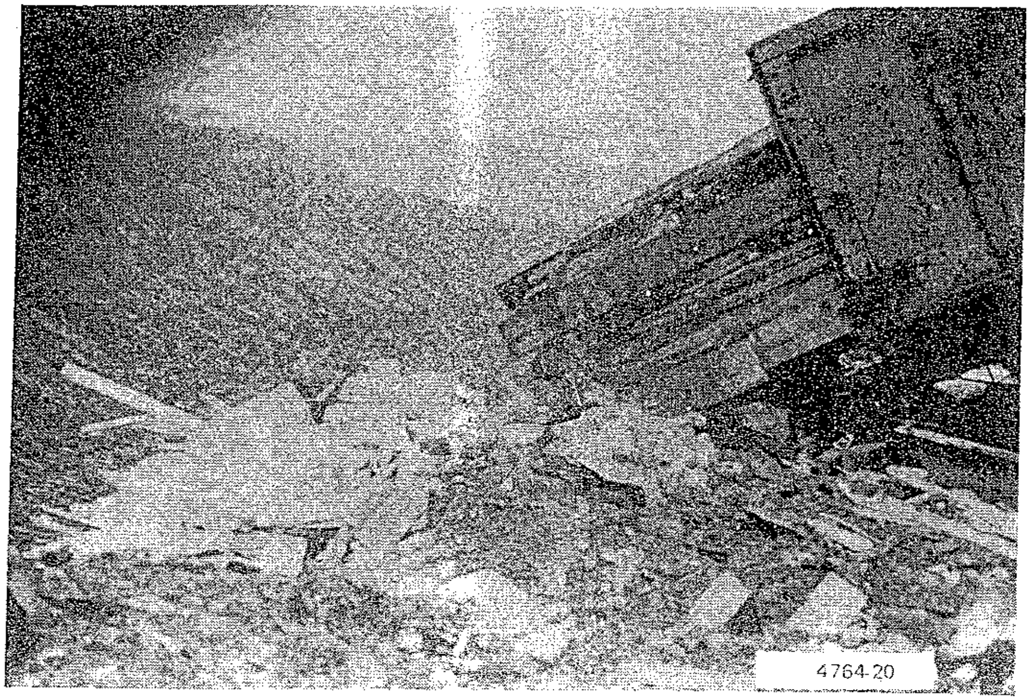


Figure 15. Clean Wood Being Dumped at Disposal Site

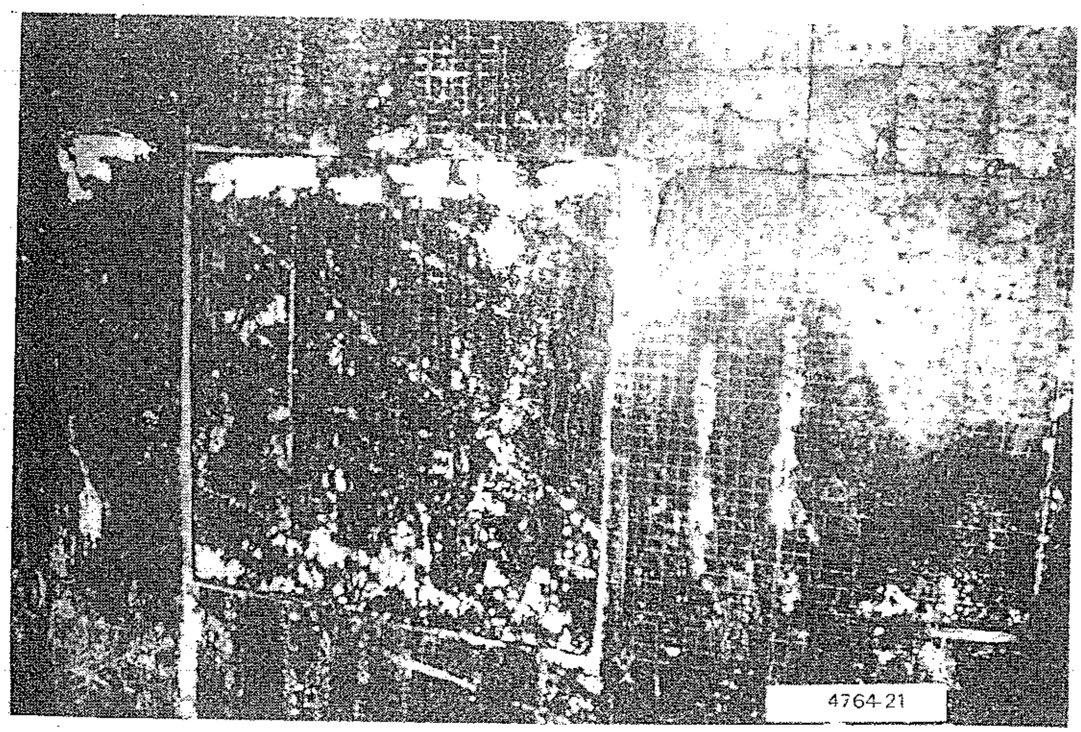


Figure 16. Phenolic-Coated Tar Paper Covering Concrete Wall of Exposure Room

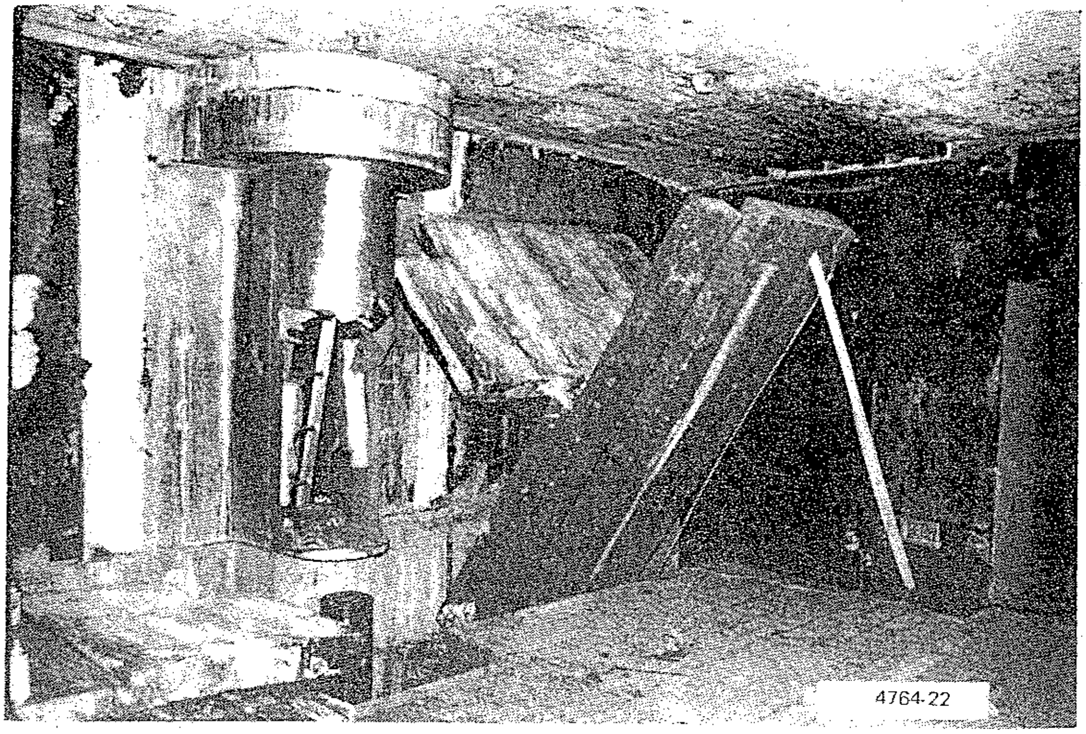


Figure 17. View of East Wall of Exposure Room Showing One Lead Shield Lying on Floor to Left and One Lead Shield in Motion on Right

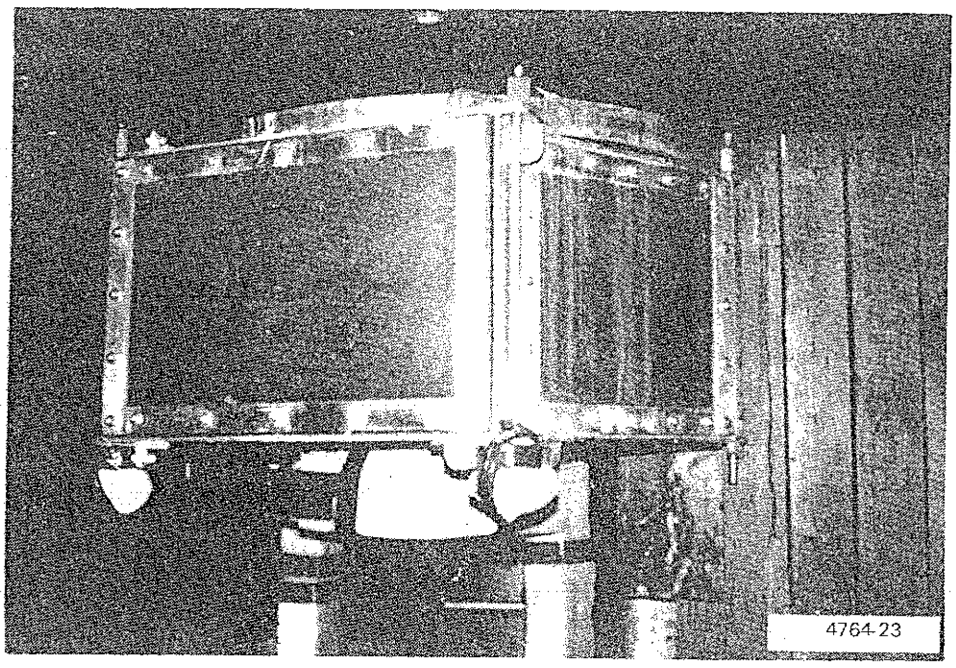


Figure 18. Lead Shields in Exposure Room Prior to Removal

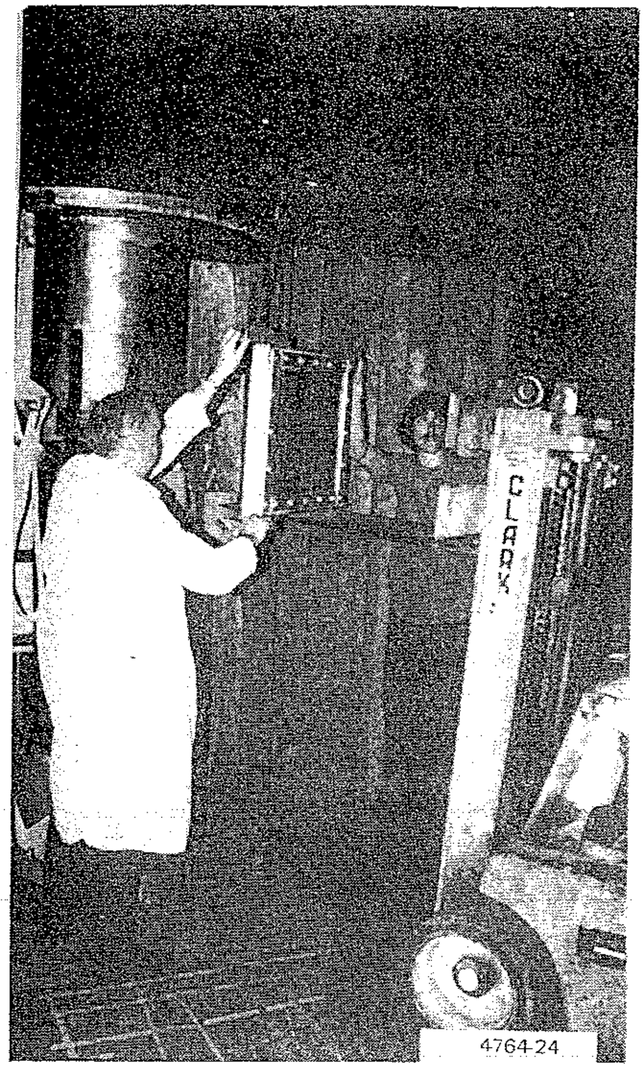
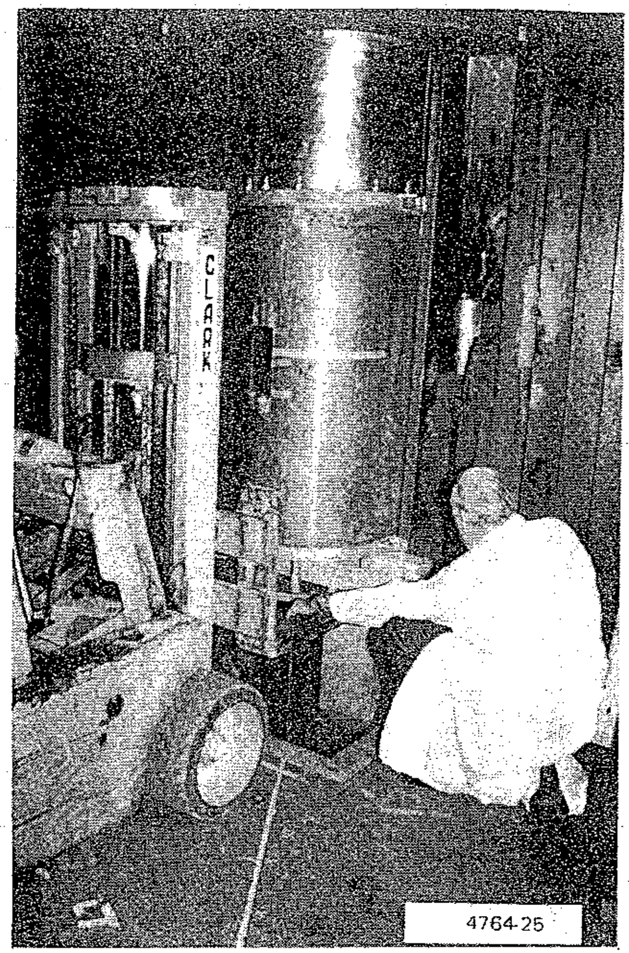


Figure 19. Shielding Being Removed From Exposure Room

Figure 20. Lead Shield Being Removed From Lead Shield Hoist



excavating the sand surrounding it (Figure 21) and then breaking the 8-in.-thick layer of concrete that surrounded its base with a jackhammer. Figure 22 shows the hoist removed. The hoist was activated and was therefore packaged and disposed of as radioactive waste.

4. Pool Tank Removal

Transformer oil and lead-shot were drained from the lead shield doors, the doors were removed from the pool tank, and the pool tank was removed from the concrete cavity.

To facilitate reactor component disassembly, an opening was cut into the pool tank to provide access to it from the exposure room. This opening was enlarged to about 7-ft^2 (Figure 23) when pool tank removal was started.

Samples of the transformer oil (Figure 24) were removed from the lead doors and analyzed by Garnett-McCreath Labs in Harrisburg, Pennsylvania, for polychlorinated biphenals (PCB). PCB concentrations were determined to be <1 ppm, a factor of 50 below the established limits for controlled disposal as given in 40 CFR Part 761. About 180 gal of oil was drained from the lead doors into four 55-gal metal drums. This oil was given to a "reclaimed oil" processor at no cost to the program.

Lead was drained from the doors into thirteen 55-gal drums. Figure 25 shows this operation. Each drum weighed about 2,150 lb or a total of 28,000 lb. Lead samples from each drum were analyzed to determine radioactive content. All samples were well under the allowable limit for release for unrestricted use. Table 6 presents the results of these analyses. When sufficient lead had been drained from the doors, they were lifted from the tank with the overhead crane and removed to a low-background area for a radiological survey. Removable and fixed contamination levels were well below limits as depicted in Table 2. The doors and the lead were disposed of as salvageable scrap.

The aluminum pool tank was cut into several sections to enable its removal from the concrete cavity. Figures 26, 27, and 28 show this activity. Each piece

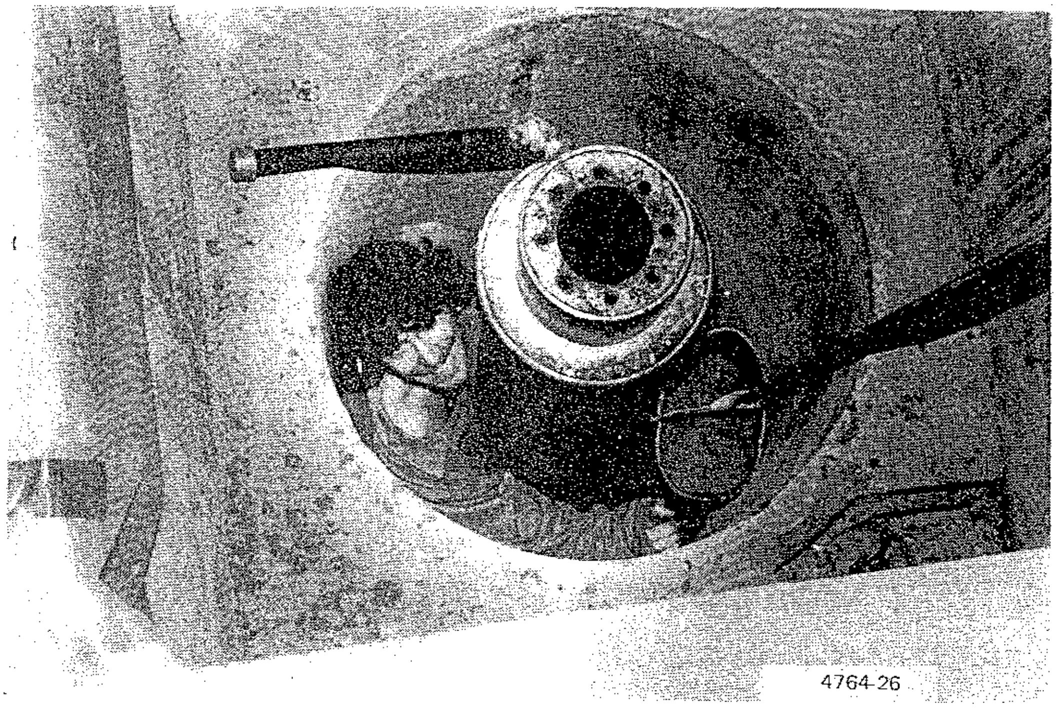


Figure 21. Excavating Sand From Around Lead Shield Hoist

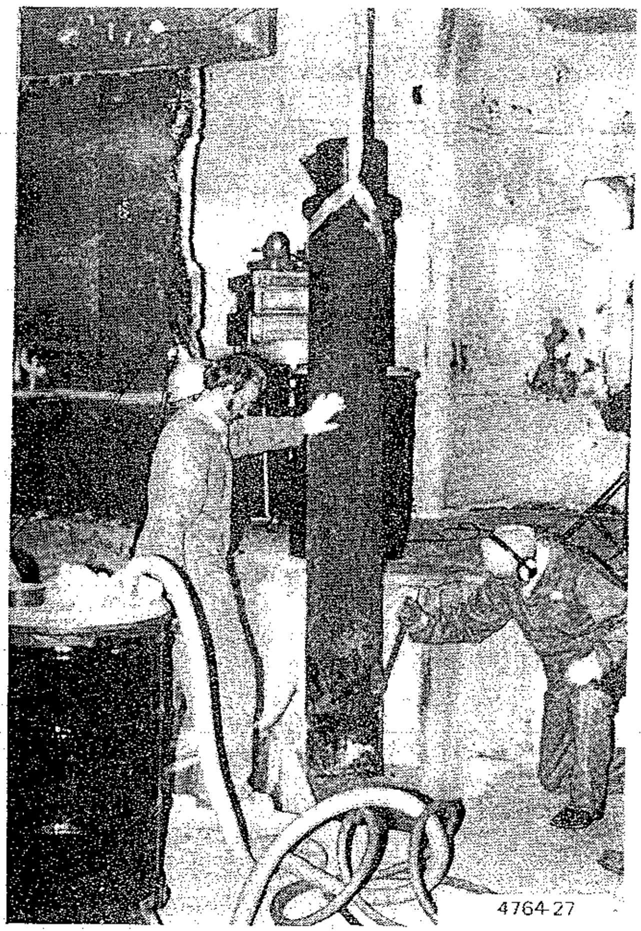


Figure 22. Removal of Lead Shield Hoist

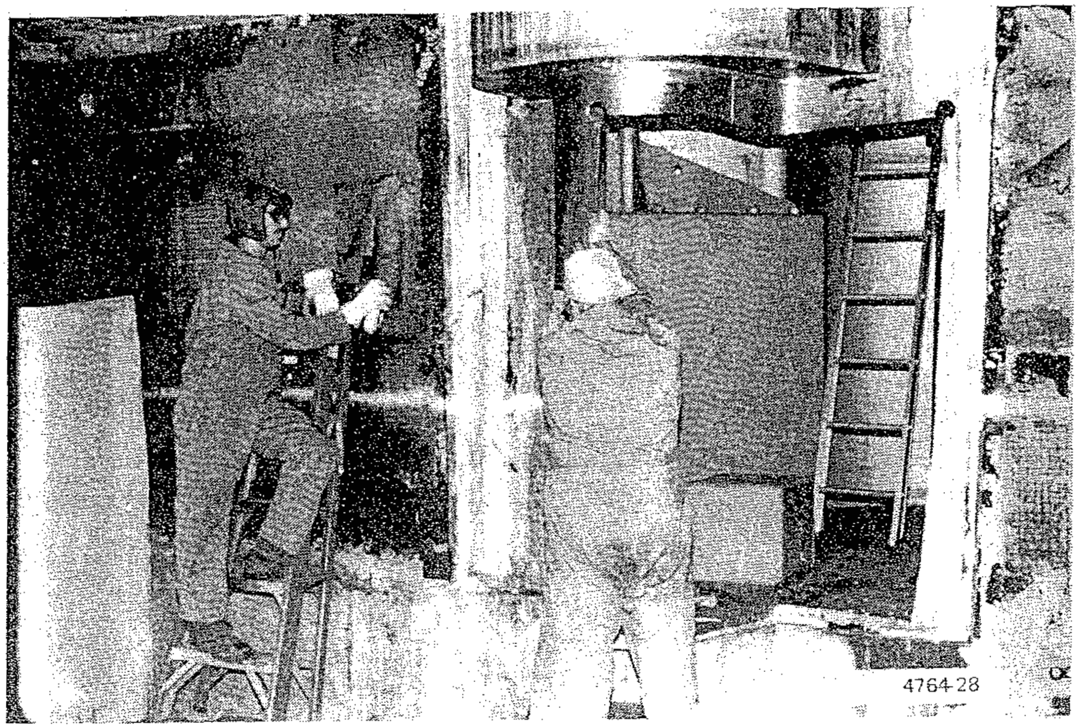


Figure 23. Cutting an Opening Into Reactor Pool Tank From Exposure Room

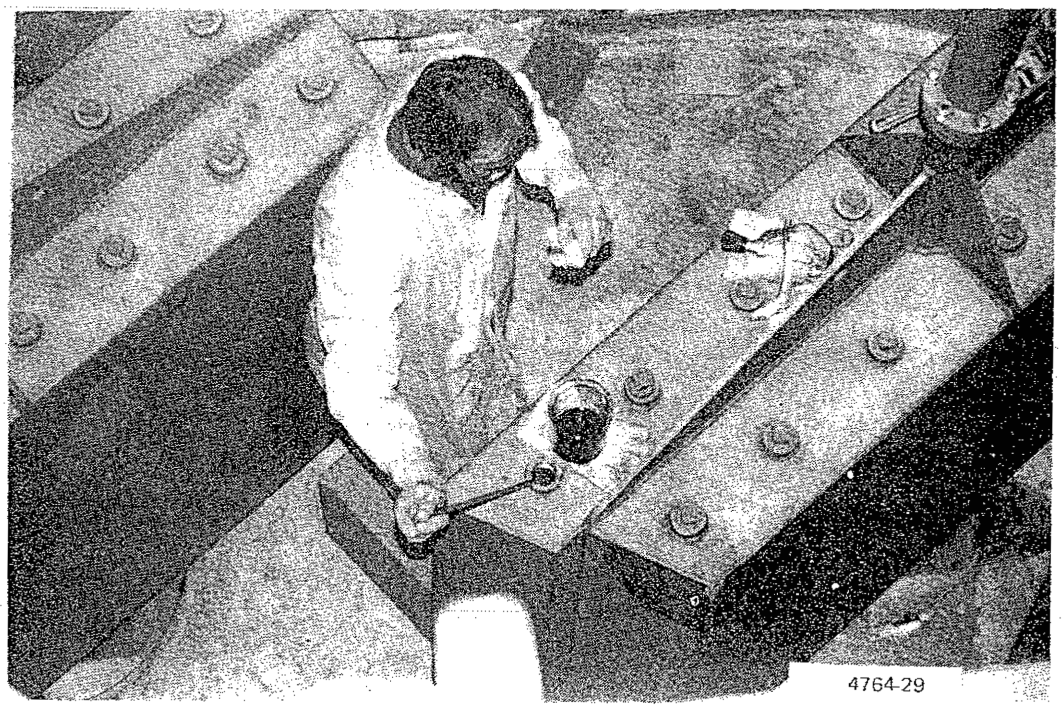


Figure 24. Sampling Transformer Oil in Lead Shield Doors for PCB Analysis

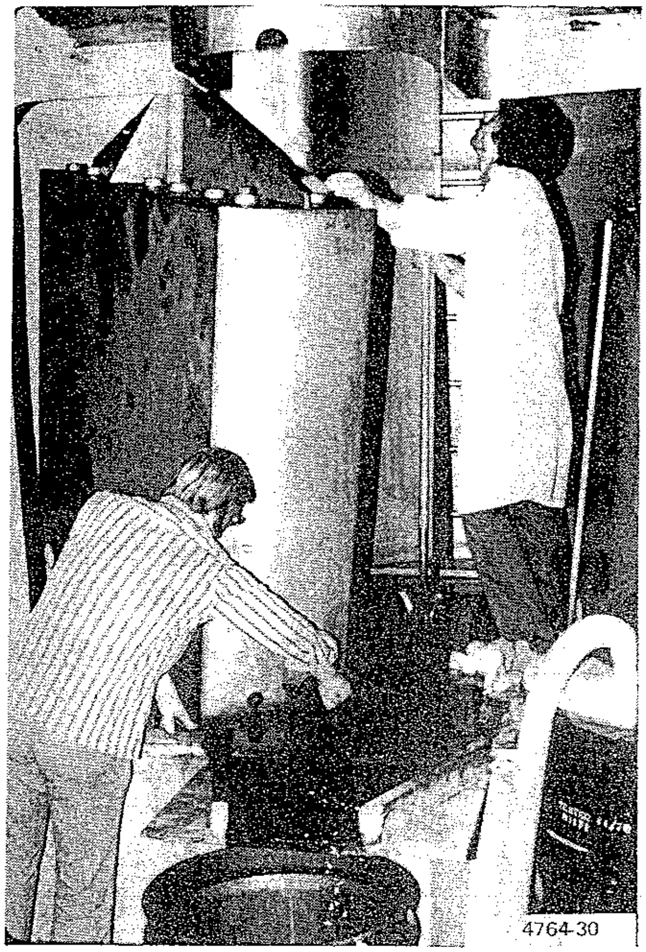


Figure 25. Draining Lead From Lead Doors

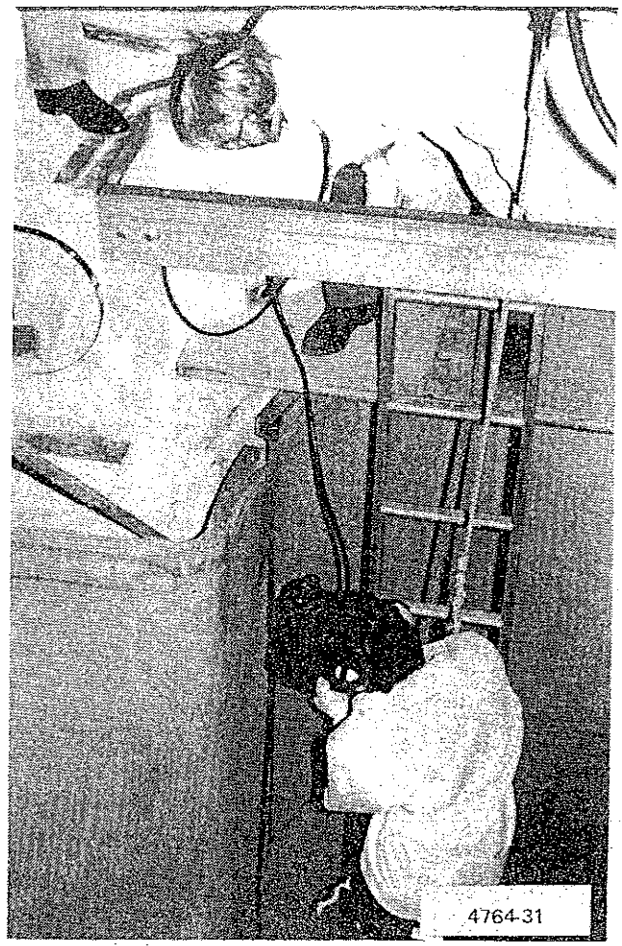


Figure 26. Cutting a Section of Reactor Pool Tank

TABLE 6

ANALYSIS OF LEAD FROM SHIELD DOORS
(Gross detectable beta activity)

Sample Number	pCi/g	Sample Number	pCi/g
Control	0.17	7	0.17
1	0.11	8	LTD
2	0.07	9	0.10
3	0.24	10	LTD
4	0.25	11	0.16
5	0.11	12	0.19
6	LTD*	13	0.10

*Less than detectable limit.

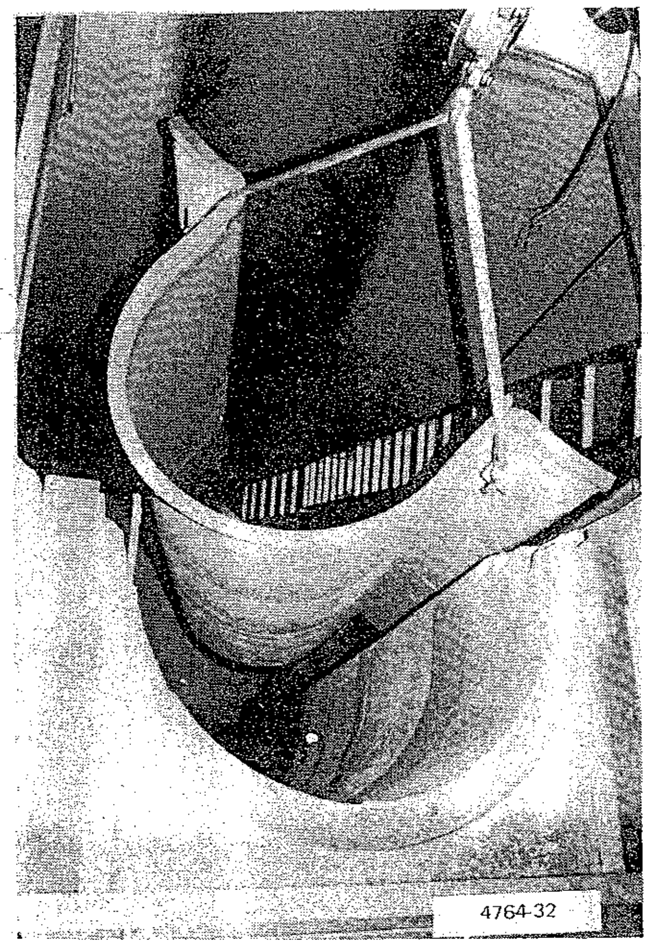


Figure 27. Hoisting Section of Pool Tank From Pool Tank Cavity

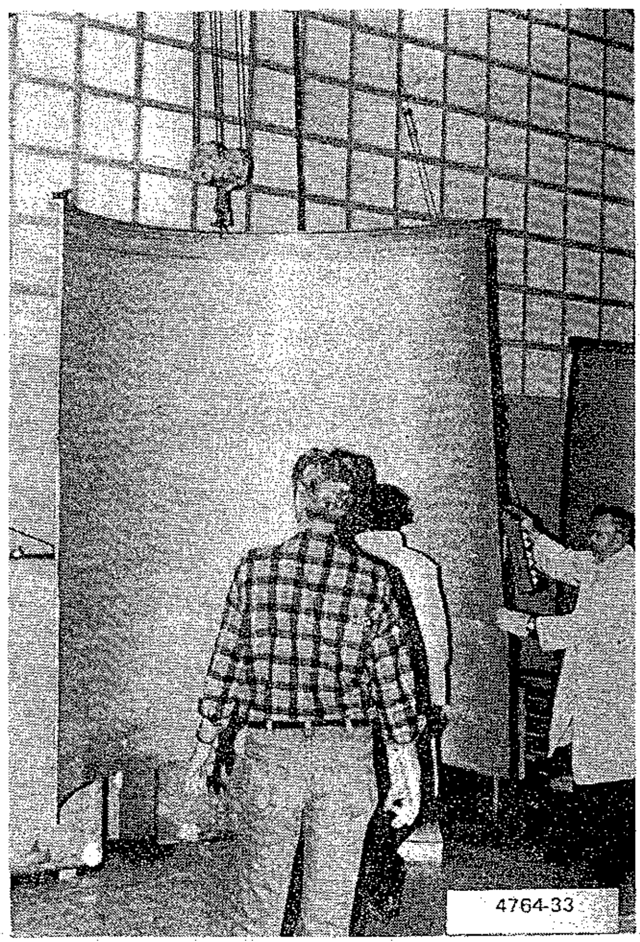


Figure 28. Transferring Section of Pool Tank to Low Background Area for Radiological Survey

was surveyed in a low background area to determine radioactive content. The area of the tank that was exposed to the exposure room and an area 180^{0} from that position and 2 ft to either side of the core centerline was removed and packaged as radioactive waste. The remaining aluminum from the tank was below the limits shown in Tables 1 and 2. This aluminum was disposed of as salvageable scrap. The pool tank had a coating of epoxy-based paint instead of the phenolic-coated tar paper liner described in the RFQ. There was very little adhesion of the tank to the concrete as a result of its being painted instead of coated with tar paper.

5. Concrete Excavation

Following exposure room dismantlement and pool tank removal, a detailed radiation survey was conducted of the exposed concrete structures to establish a map of radioactivity. Concrete samples were cored (Figure 29) from selected areas to establish the extent and levels of activation in the concrete structures. Figure 30 shows the DORF sampling plan identifying the location where core samples were taken. Table 7 shows the results of the core sample analysis and Table 8 is a comparison of the results of analysis from two independent laboratories. The core samples that were provided for comparative analysis were taken from two areas of the exposure room. Sample Nos. 3, 3A, and 3B were taken from the wall and Sample Nos. 34, 34A, and 34B were taken from the floor. Each group of samples were cored as close to each other as possible.

Core samples were prepared for analysis at DORF and at ESG using existing ESG procedures. The samples were cut with a tungsten carbide saw blade at the appropriate distance from the end designated "the surface." The powder generated by sawing was contained, weighed, and counted on an NMC Model 72, automatic counting system for alpha and beta-gamma simultaneously.

Due to preferential cutting through softer material in the core sample, i.e., binder and soft rock as opposed to the harder rock matrix, this sampling technique did not permit obtaining a fully representative sample of the total activity.

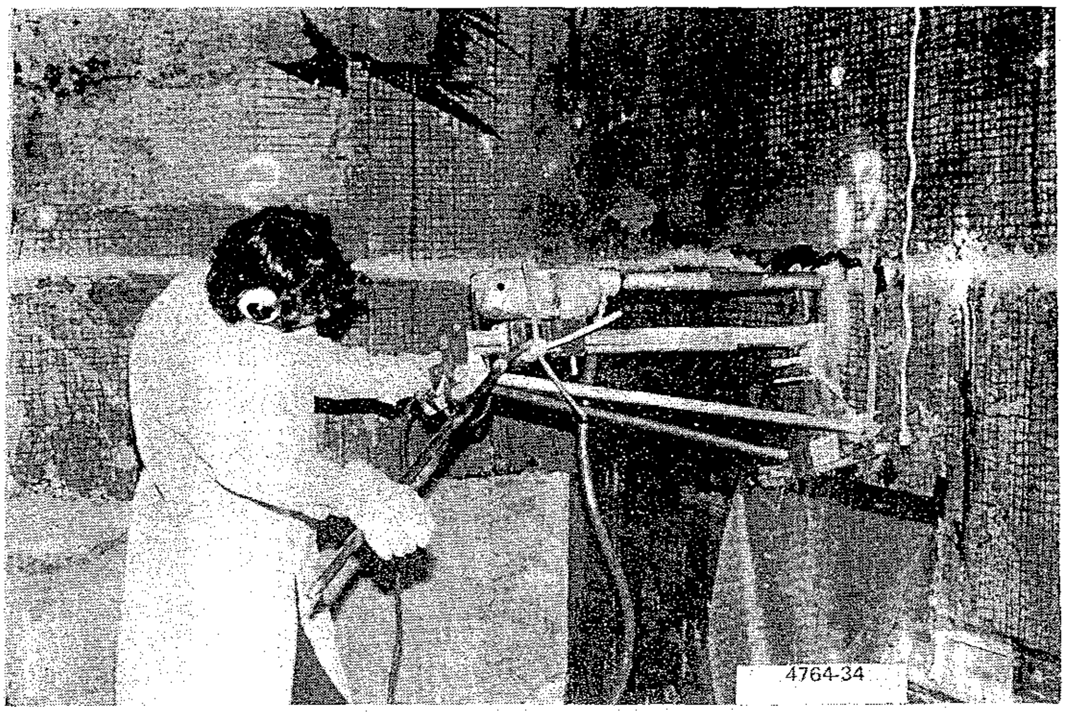


Figure 29. Core Sampling of Concrete in Exposure Room

Teledyne Isotopes prepared their samples by cutting through the entire core sample to segment it into 1-in. thick samples. The entire sample was then counted to determine activity. This technique was most representative of the total activity remaining in the concrete at DORF. The results of the concrete sample analysis formed the basis for the concrete excavation plan. Figures 31 and 32 show diagrams of the planned excavations.

Concrete excavation began in the pool tank cavity with the removal of the pedestal which extended under the tank into the exposure room. Jackhammers were used to break this pedestal (Figure 33) and the thin wall section between the pool tank cavity and the exposure room. Reinforcing bar (rebar) was removed as necessary to permit further concrete removal or because of activation. Activated concrete in the back of the pool tank cavity was then removed. This area, shown in Figure 34, extended about 2 ft to each side of the core centerline and followed the curvature of the wall. Maximum depth of the excavation was 10 in. at core centerline and tapered to about 2 in. at 2 ft from the centerline. Radiological

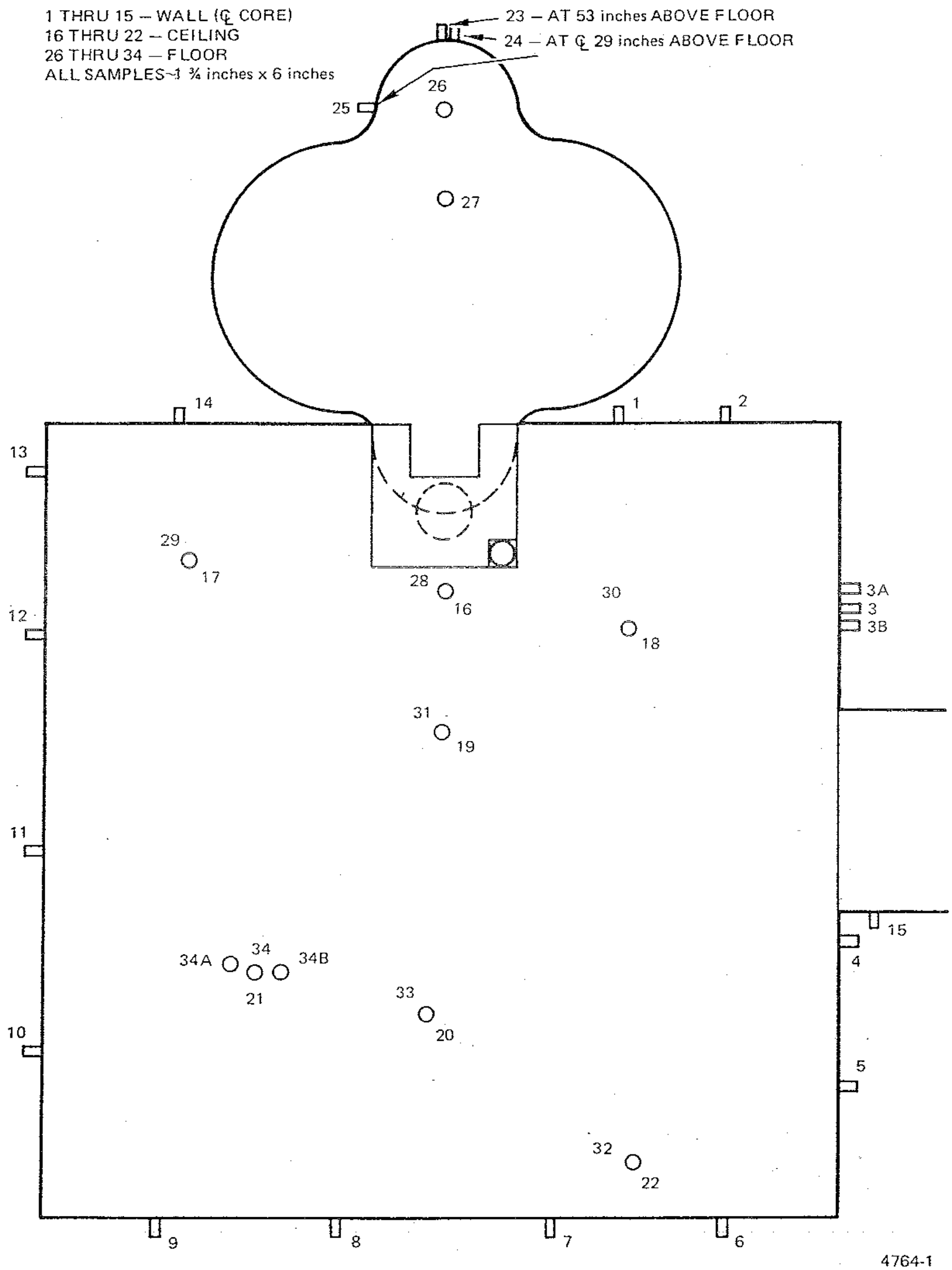


Figure 30. DORF Sampling Plan

TABLE 7

PRE-EXCAVATION ANALYSIS OF CONCRETE BY ESG AT DORF
(Gross Detectable Beta Activity)
(Sheet 1 of 2)

	Core Number	PUG*	pCi/g Distance from Concrete Surface (in.) 0	1	2	3	4	5	6
	Ref. 1	Background		3.2	5.2	9.0	LTD	5.0	LTD
	Ref. 2	Background	12.4	9.7	LTD	LTD	3.1	LTD	LTD
Exposure Room Walls	1	Background	28.0	37.7	23.8	16.6	19.7	10.4	13.3
	2	25 cpm	20.5	23.0	11.0	6.8	11.8	8.5	10.6
	3	300 cpm	126.8	98.0	63.8	65.8	31.4	36.6	35.8
	4	100 cpm	42.1	18.0	22.0	12.0	21.5	18.6	20.1
	5	Background	12.8	11.4	9.1	6.0	10.7	4.6	LTD
	6	∿25 cpm	22.8	8.3	9.5	19.1	5.2	13.1	7.3
	7	100 cpm	34.0	27.1	23.4	22.8	13.1	12.8	10.6
	8	Background	12.0	19.5	20.1	11.6	8.3	11.8	6.2
	9	25 cpm	21.5	10.2	17.0	17.4	14.7	13.7	13.7
	10	50 cpm	18.4	16.2	7.0	7.3	5.6	13.3	3.7
	11	25 cpm	25.3	22.6	19.1	6.4	12.2	5.6	16.0
	12	100 cpm	42.1	33.6	41.0	17.8	14.9	21.1	8.7
	13	50 cpm	50.4	27.6	43.3	34.6	18.4	23.8	22.6
	14	50 cpm	21.8	15.1	3.1	1.5	LTD	5.8	0.2
	15	Background	9.9	7.0	LTD	1.2	4.4	5.4	5.8
Ceiling	16	200 cpm	49.9	396.8	26.3	23.8	14.5	17.0	13.9
	17	50 cpm	43.9	21.8	27.8	20.1	15.3	6.4	7.5
	18	100 cpm	37.7	15.5	35.2	19.5	16.6	17.8	24.2
	19	150 cpm	57.2	11.0	20.7	30.9	16.6	14.7	12.6
	20	Background	17.4	9.5	15.7	1.0	1.7	8.2	9.9
	21	Background	23.8	11.4	7.0	3.5	6.2	9.1	8.1
	22	50 cpm	17.0	11.2	8.1	2.3	8.3	9.5	5.4

TABLE 7

PRE-EXCAVATION ANALYSIS OF CONCRETE BY ESG AT DORF
(Gross Detectable Beta Activity)
(Sheet 2 of 2)

	Core Number	pCi/g							
		PUG*	Distance from Concrete Surface (in.)						
			0	1	2	3	4	5	6
Tank	23	Background	—	11.8	10.3	—	—	—	5.0
	24	400 cpm	59.5	28.2	31.3	29.2	12.6	22.8	19.7
	25	150 cpm	29.8	14.7	25.5	18.4	5.4	9.9	9.3
	26	Background	9.7	15.1	9.5	6.0	1.0	2.9	4.6
	27	Background	3.5	0.6	1.2	LTD	LTD	LTD	2.1
Floor	28	100 cpm	43.9	32.9	24.0	26.7	38.3	13.5	18.0
	29	100 cpm	20.9	17.6	7.9	11.6	2.3	5.6	12.2
	30	50 cpm	18.6	13.3	14.7	14.3	2.5	10.8	10.4
	31	50 cpm	19.9	12.6	5.6	9.9	8.3	12.2	LTD
	32	Background	7.3	7.7	8.3	3.3	0.6	5.6	11.8
	33	25 cpm	19.3	2.3	1.7	7.5	3.7	5.2	LTD
	34	Background	15.3	6.0	10.2	11.2	LTD	8.3	5.0

*Count rate meter with a 2-in. thin window pancake G-M detector.

survey of the pool tank cavity indicated compliance with Regulatory Guide 1.86, stipulations of which are listed in Tables 1 and 2.

Nuclear Controls Corporation (NCC) was contracted to break the activated concrete from the rolling door and to remove the remainder of the door from the site. This operation, shown in Figures 35, 36, and 37, was supported by the Rockwell staff and took place between January 28 and February 4, 1980. NCC used a rock splitter, a jackhammer, and a mobile hydraulic ram (hyram) to break up the door and remove it from the facility. The clean rubble from the door was staged on site for removal during Phase III. The activated rubble was packaged by Rockwell for disposal as radioactive waste.

TABLE 8

PRE-EXCAVATION ANALYSIS OF CONCRETE BY TELEDYNE ISOTOPES

Depth	pCi/g						
	Surface 0-1	1-2	2-3	3-4	4-5	5-6	6-7
3A							
K^{40}	12.7	10.4	5.4	—	12.3	—	—
Co^{60}	154.0	136.0	86.6	443.0	55.3	27.5	20.2
Eu^{152}	281.0	188.0	141	96.6	166.0	93.0	31.2
Eu^{154}	19.0	13.6	5.29	7.3	12.2	7.0	2.3
	466.7	348.0	238.3	546.9	245.8	127.5	53.7
34A							
K^{40}	—	3.8	—	4.0	—	1.9	1.9
Co^{60}	18.2	15.8	6.8	7.8	5.0	4.5	2.2
Eu^{152}	34 6	36.9	14.2	18.1	14.6	9.7	3.2
Eu^{154}	2.9	2.5	0.9	1.2	<0.6	0.9	<0.3
	55.7	59.0	21.9	31.1	20.2	17.0	7.6
	ESG Data*						
	0 in.	1 in.	2 in.	3 in.	4 in.	5 in.	6 in.
3B	74.5	70.2	41.3	26.2	24.4	19.0	17.8
34B	12.9	6.7	6.7	11.3	6.3	9.0	4.5

*Gross βγ pCi/g

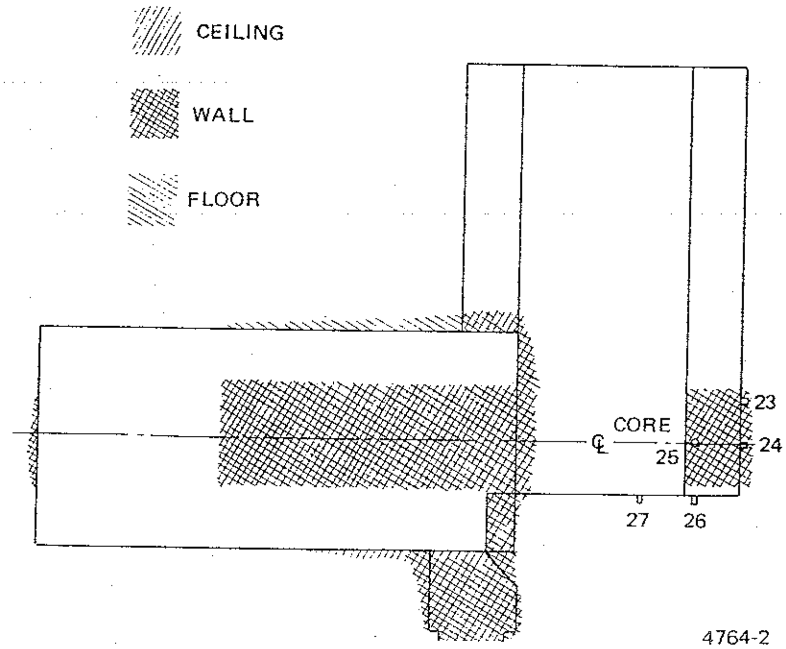


Figure 31. Planned Excavation
(Side View)

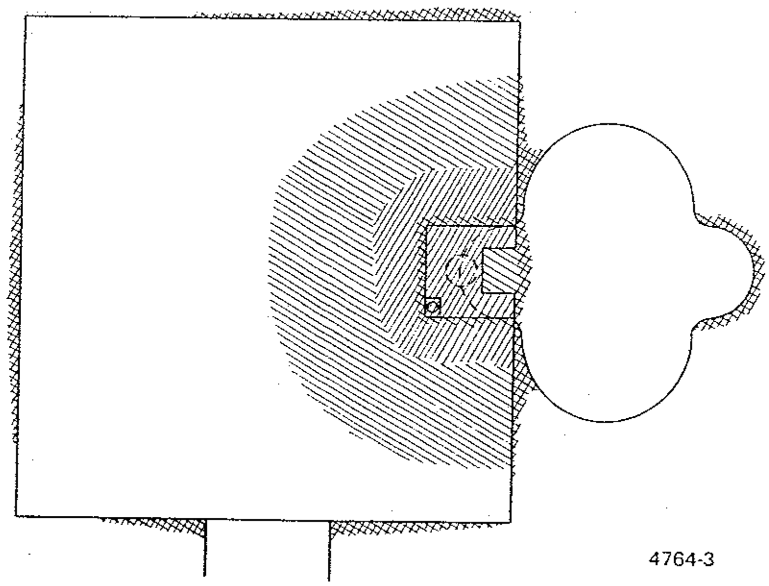


Figure 32. Planned Excavation
(Top View)

Figure 33. Excavating Concrete From Area Below Primary Reactor Operating Position

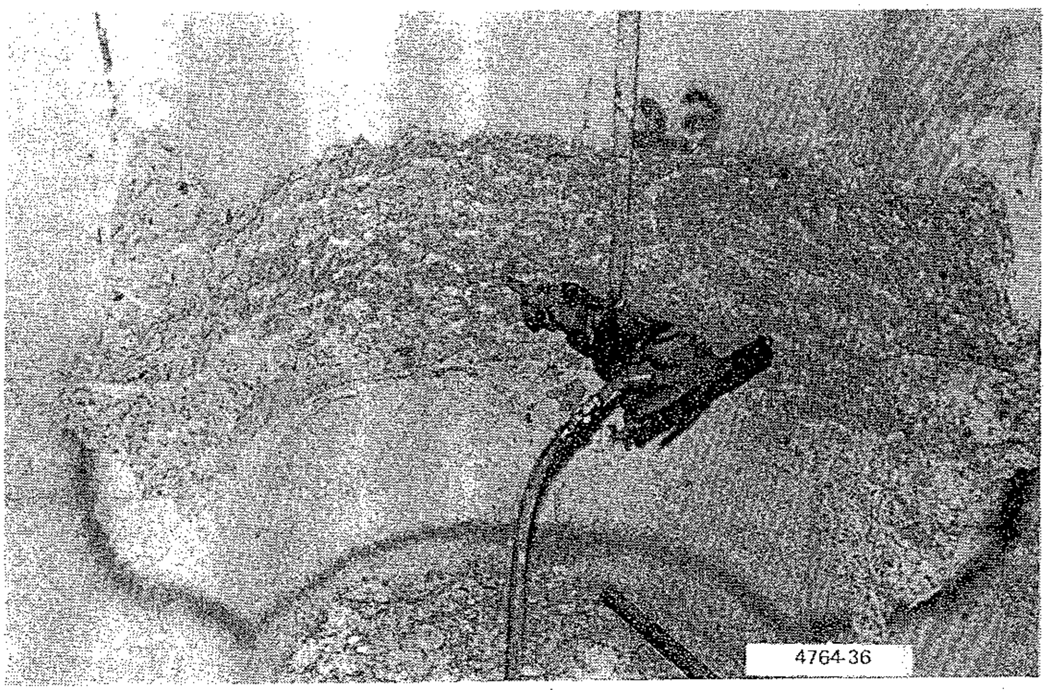


Figure 34. Excavating Concrete From Wall Surrounding Secondary Reactor Operating Position

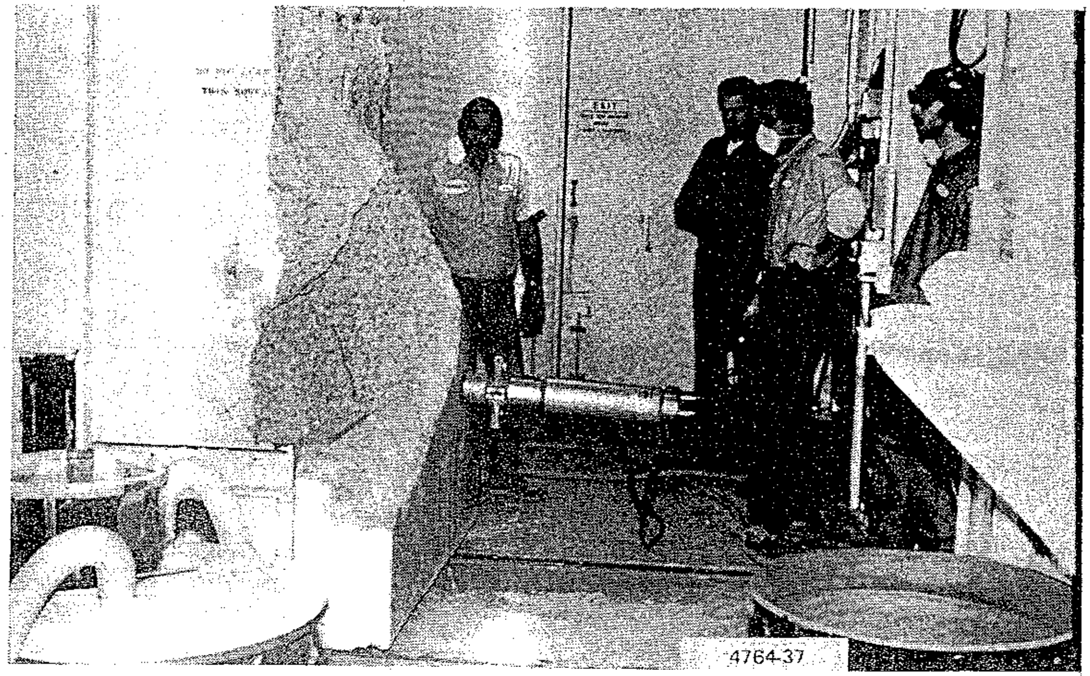


Figure 35. Using a Rock Breaker to Form Cracks in Activated Portion of Rolling Door

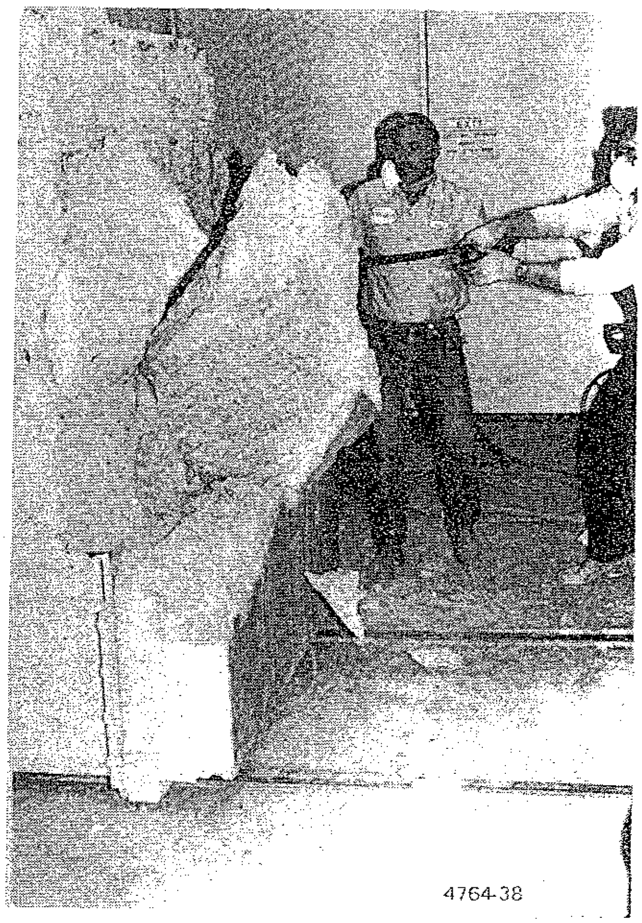


Figure 36. Using a Pry Bar to Remove Fractured Concrete

The first effort in the exposure room was the removal of some of the phenoline liner (tar paper) from the concrete walls to determine its effect on background radiation in the room. A surface area of about 150 ft^2 of the north and east walls was removed by scabbling with bushing tools in 15-lb chipping hammers. This activity is shown in Figure 38. Radiation measurements with a PUG 1 before and after scabbling indicated no difference in reading even to where 1/4 in. of concrete was removed. Based on these results, removal of the phenoline liner from the concrete was terminated.

Based on the core sample analysis and contact radiation readings, there were five areas of the exposure room significantly above background. These areas are identified in Figures 31 and 32. Concrete was removed from these areas to depths of 6 to 8 in. using a combination of hydraulic crack forming and impact. A commercial rock splitter was used to hydraulically compress the concrete to form cracks, a jackhammer or chipping hammer was then used to break the concrete from the walls and ceiling. Figures 39 through 42 show this activity. About 40,000 lb of concrete was removed from the exposure room, packaged in DOT-approved shipping containers, and disposed of as radioactive waste.

6. Site Survey

Concurrent with and following the removal of radioactive components and materials from the DORF site, radiological surveys were conducted to document the levels of radioactivity left in the facility. Table 9 presents data generated from analyzing concrete for fixed contamination. No fixed or removable contamination or activation was detectable outside of the exposure room above Tables 1 and 2 limits. Concrete activation in the exposure room was greater than the Table 2 limits but less than the limits specified in the NRC Regulatory Guide 1.86. Figures 43 through 45 show the detectable activity remaining in the exposure room as determined during the final site survey. Continuous air monitors were operated in the reactor building during the demolition. An NMC Model was located in the high bay and an Eberline Model was located in the exposure room. Table 10 shows the air sample results for those areas.

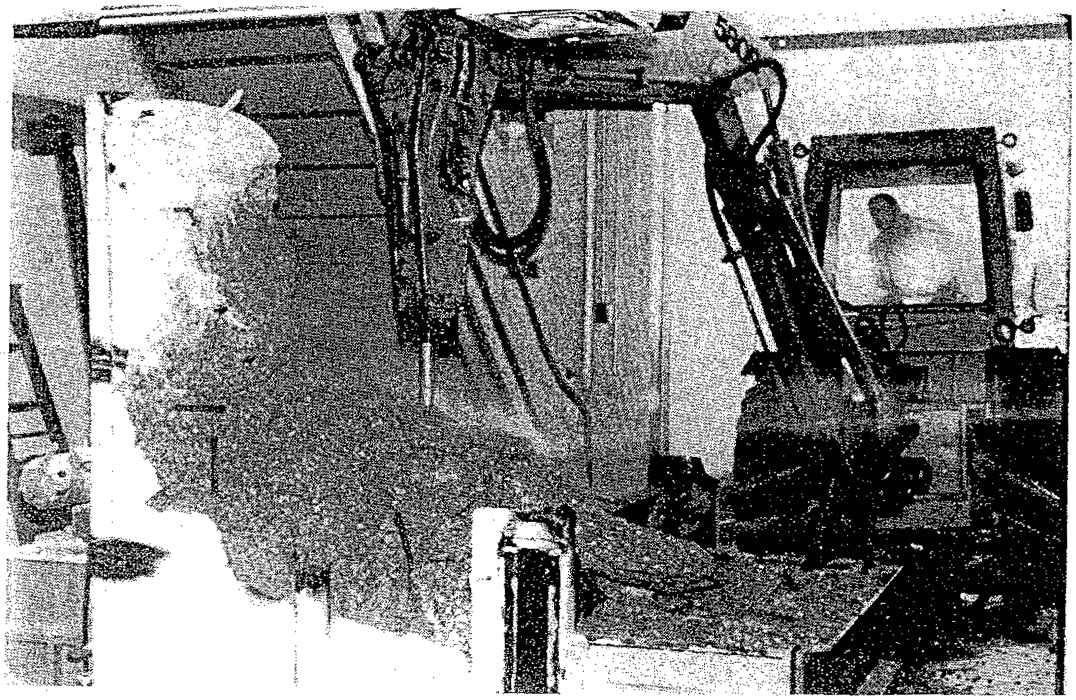


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Figure 37. Mobile Hydraulic Ram Being Used to Break Nonradioactive Portion of Rolling Door

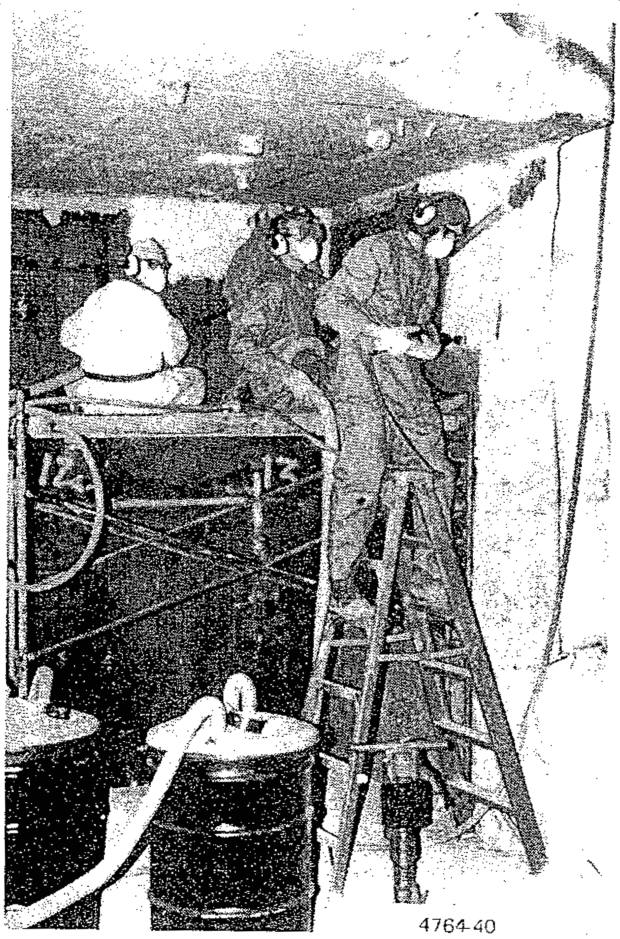

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Figure 38. Scabbling Concrete from Surface of Exposure Room Wall

Figure 39. Using Rock Drill to Bore Holes Into Ceiling of Exposure Room (preparation for inserting rock splitter)

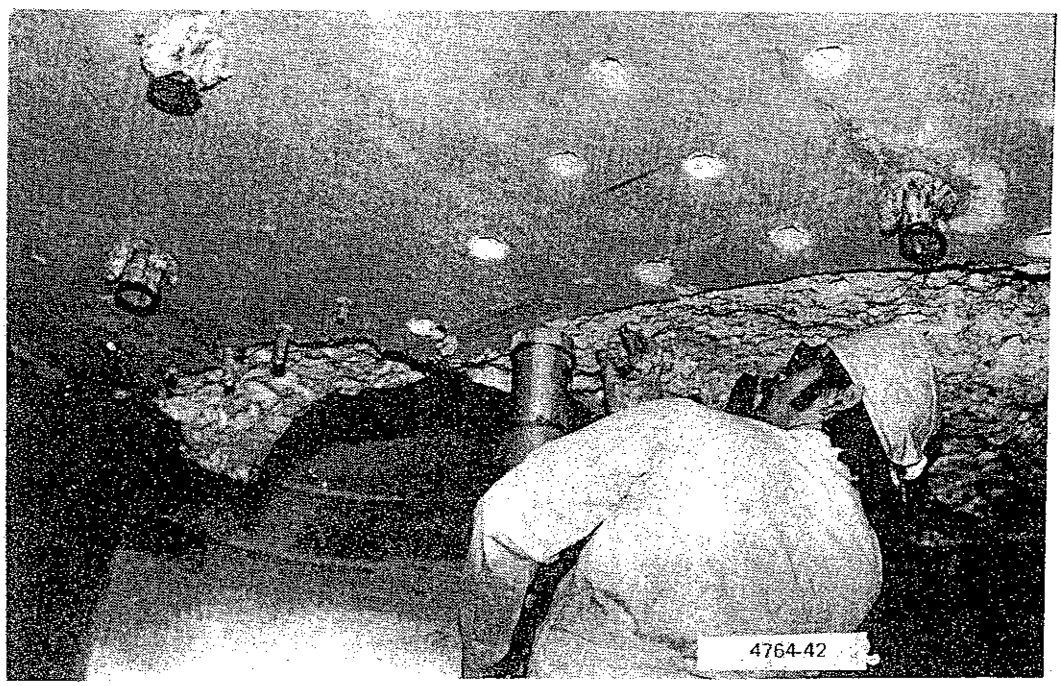


Figure 40. Using Rock Splitter to Form Cracks in Concrete Ceiling

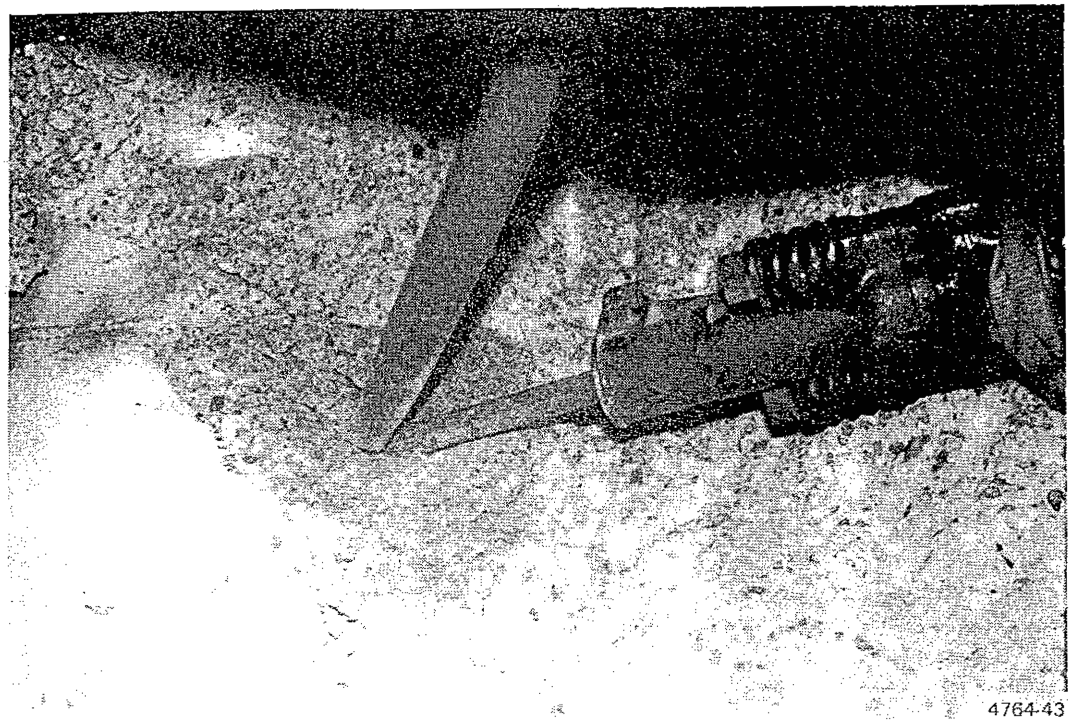


Figure 41. Spalling Concrete From Surface of Exposure Room After Crack Forming

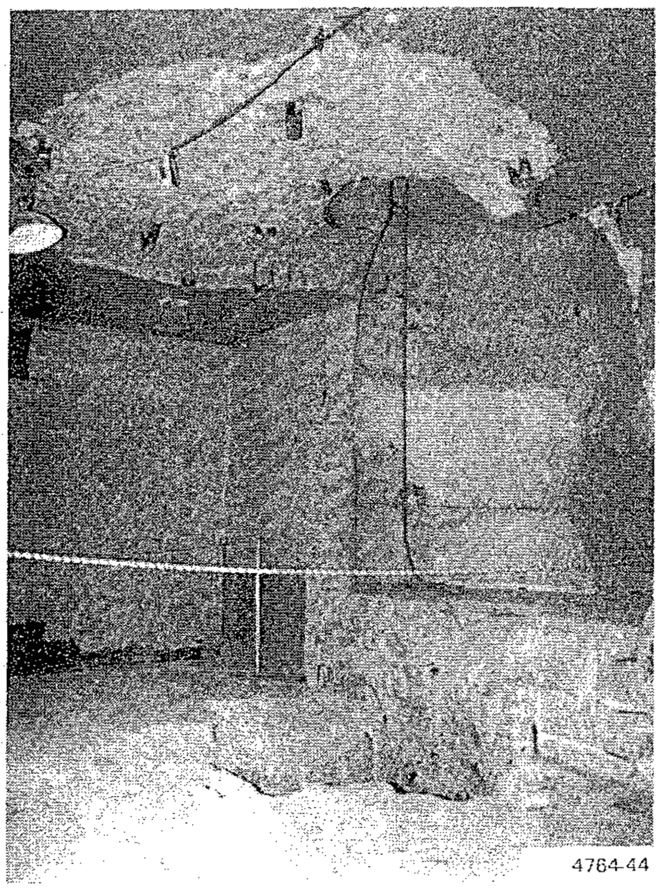


Figure 42. View of Concrete Excavation of Ceiling, Floor, East and North Walls of Exposure Room and Portion of Pool Tank Cavity

TABLE 9

POST-EXCAVATION ANALYSIS FOR FIXED CONTAMINATION IN CONCRETE

Sample No.	Location	PCi/g Dry			
		Co^{60}	Eu^{152}	Eu^{154}	Total
P-1	Excavated South Wall	27	--	4	31
P-2	Nonexcavated South Wall	21	54	4	79
P-3	Excavated Pit Wall	14	37	2	53
P-4	Excavated Pit Wall	19	43	2	67
P-5	Nonexcavated Floor	32	68	4	104
P-6	Nonexcavated South Wall	16	33	—	49
P-7	Excavated South Wall	13	28	2	43
P-8	Nonexcavated West Wall	28	61	4	93
P-9	Excavated West Wall (Plug)	29	37	3	69
P-10	Nonexcavated North Wall	134	28	—	162
P-11	Excavated North Wall	18	36	2	56
P-12	Scabbled East Wall	15	24	—	39
P-13	Nonexcavated South Reactor Wall	1	6	—	7
P-14	Excavated South Reactor Wall	5	30	2	37
P-15	Excavated Ceiling	15	68	5	88
P-16	Nonexcavated Ceiling	34	82	6	122
P-17	Nonexcavated Ceiling	19	56	4	79

Note: Five samples contained approximately 10 pCi/g K^{40}, two samples contained approximately 15 pCi/g Co^{57}, and one sample contained 96 pCi/g Cs^{134} (P-17).

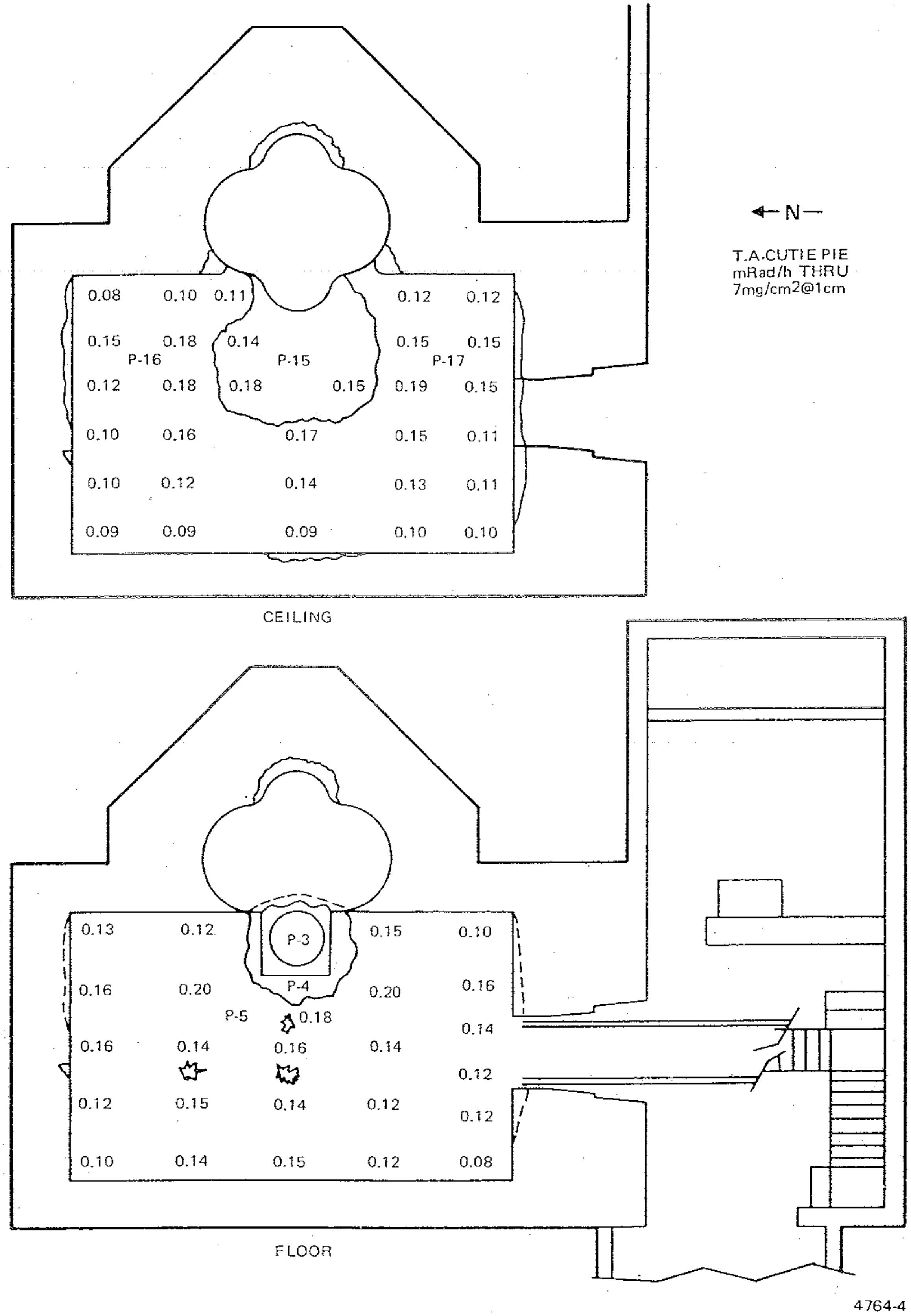


Figure 43. DORF Final Survey, Exposure Room

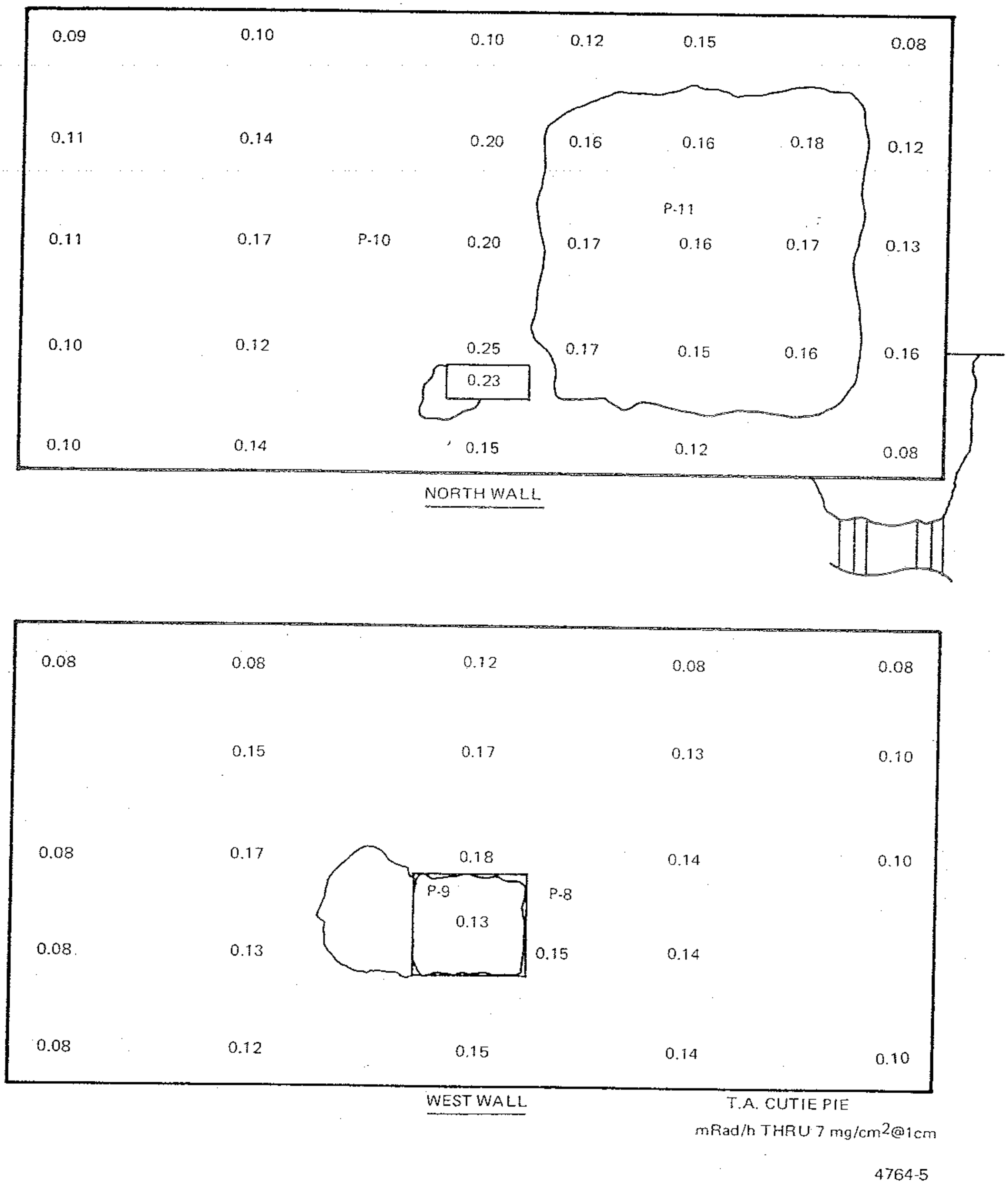


Figure 44. DORF Final Survey, Exposure Room

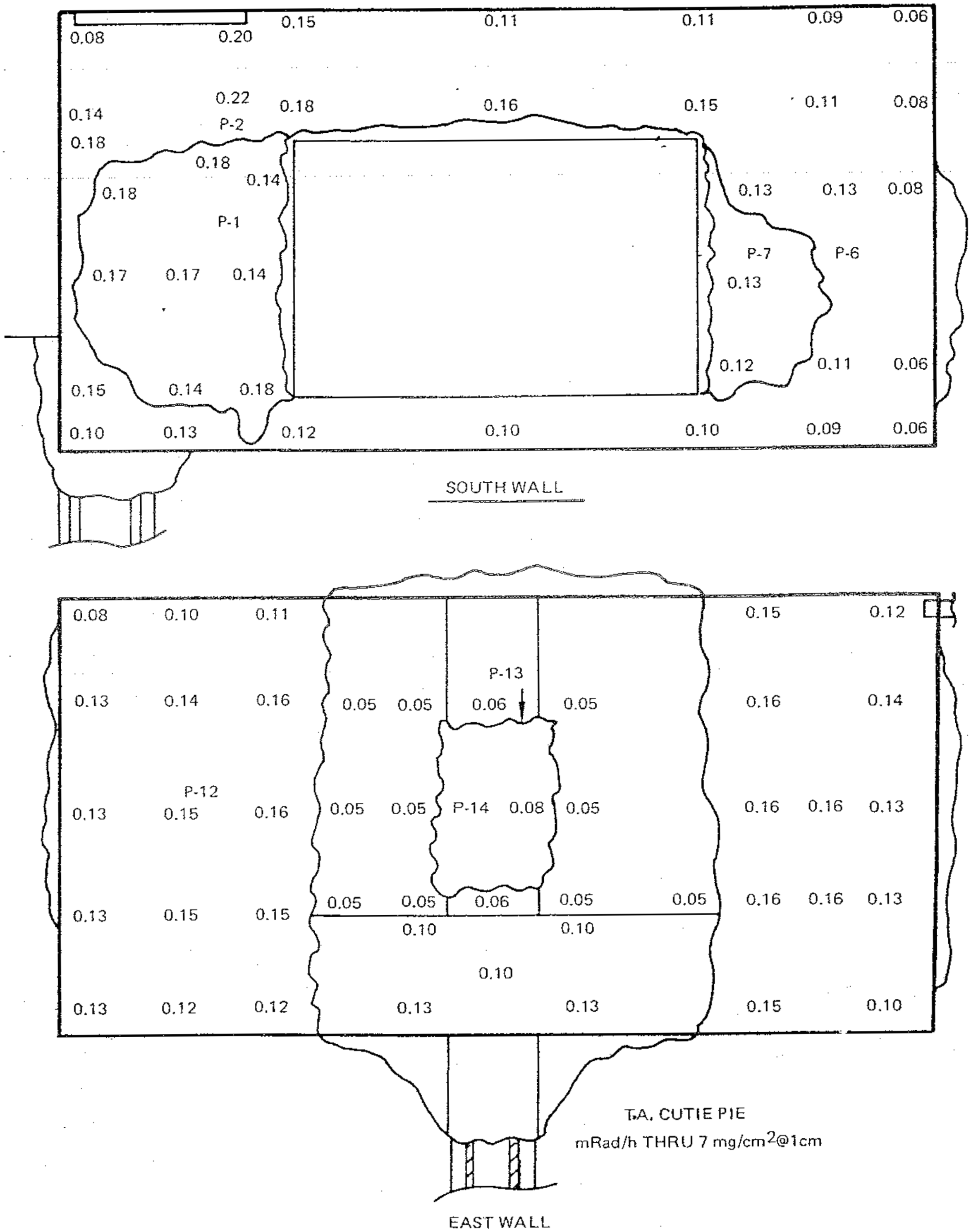


Figure 45. DORF Final Survey, Exposure Room

TABLE 10

RESULTS OF AIR SAMPLING AT DORF

	μCi/cc	
	High Bay	Exposure Room
<u>1979</u>		
27 Nov. to 3 Dec.	1.6×10^{-14}	1.8×10^{-13}
3 Dec. to 10 Dec.	1.3×10^{-14}	1.1×10^{-13}
10 Dec. to 17 Dec.	1.3×10^{-14}	1.9×10^{-13}
17 Dec. to 20 Dec.	1.7×10^{-14}	9.1×10^{-13}
<u>1980</u>		
8 Jan. to 14 Jan.	1.7×10^{-14}	9.0×10^{-14}
14 Jan. to 18 Jan.	1.2×10^{-14}	4.6×10^{-14}
18 Jan. to 22 Jan.	1.3×10^{-14}	1.2×10^{-13}
22 Jan. to 25 Jan.	1.7×10^{-14}	6.1×10^{-13}
25 Jan. to 28 Jan.	2.0×10^{-14}	4.6×10^{-13}
28 Jan. to 31 Jan.	1.7×10^{-14}	2.2×10^{-13}
31 Jan. to 7 Feb.	0.7×10^{-14}	7.3×10^{-14}
7 Feb. to 12 Feb.	1.2×10^{-14}	4.3×10^{-14}
12 Feb. to 16 Feb	3.6×10^{-14}	0.7×10^{-14}

Note: MPC for Eu^{154} in air is 4×10^{-9} μCi/cc. This was the most restrictive isotope present.

7. Waste Disposal

Radioactive (RA) waste was packaged into DOT-approved containers as the waste was generated. The types of material removed from the DORF site as RA waste included concrete, wood, aluminum, steel, plastic, and rubber. Forty-seven steel drums and eight wooden boxes containing 1143.5 ft^3, weighing 60,425 lb, and 1.17×10^{-4} Ci were removed from DORF. The waste was disposed of by land burial. Due to State of South Carolina restrictions on the volume of RA waste permitted for burial at the Barnwell site, the earliest space allocation available to DORF waste was in April. In order to complete the DORF contract on schedule, the waste was shipped to the Nuclear Engineering Company (NECO) site in Beatty,

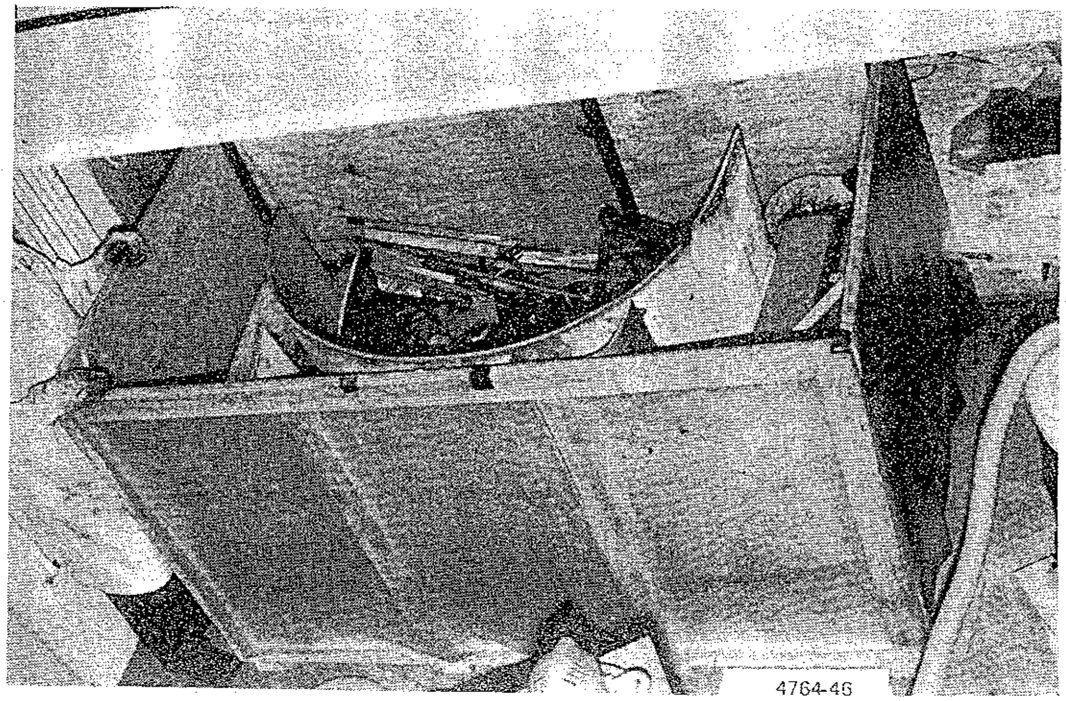


Figure 46. Partially Filled Box of Radioactive Waste

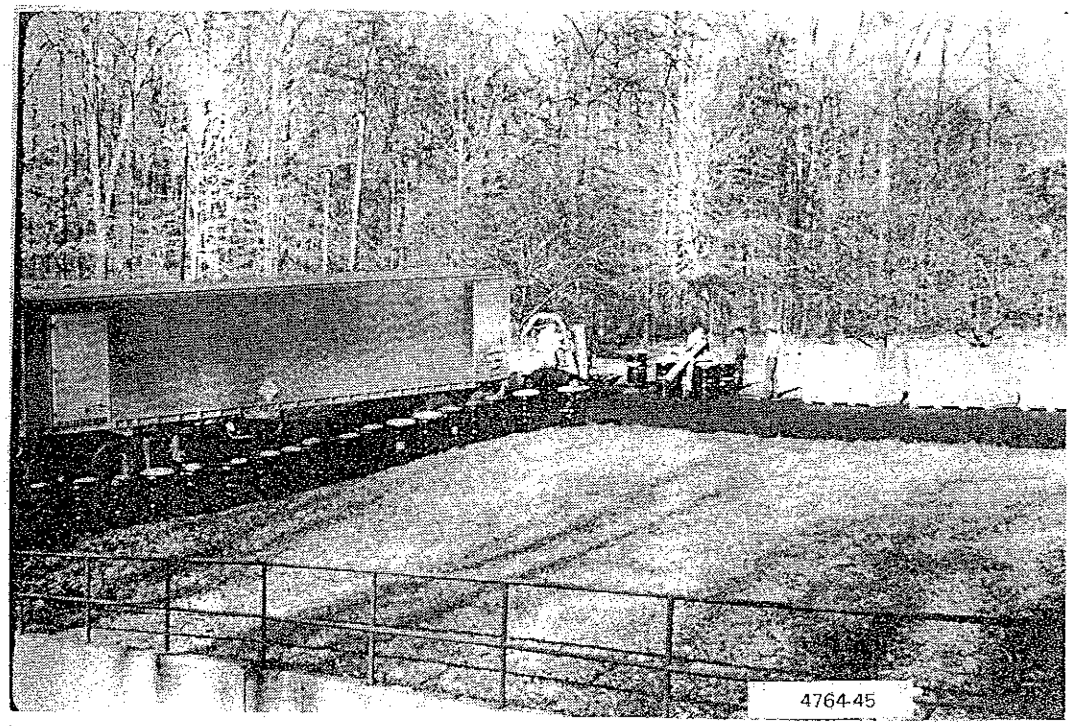


Figure 47. Radioactive Waste Containers Being Loaded Onto Truck for Disposal by Land Burial

Nevada. Chem-Nuclear Systems was contacted to act as broker for the disposal of the RA waste. In that capacity, they handled the arrangements for the transportation and disposal of the waste taking possession of it at the DORF site boundary. Figures 46 and 47 show waste being loaded for shipment to land burial.

C. CONFIRMATORY SURVEY

A survey of the DORF site was conducted between February 25, 1980 and February 27, 1980 by a U.S. Army Environmental Health Agency radiation survey team. This survey was conducted to confirm compliance with NRC Regulatory Guide 1.86 prior to the Army's acceptance of the facility for unrestricted use. The survey team's recommendation, following analysis of the data from the onsite survey, was to accept the facility for unrestricted use. A copy of this recommendation and the preliminary results of their survey are appended in Appendix A. Compliance with NRC Regulatory Guide 1.86 was a contracted prerequisite to conducting Phase III tasks.

D. PHASE III

Phase III consisted of dismantling the concrete parapet to the main floor level, the restoration of any disrupted services to the building, and the repair of facilities damaged by the dismantling activities. An option was given to haul all of the debris from the site or to put it in the pool tank cavity, provided a barrier was erected between the exposure room and the cavity.

Phase III began on April 21, 1980 when ESG was officially notified that DORF was in compliance with NRC Regulatory Guide 1.86, and was completed on May 9, 1980.

Nuclear Controls Corporation (NCC), a subsidiary of the Penhall Company, was contracted by ESG to perform all of the Phase III work. They elected to construct a barrier between the exposure room and the pool tank cavity to permit the debris from the parapet to be disposed in the cavity. The barrier was constructed with a single course of the No. 4 reinforcing bar in the wall and a double course in the roof extension to the exposure room ceiling. The barrier wall was formed by

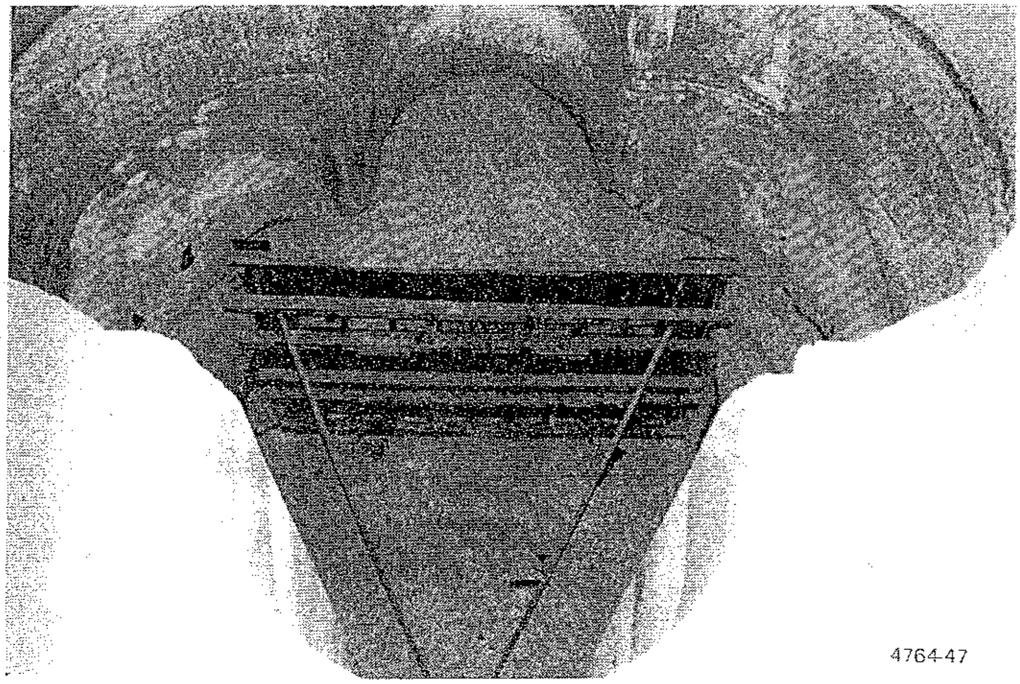


Figure 48. View of Barrier Wall From Parapet

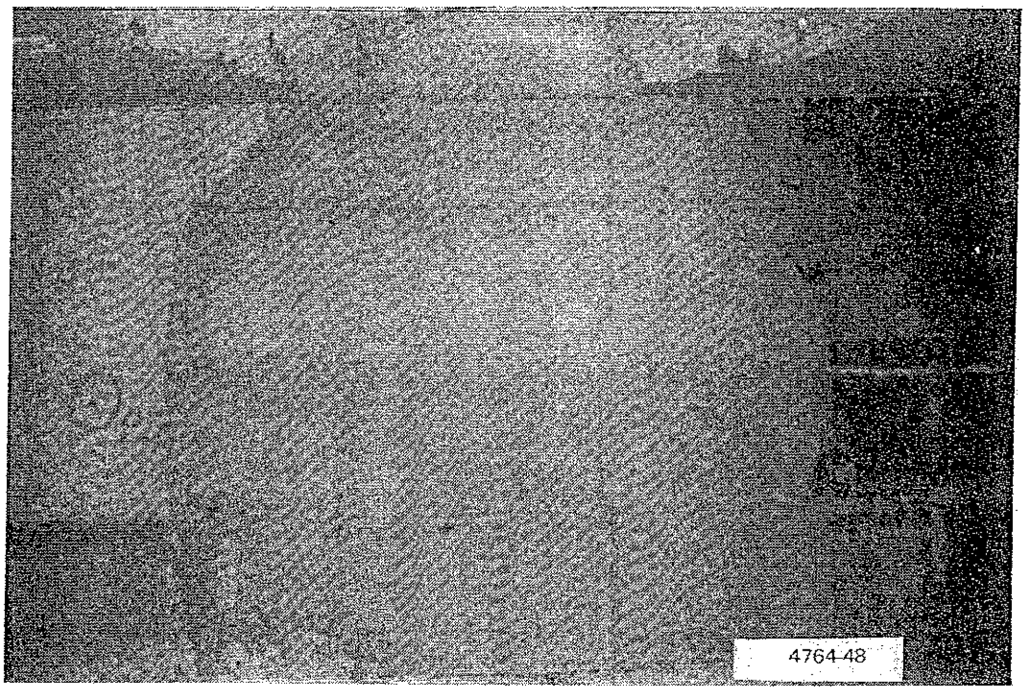


Figure 49. View of Barrier Wall From Exposure Room

a single pour of concrete to a minimum thickness of 8 in. Figures 48 and 49 show the barrier.

NCC contracted Controlled Demolition Incorporated (CDI) to dismantle the concrete parapet. CDI specializes in explosive demolition. The parapet was reduced to rubble and placed into the pool tank cavity by the combined use of explosive charges, jackhammers, and a "Bobcat" skiploader. Figures 50, 51, and 52 show the parapet during dismantlement. Figure 53 shows the parapet demolition completed with the rubble completely below the floor level. A railing was installed around the open pit to provide a safety barrier. Restoration of the floor was not included in this contract.

Rubble, remaining on site from the demolition of the rolling door during Phase II, was removed from the site and, that area where the rubble had been piled was leveled and reseeded.

The building was cleaned of the debris generated by the demolition work. Minor repairs were made to restore the facility's lighting system and other utilities were verified to be functioning satisfactorily.

DORF facility drawings were redlined to reflect the changes made by the performance of the contracted work. These drawings were presented to the DORF Contracting Officer's Representative. A list of the drawings and a description of the changes are given in Table 11.

Figure 50. Explosive Charges Being Loaded Into
Pre-drilled Holes

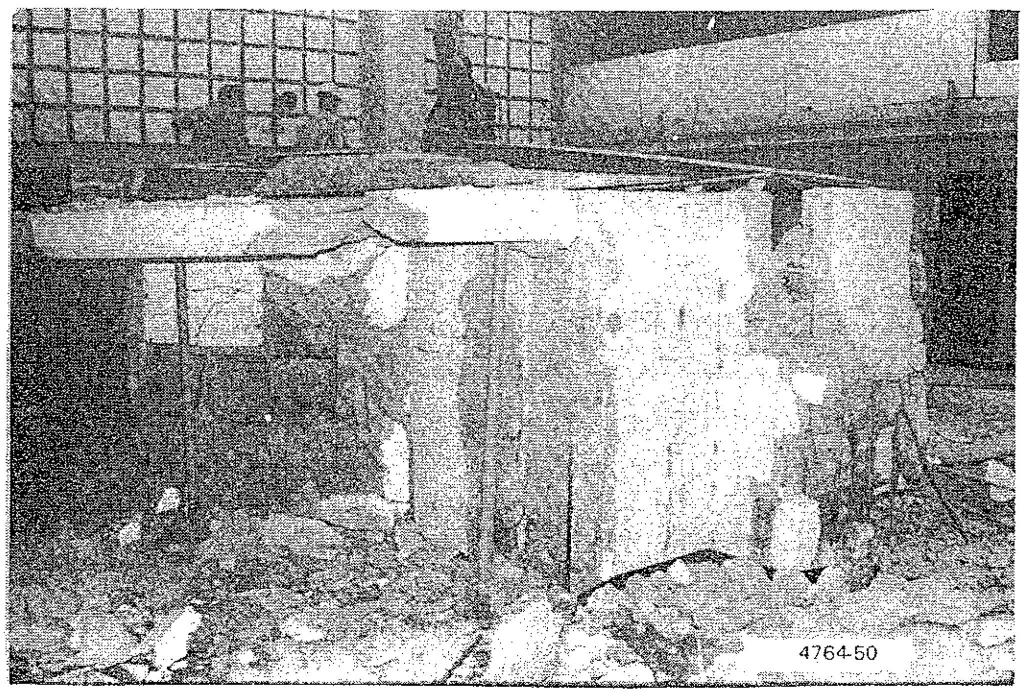


Figure 51. Parapet After Several Explosive Charges Set Off

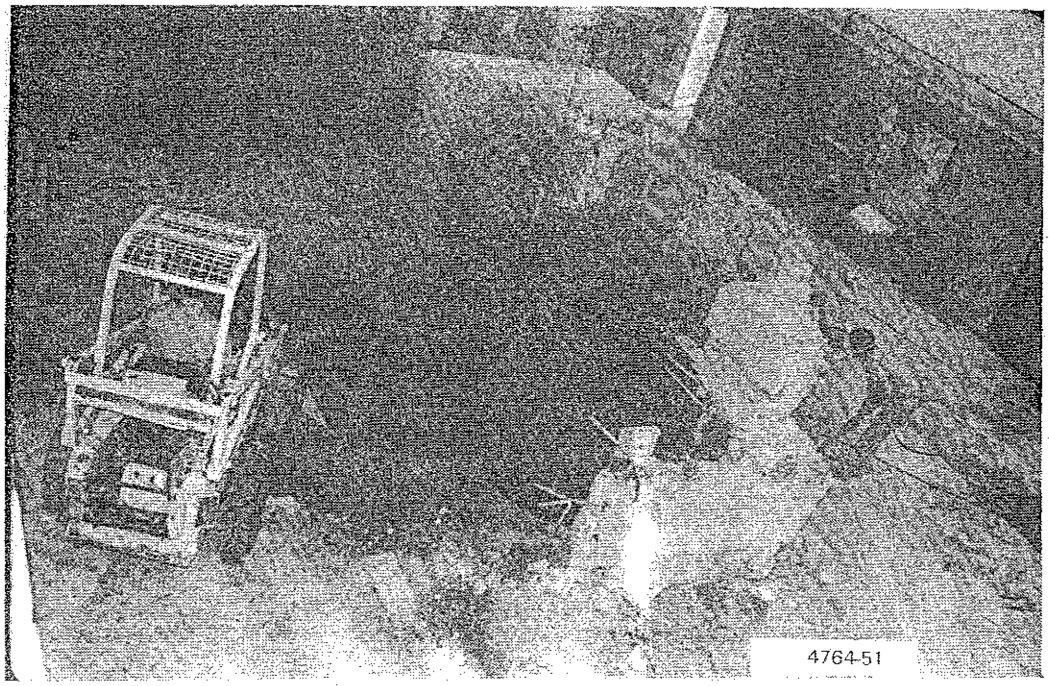


Figure 52. Concrete Rubble Being Pushed Into Pool Tank Cavity

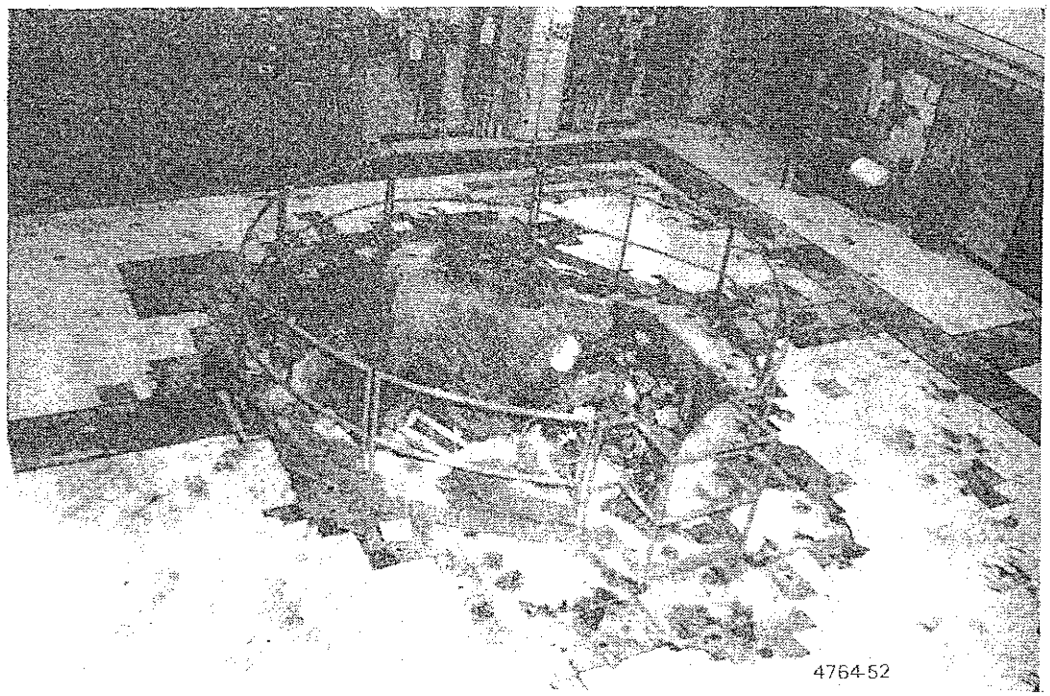


Figure 53. Parapet Demolition Completed

TABLE 11

REDLINE CHANGES TO DORF DRAWINGS
(Sheet 1 of 2)

DRAWING NUMBER

A-1 FLOOR PLANS AND SCHEDULES

Shows parapet removed.

Shows lead shielding removed.

Reference to Drawings #S-1, 2, 3, and 4 for changes.

A-2 ELEVATIONS AND SECTIONS

Crosshatched Section A to show deletion of parapet and Exposure Room wood.

Also crosshatched elevation B-1 to show deletion of details for Exposure Room.

A-3 SECTIONS AND DETAILS

Crosshatched Details 5, 10, 11, and 12 to show deletion of wood in Exposure Room.

E-1 SINGLE LINE DIAGRAM

Circled portion of line diagram showing the recirculating water pumps, lead shield door, rolling door, and core dolly electrical systems disconnected.

E-2 GROUNDING PLANS AND SYMBOLS

Circled parapet and Exposure Room grounding references to show them removed.

E-4 POWER

Marked up print to reflect disconnected circuits.

E-6 SAFETY INTERLOCK DIAGRAMS

Noted that fuses were removed from the relay panel in order to disable those circuits.

M-1 PLANS AIR CONDITIONING

Noted that wood was removed from the Exposure Room.

TABLE 11

REDLINE CHANGES TO DORF DRAWINGS
(Sheet 2 of 2)

DRAWING NUMBER (Continued)

M-2 EQUIPMENT ROOM DETAILS

Showed that reactor cooling equipment was removed and that the hydraulic system for the lead shield hoist was drained and the hydraulic cylinder removed.

M-3 PLUMBING

Crosshatched details 5, 6, and 8, and Section B to show deletion.

M-5 ROLLING DOOR - EXPOSURE ROOM

Noted that the door was demolished.

S-1 PLANS

Crosshatched Detail 4 to show deletion.

S-2 SECTIONS

Crosshatched Detail 1, Section C and Section D to show deletion. Also crosshatched to show approximate areas affected by the excavation of concrete.

S-3 DETAILS

Crosshatched rolling door frame to show deletion. Also circled and crosshatched area of doorway affected by concrete removal.

S-4 RAMP PLANS AND DETAILS

Crosshatched ramp plan, ramp reinforcing plan, Section A, and Details 1 and 2 to show deletion.

APPENDIX A

DEPARTMENT OF THE ARMY HARRY DIAMOND LABORATORIES

DEPARTMENT OF THE ARMY
HARRY DIAMOND LABORATORIES
2800 POWDER MILL ROAD
ADELPHI, MD. 20783

DELHD-N-RBI

22 April 1980

RECEIVED
APR 25 1980
Correspondence Dept.

Rockwell International Corp.
Energy Systems Group
8900 De Soto Avenue
Canoga Park, CA 91304

Reference: Contract DAAK 21-79-C-0136

The inclosed letter of 17 April 1980 from the U.S. Army Material Development and Readiness Command, D. Taras states that the NRC Regulatory Guide 1.86 criteria for unrestricted use has been achieved at the Diamond Ordance Radiation Facility (DORF) based upon the preliminary report from the U.S. Army Environmental Hygiene Agency, 3 April 1980. Accordingly, this letter constitutes official notification that Section F.5.1 of the above contract has been accomplished and you now have thirty days from the date of this letter to initiate the phase III tasks (i.e. F.5.2 - F.10 as amended) as indicated in Section H.3 of the contract.

Sincerely,

Charles Ware

Charles Ware
Contracting Officers
Representative

incl. - 2
CF
J. Rosado 22000
D. Schallhorn 22900
A. Mazzone 091
J. Harris Rockwell Int.

DEPARTMENT OF THE ARMY
HEADQUARTERS US ARMY MATERIEL DEVELOPMENT AND READINESS COMMAND
5001 EISENHOWER AVENUE, ALEXANDRIA, VA. 22333

D. Taras/seb/AUTOVON 284-9340

DRCSF-P

17 April 1980

SUBJECT: Decontamination of Diamond Ordnance Radiation Facility

THRU: Commander
US Army Electronics ~~Research~~ and Development Command
ATTN: ~~DRDEL-SS~~
Adelphi, MD 20783

TO: Commander
Harry Diamond Laboratories
ATTN: DELHD-N-RBI
Adelphi, MD 20783

1. Reference is made to the following report: Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980.

2. On 10 April the Army Reactor Committee for Health and Safety reviewed the referenced report and concluded that decontamination is consistent with the criteria in NRC Regulatory Guide 1.86 and is as low as reasonably achievable. In PHONECON, 17 April 80, LTC Quillin, WRAMC Radiation Protection Officer, stated these acheived levels are acceptable to WRAMC. Based on the above, the facility is suitable for unrestricted use and occupancy.

DARWIN N. TARAS
Member, Army Reactor Committee
for Health and Safety

CF:
HQDA(DASG-PSP-E); (DAPE-HRS)
DRCIS
DRCSG

DEPARTMENT OF THE ARMY
U. S. ARMY ENVIRONMENTAL HYGIENE AGENCY
ABERDEEN PROVING GROUND, MARYLAND 21010

Mr. Lodde/cw/AUTOVON
584-3526

HSE-RH/WP

3 APR 1980

SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Commander
US Army Materiel Development and
Readiness Command
ATTN: DRCSG
5001 Eisenhower Avenue
Alexandria, VA 22333

1. AUTHORITY. Letter, DELHD-N-RBI, Harry Diamond Laboratories, 2 November 1979, subject: Request for a Radiological Health Special Study, and indorsement thereto.

2. PURPOSE. This special study was performed to determine the presence and extent of radioactive contamination and whether the facility met the radioactive contamination levels stated in Nuclear Regulatory Commission, Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974, following decontamination.

3. GENERAL.

a. This radiation protection special study was conducted by Mr. Gordon M. Lodde, Health Physicist, and 2LT Roger M. Davis, Jr., Health Physics Division, this Agency, during the period 25-28 February 1980.

b. An entrance interview and an exit briefing were provided to Mr. Charles Ware, Contracting Officer's Representative, Harry Diamond Laboratories.

4. FINDING.

a. The results of smear surveys are provided in Inclosure 1.

b. The results of concrete analysis are provided in Inclosure 2.

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

c. Surveys by direct radiation measurements indicated that the highest radiation values were obtained on the north, south, and west walls of the exposure room. The values ranged from 20-400 microroentgen per hour (μR/h) on contact as measured with an Eberline, Model PRM-7, Micro-R-Meter and up to 350 μR/Hr when measured with a Victoreen, Model 440, Ionization Chamber. These two methods of radiation measurements are in close agreement.

5. DISCUSSION.

a. Samples were taken from the wastewater holding tanks and the sewage system down stream from the holding tanks.

b. Core samples were taken off site and soil and vegetation samples were taken both on and off site.

c. The final report will be forwarded in about 60 days following analysis of the samples.

6. CONCLUSION. A review of the findings indicated that after decontamination the facility conformed to the requirements of Regulatory Guide 1.86.

7. RECOMMENDATION. None

FOR THE COMMANDER:

2 Incl
as

FRANK E. McDERMOTT
COL, MSC
Director, Radiation and
Environmental Sciences

CF:
Cdr, ERADCOM
Cdr, HSC (HSPA-P)

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

RESULTS OF ANALYZING WIPE TEST SAMPLES

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm² Gross Alpha Activity	Gross Beta Activity
1	L244	< 1.4	4.4 ± 2.5
2	L245	< 1.4	< 2.5
3	L246	< 1.4	< 2.5
4	L247	< 1.4	< 2.5
5	L248	< 1.4	< 2.5
6	L249	< 1.4	< 2.5
7	L250	< 1.4	< 2.5
8	L251	< 1.4	2.8 ± 2.0
9	L252	< 1.4	< 2.5
10	L253	< 1.4	6.0 ± 2.7
11	L254	< 1.4	2.6 ± 2.0
12	L255	< 1.4	< 2.5
13	L256	< 1.4	< 2.5
14	L257	< 1.4	< 2.5
15	L258	< 1.4	< 2.5
16	L259	< 1.4	3.6 ± 1.9
17	L260	< 1.4	< 2.5
18	L261	< 1.4	< 2.5
19	L262	< 1.4	14.6 ± 3.7
20	L263	4.7 ± 2.4	14.0 ± 3.6
21	L264	< 1.4	< 2.5
22	L265	< 1.4	6.2 ± 2.3
23	L266	< 1.4	7.0 ± 2.6
24	L267	3.2 ± 1.9	< 2.5
25	L268	< 1.4	5.2 ± 2.4
26	L269	< 1.4	< 2.5
27	L270	< 1.4	3.0 ± 2.0
28	L271	< 1.4	< 2.5
29	L272	< 1.4	< 2.5

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm^2 Gross Alpha Activity	Gross Beta Activity
30	L273	< 1.4	3.2 ± 2.2
31	L274	< 1.4	9.8 ± 3.2
32	L275	< 1.4	3.2 ± 2.3
33	L276	< 1.4	< 2.5
34	L277	< 1.4	< 2.5
35	L278	< 1.4	3.2 ± 2.4
36	L279	< 1.4	3.2 ± 2.1
37	L280	< 1.4	5.0 ± 2.4
38	L281	< 1.4	4.8 ± 2.3
39	L282	< 1.4	< 2.5
40	L283	< 1.4	< 2.5
41	L284	< 1.4	3.4 ± 2.1
42	L285	< 1.4	< 2.5
43	L286	< 1.4	< 2.5
44	L287	< 1.4	< 2.5

Alphus L. Jones
ALPHUS L. JONES, Chief
Radl & Biol Chem Div, USAEHA

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

INTERIM RESULTS OF ANALYZING CONCRETE SAMPLES

Sample Identification	RCB Lab No.	Microcurie per Gram ±2 Standard Deviations Europium-152 Activity	Europium-154 Activity	Cobalt-60 Activity
EX-N	RC1	$3.5 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.8 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$1.0 \times 10^{-5} \pm 0.4 \times 10^{-6}$
EX-S	RC2	$5.9 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$4.5 \times 10^{-6} \pm 0.8 \times 10^{-6}$	$3.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
ES In Pool	RC3	$1.6 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$1.4 \times 10^{-6} \pm 0.4 \times 10^{-6}$	$5.4 \times 10^{-6} \pm 0.3 \times 10^{-6}$
ES-W	RC4	$2.8 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.2 \times 10^{-6} \pm 0.5 \times 10^{-6}$	$1.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
EX LIFT-S	RC5	$1.1 \times 10^{-4} \pm 0.2 \times 10^{-5}$	$7.9 \times 10^{-6} \pm 0.9 \times 10^{-6}$	$3.0 \times 10^{-5} \pm 0.1 \times 10^{-5}$

Alphus L. Jones
ALPHUS L. JONES, Chief
Radl & Biol Chem Div, USAEHA

Rockwell International

Energy Systems Group
8900 DeSoto Avenue
Canoga Park, California 91304

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20314

S- 9 JUN 1969

8 APR 1969

IG TID

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (RFI-5) (FY 69)

THRU: Physicist in Charge, Diamond Ordnance Radiation Facility, Harry Diamond Laboratories, Washington, D. C. 20438
Commanding Officer, Harry Diamond Laboratories, Washington, D. C. 20438
Commanding General, United States Army Materiel Command Washington, D. C. 20315

TO: Chief of Staff
Department of the Army
ATTN: The Inspector General
Washington, D. C. 20315

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80 a reactor facility inspection of the Diamond Ordnance Radiation Facility was made during the period 12 - 14 March 1969. The facility was rated SATISFACTORY. (Only a rating of SATISFACTORY or UNSATISFACTORY was considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and the administrative and logistical support essential for its safe and efficient operation.

3. Evaluation was accomplished by questioning personnel; reviewing procedures, records and reports; observing reactor operations; inspecting the facility; and observing the response to a simulated emergency situation. Detailed results of the inspection appear in TAB B.

4. At the Diamond Ordnance Radiation Facility the team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' informal notes and a memorandum of minor irregularities. The team chief held a critique for the Commanding Officer, Harry Diamond Laboratories.

PROTECTIVE MARKING IS EXCLUDED
FROM AUTOMATIC TERMINATION
(PARA 18, AR 345-15)

Copy No. 1

IG TID
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (RFI-5) (FY 69)

II. SUMMARY OF SIGNIFICANT OBSERVATIONS

5. There were no deficiencies noted during the inspection.

6. The Physicist in Charge and his staff are commended for their competence in dosimetry and reactor physics, and the outstanding operating conditions at the reactor facility.

7. The Health Physics Section, Walter Reed Army Medical Center, is commended for providing outstanding health physics support.

III. RECOMMENDATIONS

8. It is recommended that Physicist in Charge, Diamond Ordnance Radiation Facility, correct the comments in TAB B.

IV. PROCESSING REPORT

9. AR 20-1 prescribes the procedures for processing this report.

George E. Lear

GEORGE E. LEAR
LTC, IG
Inspector General

2 Incl
TAB A - Msn & Org
TAB B - Observations

IG TID
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (RFI-5) (FY 69)

Copies furnished:

1.	Physicist in Charge, DORF	For ind and rtn thru channels
2.	Physicist in Charge, DORF	For retention
3.	CO, Harry Diamond Laboratories	For info and retention
4	and 5. CG, USAMC, ATTN: AMCSA-N	" " " "
6.	CRD, ATTN: CRDNCB	" " " "
7.	CofEngrs, ATTN: NPD	" " " "
8.	TSG	" " " "
9.	DCSLOG, HQ DA, ATTN: LOG-OM-ALB	" " " "
10.	ACSFOR, HQ DA, ATTN: FOR CM NU	" " " "
11.	DCSPER, HQ DA, ATTN: DCSPER-SD	" " " "
12.	Dir of Regulation, AEC	" " " "
13	and 14. Dir, Div of Reactor Development, AEC, ATTN: Asst Dir for Army Reactors	" " " "
15.	OTIG, HQ DA, ATTN: TID	Suspense Copy

Extract furnished:
CG, WRAMC

IG TID

TAB A

MISSION AND ORGANIZATION

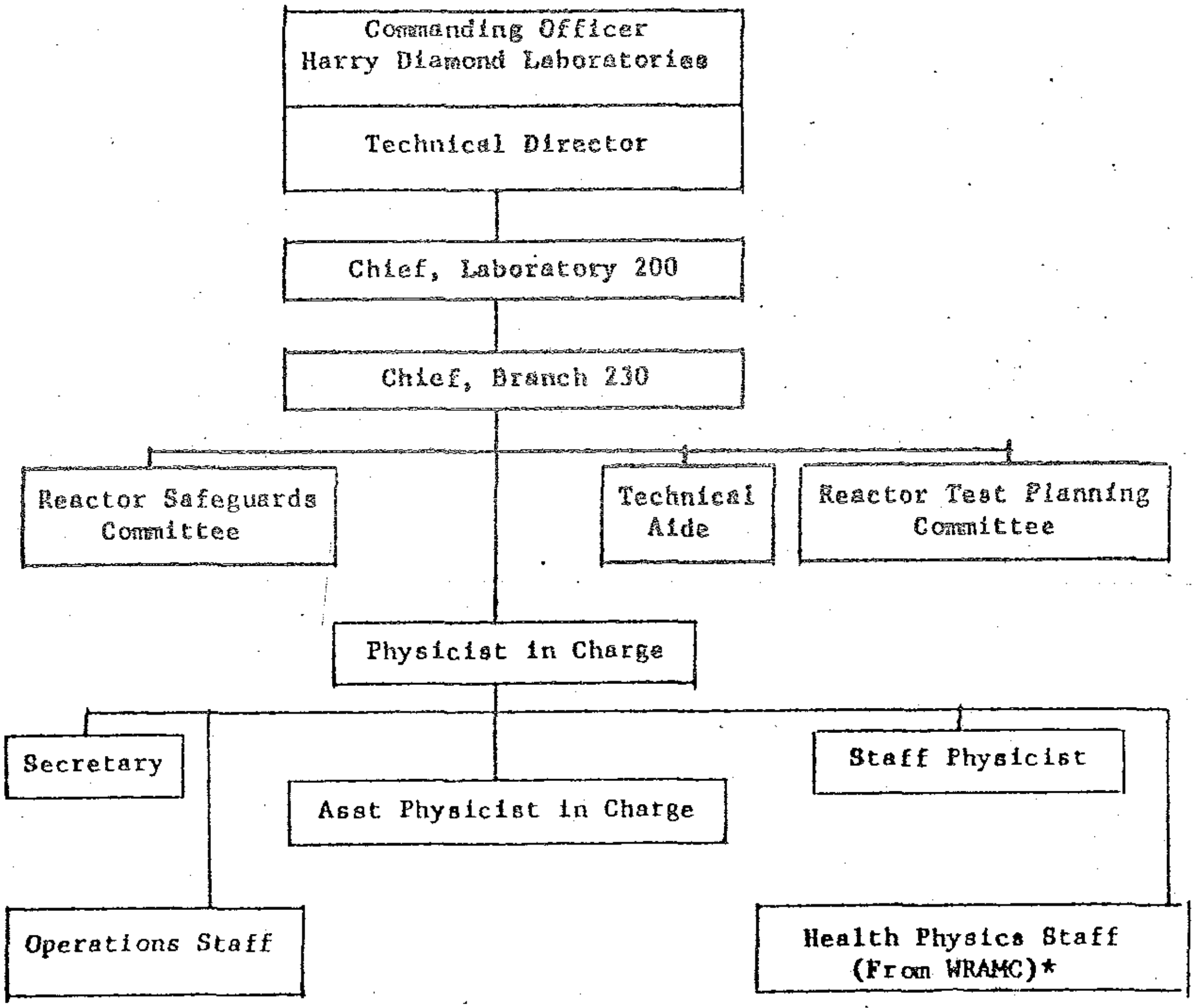
1. Mission. The mission of the Diamond Ordnance Radiation Facility as stated in Harry Diamond Laboratories Pamphlet 70-4 was: "to provide a source of radiation for experimental programs conducted at the Harry Diamond Laboratories or by other Federal Government Agencies and their contractors."

2. Organization. See attached organization chart.

PROTECTIVE MARKING IS EXCLUDED
FROM AUTOMATIC TERMINATION
(PARA 18, AR 345-15)

IG TID

ORGANIZATION



RECAP OF PERSONNEL

	Auth	Actual
Civilians	8	8
Enlisted Men*	2	2

Appendix 1 to TAB A

Protective Marking is Excluded
from Automatic Termination
(para 18, AR 345-15)

IG TID

TAB B

OBSERVATIONS

1. This inclosure contains detailed observations made during the inspection. These observations are classified as Factors Affecting Operations, Deficiencies, or Comments. The definitions of these terms are:

a. Factors Affecting Operations are observations of adverse situations or conditions correctable by a staff agency or headquarters other than that inspected.

b. Deficiencies are observations of adverse situations or conditions which are within the capability of the inspected reactor facility to correct and are important enough to require correction and report of corrective action.

c. Comments are observations of conditions which require correction and which can be corrected by the inspected reactor facility. The conditions are considered not serious enough to warrant classification as deficiencies. Corrective action is not reported to The Inspector General but is kept on record at the reactor facility.

2. No factors, deficiencies, or comments were reported for the following areas inspected: Mechanical Systems, Health Physics, Environmental Monitoring, and Industrial Safety.

3. The appendices to this TAB are:

APPENDIX	TITLE
1	Operations
2	Electrical and Instrumentation Systems
3	Simulated Emergency Situation

PROTECTIVE MARKING IS EXCLUDED
FROM AUTOMATIC TERMINATION
(PARA 18, AR 345-15)

IG TID

OPERATIONS

1. Factors Affecting Operations. None.

2. Deficiencies. None.

3. Comments.

a. At the conclusion of the FY 68 Reactor Facility Inspection made by the DAIG inspection team, a Memorandum of Minor Irregularities and Deficiencies was given to the Physicist in Charge. Paragraph 2 of that memorandum required that a memorandum of corrective action taken be retained on file. Corrective action had been taken but no memorandum of corrective action was available.

b. A comment under "Operations" in the DAIG Reactor Facility Inspection Report, 30 April 1968, was the observation that HDL Pamphlets 70-4 and 70-5 required "review and printing of new pamphlets". HDL Pamphlet 70-4 has been reviewed and reprinted. First indorsement, AMXD-RBB, Harry Diamond Laboratories, 14 June 1968, stated the "DORF staff will review, rewrite, and reissue" HDL Pamphlet 70-5 so that these tasks" will be completed within 180 days (1 December 1968)". The rewriting and reprinting of HDL Pamphlet 70-5 has not been made; however, the review was in progress.

PROTECTIVE MARKING IS EXCLUDED
FROM AUTOMATIC TERMINATION
(PARA 18, AR 345-15)

Appendix 1 to TAB B

IG TID

ELECTRICAL AND INSTRUMENTATION SYSTEMS

1. Factor Affecting Operations. None.

2. Deficiency. None.

3. Comment. Records indicated that scheduled maintenance on auxiliary Servo Chassis and Thermocouple Sampling circuit Nr. 1-26A had not been performed since 14 December 1967. Equipment Maintenance Log (DA Form 2409) required a twelve months maintenance interval.

IG TID

SIMULATED EMERGENCY SITUATION

1. Factors Affecting Operations. None.

2. Deficiencies. None.

3. Comments.

a. During monitoring of personnel leaving the reactor area, a health physics technician used his finger to gauge the distance from the survey instrument detector to the individual being monitored. This procedure could cause contamination of the health physics technician and the survey instrument.

b. Protective gloves were not worn by a health physics technician while handling possibly contaminated re-entry personnel dosimetry belts.

c. After return to the health physics control point, initial re-entry personnel removed their face masks before removing other protective clothing. Masks should be the last protective item to be removed by the initial re-entry team.

d. Initial re-entry personnel removed booties on the "hot side" of the health physics control line and then stood in that area. (FM 3-15, Section VIII, Nuclear Accident Contamination Control, 17 Jun 66 and NBS Handbook 92, Sections 3 and 5, 9 Mar 64)

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20314

S-

IG TID

5 FEB 1971

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 71)

THRU: Commanding Officer
Harry Diamond Laboratories
Washington, D. C. 20438

TO: Chief of Staff
United States Army
ATTN: The Inspector General
Washington, D. C. 20314

I. <u>GENERAL</u>

1. In accordance with AR 20-1 and AR 385-80 a reactor facility inspection of the Diamond Ordnance Radiation Facility was made during the period 18 - 20 January 1971. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and to the administrative and logistical support essential for its safe and efficient operation.

3. Evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations; and inspecting the facility. Detailed results of the inspection appear in TAB A.

4. The team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' informal notes and a memorandum of minor irregularities. The Commanding Officer, Harry Diamond Laboratories, was briefed on the results of the inspection.

Copy No. 9

[illegible] ONLY

[illegible]

II. SUMMARY OF SIGNIFICANT OBSERVATIONS

5. Radiation monitoring equipment was not installed in accordance with approved plans. (TAB A, para 2a)

6. There was a need to improve calibration support for radiation survey instruments. (TAB A, para 2b)

III. RECOMMENDATIONS

7. It is recommended that:

a. Commanding Officer, Harry Diamond Laboratories:

(1) Coordinate with the Commanding General, Walter Reed Army Medical Center, to correct the factor affecting operations in TAB A, paragraph 2a.

(2) Correct the comments in TAB A, paragraph 4.

b. Commanding General, Walter Reed Army Medical Center:

(1) Coordinate with the Commanding Officer, Harry Diamond Laboratories, to correct the factor affecting operations in TAB A, paragraph 2a.

(2) Correct the factors affecting operations in TAB A, paragraphs 2b and 2c.

IV. PROCESSING REPORT

8. AR 20-1 prescribes the procedures for processing this report.

SIGNED

1 Incl
TAB A - DORF

ROBERT K. O'CONNELL
LTC, IG
Inspector General

Copies furnished:
(See next page)

IG TID
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 71)

Copies furnished:

1.	CO, Harry Diamond Laboratories	For ind and rtn thru channels
2	and 3. CO, Harry Diamond Laboratories	For retention
4	and 5. CG, USAMC, ATTN: AMCSA-N	For info and retention
6.	DCSPER, DA, ATTN: DCSPER-SD	" " " "
7.	DCSLOG, DA, ATTN: LOG-OM-ALB	" " " "
8.	CRD, ATTN: CRDNCB	" " " "
9.	CofEngrs, ATTN: ENG-SO	" " " "
10.	TSG	" " " "
11.	Dir of Regulation, AEC	" " " "
12	and 13. Dir, Div of Reactor Development, AEC, ATTN: Asst Dir for Army Reactors	" " " "
14.	TIG, DA, ATTN: TID	Suspense Copy

IG TID

TAB A

DIAMOND ORDNANCE RADIATION FACILITY
WASHINGTON, D. C. 20438

1. Mission. The mission of the facility, as stated in Harry Diamond Laboratories Pamphlet 70-4, was: "to provide a source of radiation for experimental programs conducted at the Harry Diamond Laboratories or by other Federal Government agencies and their contractors."

2. Factors Affecting Operations

a. There was no instrument installed to monitor the reactor effluent air for particulates after it passed through the absolute air filters. The Walter Reed Army Medical Center Environmental Radiological Monitoring Plan, which was applicable to the facility, indicated that such a monitor was installed and was being checked daily. (WRAMC ENRADMON Plan, 15 Aug 66, para 6a(2)(b))

b. Recently, the facility had not been receiving timely support in the calibration of radiation survey instruments. From 19 November 1970 to 11 January 1971, only one of the thirteen instruments on hand at the facility was in calibration. The Health Physics Section, Walter Reed Army Medical Center, was responsible for providing calibration support.

c. During November and December 1970, environmental monitoring technicians collected fallout-washout samples at intervals ranging from 11 to 14 days rather than weekly as prescribed. (WRAMC ENRADMON Plan, TP 2-2-3, para 2)

3. Deficiencies. None.

4. Comments.

a. The console regulated AC voltage was not within tolerance. Since 28 September 1970, the voltage had exceeded continuously the specified limits (117 ± 3 volts). This overvoltage detracted from the accuracy of other parameters. (Harry Diamond Laboratory (HDL) Pamphlet 70-5, 1 Dec 69, para 2.3.3)

b. The voltmeter used to measure console regulated AC voltage was not in calibration. (HDL Pamphlet 70-5, para 2.3.3)

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20314

S-

IG TID

8 FEB 1971

SUBJECT: Reactor Facility Inspection of the Walter Reed Army Medical Center Research Reactor (FY 71)

THRU: Director, Walter Reed Army Institute of Research
Washington, D. C. 20012
Commanding General, Walter Reed Army Medical Center
Washington, D. C. 20012
The Surgeon General, Department of the Army
Washington, D. C. 20314

TO: Chief of Staff
United States Army
ATTN: The Inspector General
Washington, D. C. 20314

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80 a reactor facility inspection of the Walter Reed Army Medical Center (WRAMC) Research Reactor was made during the period 21 - 22 January 1971. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and to the administrative and logistical support essential to its safe and efficient operation.

3. Evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; and inspecting the facility. Detailed results of the inspection appear in TAB A.

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4. The team chief held a critique of the inspection and gave the Deputy Director, Division of Biochemistry, Walter Reed Army Institute of Research (WRAIR), a copy of the inspectors' informal notes and a memorandum of minor irregularities and deficiencies. The Executive Officer, WRAIR, was briefed on the results of the inspection.

II. SUMMARY OF SIGNIFICANT OBSERVATIONS

5. A contract had been awarded to deactivate the reactor facility. (TAB A, para 1)

III. RECOMMENDATIONS

6. It is recommended that:

a. Director, Walter Reed Army Institute of Research, correct the deficiency and comments in TAB A, paragraphs 3 and 4.

b. Commanding General, Walter Reed Army Medical Center, ensure expeditious deactivation of the reactor and release of excess personnel. (TAB A, para 1)

IV. PROCESSING REPORT

7. AR 20-1 prescribes procedures for processing this report.

1 Incl
TAB A - WRAMCRR

ROBERT K. O'CONNELL
LTC, IG
Inspector General

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IG TID
SUBJECT: Reactor Facility Inspection of the Walter Reed Army Medical Center Research Reactor (FY 71)

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TAB A

WALTER REED ARMY MEDICAL CENTER RESEARCH REACTOR
WASHINGTON, D. C. 20012

1. Mission. The mission of the Walter Reed Army Medical Center Research Reactor had been terminated. A contract had been awarded for deactivation of the reactor. Dismantling was expected to begin in May 1971 following approval of proposed procedures by the Army Reactor Systems Health and Safety Review Committee and The US Atomic Energy Commission. Pending deactivation, the staff remained on duty to insure that the reactor was maintained in a safe condition.

2. Factors Affecting Operations. None.

3. Deficiency. Environmental monitoring technicians were not maintaining quality control charts to insure the accuracy of each counting system used in radioanalysis. (WRAMC ENRADMON Plan, 10 Aug 66, SOP #1-2, para 2d)

4. Comments.

a. The female plug used to connect the horizontal thermocolumn shield door motor to a 208-volt power source was cracked and in need of replacement.

b. The standby electrical power source for operation of the emergency communication system did not function during a test involving loss of external power.

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DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20314

S- 3 0 JUN 1972

3 MAY 1972

DAIG-TI

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 72)

Commanding Officer
Harry Diamond Laboratories
Washington, DC 20438

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Diamond Ordnance Radiation Facility, Washington, DC, was made during the period 3 - 5 April 1972 by LTC Robert K. O'Connell, Office of The Inspector General. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and to the administrative and logistical support essential for its safe and efficient operation.

3. Evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations; and inspecting the facility. Detailed results of the inspection appear in TAB A.

4. The team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' informal notes and a memorandum of minor irregularities. The Chief, Nuclear Radiation Effects Laboratory, Harry Diamond Laboratories, was briefed on the results of the inspection.

II. SUMMARY OF SIGNIFICANT OBSERVATIONS

5. There was a need for a more effective preventive maintenance program. (TAB A, para 2a)

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DAIG-TI
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 72)

III. RECOMMENDATIONS

6. It is recommended that:

a. Commanding Officer, Harry Diamond Laboratories, correct the factor affecting operations, deficiencies, and comments in TAB A, paragraphs 2e, 3 and 4.

b. Commanding General, Walter Reed Army Medical Center, correct the factors affecting operations in TAB A, paragraphs 2a, 2b, 2c and 2d.

IV. PROCESSING REPORT

7. AR 20-1 prescribes the procedures for processing this report.

FOR THE INSPECTOR GENERAL:

1 Incl
TAB A - DORF

ROWLAND B. SHRIVER, JR.
Colonel, IG
Chief, Technical Inspections Office

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12	and 13. Dir, Div of Reactor Development, AEC, ATTN: Asst Dir for Army Reactors	" " " "
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TAB A

DIAMOND ORDNANCE RADIATION FACILITY
WASHINGTON, DC 20438

1. Mission. The mission of the facility, as stated in Harry Diamond Laboratories Pamphlet 70-4, was: "to provide a source of radiation for experimental programs conducted at the Harry Diamond Laboratories or by other Federal Government agencies and their contractors."

2. Factors Affecting Operations.

a. There was a need for a more effective facility preventive maintenance program.

(1) Safety relief valves on the air compressor and electric hot water generator had not been tested since installation. (EM 385-1-1, 1 Mar 67, para 21A02)

(2) Electrical distribution breakers were not being tested regularly. A recent courtesy test by an outside agency identified several inoperative breakers, which were replaced during the inspection. (Engineer Regulation 1130-2-303, 15 Dec 67, App 1, para 1203)

(3) There had been no routine preventive maintenance on the diesel-generator since 1968. A maintenance contract was under consideration.

b. Procedures for releasing radioactive liquid waste to the sanitary sewer did not insure that effluent activity was within prescribed limits. Although data on gross gamma activity of the effluent was recorded, there was no evidence of analysis to identify all contributing elements and establish proper limits accordingly. (AR 755-15, 4 Nov 66, para 22a(3))

c. During CY 1971, environmental samples were not collected at one soil station (Number 10) and two fallout-washout stations (Numbers 5 and 10). (WRAMC ENRADMON Plan, 15 Aug 66, Tab G, pg 1; Tab D, pg 7)

d. During the period 18 October 1971 - 26 November 1971, technicians did not complete weekly analysis of reactor water samples for the presence of Tritium because of faulty counting equipment. Later analysis indicated that the Tritium level had been much higher than normal.

e. The overhead hoist used to lift and move heavy objects in the reactor room had not been load-tested since 1970. (A recurring observation.)

3. Deficiencies.

a. Reactor operators were using an uncalibrated voltmeter (Tripplet 630A) to check console AC regulated voltage. A calibrated electrostatic voltmeter was available for cross-check, but was considered undesirable for routine use. It was indicated that support personnel would not calibrate the Tripplet voltmeter. (HDL Pamphlet 70-5, 1 Dec 69, para 2.3.3)

b. There was no assurance that the quantity of neutron-activated materials on hand was within prescribed limits; the facility had recently acquired a large quantity of activated graphite without establishing the total activity involved. (AEC License No. BML 08-02534006)

c. There were no portable neutron survey instruments available. Beginning in January 1972, all seven instruments belonging to the facility had been sent away for calibration.

4. Comments

a. During modification of the reactor pool water-level alarm system, wires of a 6-volt system were left bare and exposed. A short-circuit in this system would have opened a common fuze and deactivated the water activity monitor and the water temperature indicator.

b. There were no records or markings associated with CO_2 fire extinguishers to indicate original charged-weight and dates on which future testing would be required. (EM 385-1-1, 1 Mar 67)

c. The odor of gas fumes was detectable near the second floor boiler.

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20314

S- 3 0 JUN 1972

DAIG-TI

9 MAY 1972

SUBJECT: Reactor Facility Inspection of the Army Pulse Radiation Facility (FY 72)

Commanding Officer
US Army Aberdeen Research and Development Center
Aberdeen Proving Ground, Maryland 21005

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Army Pulse Radiation Facility (APRF), Aberdeen Proving Ground, Maryland, was made during the period 10 - 12 April 1972, by LTC Robert K. O'Connell, Office of The Inspector General. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and to the administrative and logistical support essential for its safe and efficient operation.

3. Evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations; and inspecting the facility. Detailed results of the inspection appear in TAB A.

4. The team chief held a critique of the inspection and gave the Chief, Reactor Support Branch, a copy of the inspectors' informal notes and a memorandum of minor irregularities and deficiencies. The Commanding Officer, US Army Aberdeen Research and Development Center (ARDC), attended the critique.

II. SUMMARY OF SIGNIFICANT OBSERVATIONS

5. There were several recurring observations indicating a need for additional corrective effort. (TAB A, paras 2a, 2b, 3b, 3f, and 3g)

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6. Excess special source material had been on hand since October 1969. (TAB A, para 2a)

7. The facility required more effective support in eliminating potential fire and safety hazards. (TAB A, paras 2b, 2c, and 2d)

8. There was a need for greater precision in the monitoring and release of radioactive substances. (TAB A, paras 3a, 3b, 3c, and 3d)

III. RECOMMENDATIONS

9. It is recommended that:

a. Commanding Officer, US Army Aberdeen Research and Development Center, correct the observations in TAB A, paragraphs 2a, 3, and 4.

b. Commanding Officer, Aberdeen Proving Ground, correct the factors affecting operations in TAB A, paragraphs 2b, 2c, and 2d.

c. Commanding General, United States Army Materiel Command, assist the facility in correcting the factor affecting operations in TAB A, paragraph 2a.

IV. PROCESSING REPORT

10. AR 20-1 prescribes the procedures for processing this report.

FOR THE INSPECTOR GENERAL:

1 Incl
TAB A - APRF

ROWLAND B. SHRIVER, JR.
Colonel, IG
Chief, Technical Inspections Office

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DAIG-TI
SUBJECT: Reactor Facility Inspection of the Army Pulse Radiation Facility (FY 72)

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TAB A

ARMY PULSE RADIATION FACILITY
ABERDEEN PROVING GROUND, MARYLAND 21005

1. Mission. The mission of the facility, as provided by the Commander, Aberdeen Research and Development Center, was: "to operate and maintain a pulse reactor facility to provide a radiative environment simulating nuclear radiation (neutron) produced by nuclear weapons."

2. Factors Affecting Operations.

a. Excess special source material from a damaged reactor core had been awaiting disposal since October 1969. During February 1972, the facility requested additional funds to finance reprocessing of the material in conjunction with returning it to the Atomic Energy Commission. Informal inquiry at Headquarters, United States Army Materiel Command, indicated that necessary funds would have to be reprogrammed locally. This observation was included in two previous inspection reports to insure compliance with an AEC/DA Memorandum in which the Army agreed to return such material when it was no longer required at the facility. (Memorandum of Understanding between the Atomic Energy Commission and the Department of the Army, signed 1 September 1966 by the Secretary of the Army)

b. Circuit breakers, transformers, and relays supporting the facility were not being tested to insure operability. (A recurring observation.) (Engineer Regulation 1130-2-303, 15 Dec 67, App 1, para 1203)

c. Procedures for testing facility safety valves for pressurized systems were not adequate. No effort was being made to demonstrate the reliability of equipment gages used as part of the test. (EM 385-1-1, 1 Mar 67, para 21A02)

d. There were fire hazards associated with a temporary fuel oil storage and distribution system located in the mechanical equipment room:

(1) Frail fuel lines on the floor surface had no protection against physical damage.

(2) The mobile fuel tank was not adequately secured to prevent movement.

(3) The fuel tank was not electrically grounded.

3. Deficiencies.

a. Procedures for disposing of radioactive liquid wastes did not insure that the activity of such wastes was within allowances. Although liquid wastes were being released within the facility grounds rather than to a sewer, the activity at the boundary of the restricted area had not been established. Since the identity of the waste radionuclides had not been precisely determined, the activity recorded for liquid released within the restricted area was often greater than that allowed at the boundary. (AR 755-15, 4 Nov 66, para 22a(2))

b. There was not sufficient assurance that the radioactivity of the facility air effluent was within acceptable levels. Since air effluent was not being monitored continuously and air flow meters were not calibrated, facility data concerning maximum release rates and total activity released were based largely on estimates without any identifiable supporting analysis. In addition, there was no information regarding concentration of radioactive gases at the boundary of the restricted area. (A recurring observation.) (AR 755-15, para 22a(2); AR 385-80, 16 Oct 70, para 5-1(b)(5); Technical Specifications for APRF, 12 Apr 71, Chap VII, para 3c)

c. Liquid radioactive waste was being released indirectly into an unrestricted area without obtaining the approval required by the local government. (AR 755-15, para 22a(2))

d. Remote area radiation monitors and particulate air monitors were not being calibrated at quarterly intervals. Area monitors had been last calibrated during July 1971. Particulate air monitors were being calibrated during March and October, except for the air-flow meters which were being ignored. (Technical Specifications for APRF, Chap V, paras Q and R)

e. There was no evidence that quarterly checks of the reactor motion interlock were being performed. (Technical Specifications for APRF, Chap V, para L; Chap VII, para E2)

f. There was no preventive maintenance being accomplished on the batteries used to power radio communications and remote annunciation from the reactor to the laboratory building. (A recurring observation.) (National Electric Code, para 700-4)

g. Corrective action had not been taken or recorded with respect to the Memorandum of Minor Irregularities left at the facility following the previous reactor facility inspection. (AR 20-1, 22 Aug 68, para 2-12)

4. Comment. Entries on tags attached to facility fire extinguishers indicated these items were not being checked each month for serviceability. (AR 420-90, 27 Apr 70, paras 2-7a and d)

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20314

S-4 JUL 1973

REPLY TO
ATTENTION OF:

DAIG-TI

3 MAY 1973

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 73)

Commander
Harry Diamond Laboratories
Washington, D. C. 20438

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Diamond Ordnance Radiation Facility, Washington, DC, was made during the period 16 - 18 April 1973, by LTC Vincent P. De Fatta, Office of The Inspector General. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and to the administrative and logistical support essential for its safe and efficient operation.

3. Evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations; and inspecting the facility. Detailed results of the inspection appear in TAB A.

4. The team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' informal notes and a memorandum of minor irregularities and deficiencies. The Chief, Nuclear Vulnerability Branch, Harry Diamond Laboratories, attended the critique, and the Commander, Harry Diamond Laboratories, was briefed on the results of the inspection.

II. SUMMARY OF SIGNIFICANT OBSERVATIONS

5. There was a need for a more effective proventive maintenance program. (TAB A, paras 2c, 2d, 3k, 4e, and 4g)

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6. There was a need for review, revision and updating of basic facility documentation. (TAB A, paras 3f, i, and 4f)

7. Additional command emphasis was needed on the facility's personnel reliability program. (TAB A, paras 3a, 3b, and 3c)

III. RECOMMENDATIONS

8. It is recommended that:

a. Commander, Harry Diamond Laboratories, correct and ensure non-recurrence of the observations in TAB A, paragraphs 3 and 4c through 4g.

b. Commander, Health Services Command, correct and ensure non-recurrence of the observations in TAB A, paragraphs 2, 4a, and 4b.

IV. PROCESSING REPORT

9. AR 20-1 prescribes the procedures for processing this report.

FOR THE INSPECTOR GENERAL:

SIGNED

ROWLAND B. SHRIVER, JR.
Colonel, IG
Chief, Technical Inspections Office

1 Incl
TAB A - DORF

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Cdr, Health Svcs Cmd

DAIG-TI

TAB A

DIAMOND ORDNANCE RADIATION FACILITY
WASHINGTON, DC 20438

1. Mission. The mission of the Diamond Ordnance Radiation Facility is contained in the HDL Pamphlet 70-4, paragraph 3.1, which states: The mission of the Harry Diamond Laboratories is, in part, to:

a. Investigate and determine the susceptibility of electronic materiel to nuclear weapons radiation environment, mechanisms of those effects, and ways and means of developing less susceptible materiel.

b. Conduct research and development in the various physical science and engineering fields directed toward meeting the military characteristics for fuzing or of related items.

2. Factors Affecting Operations.

a. The dosimeters issued for health and safety purposes did not have valid DA labels 80 overprinted with the letters "I & C" (Inspection and Certification) affixed. (TB 750-242-3, 21 Nov 69, para 3a(2))

b. Fallout - washout sample station 5, Main Section (roof of out-patient building) was not collected or counted from 22 September 1972 to 10 November 1972. The Health Physics Officer did not make this fact known to the Physicist in Charge and therefore the incident was not reported in accordance with applicable regulations. (WRAMC ENRADMON PLAN, 15 Feb 72, App C, para B; Technical Specifications for the DORF Reactor, 24 Nov 71, Vol 6, para 8e; AR 385-40, Aug 72, para 2-14 (C2))

c. Safety relief valves on the electric hot water heater had not been tested since installation. (EM 385-1-1, 1 Mar 72, para 21A 02) (Recurring)

d. There was no record available to indicate that electrical distribution breakers were being tested regularly. (ER 1130-2-303, 15 Dec 67, App 1, para 1203) (Recurring)

3. Deficiencies.

a. In at least one instance an individual assigned to a controlled position had not initialled the new job description on his DA Form 2706. (AR 50-5, paras 3-21a and 3-26(2))

b. Nuclear Duty Position Strength and Reliability Reports, RCS CSFOR-161 (DA Form 3637-R) were not prepared and forwarded. (AR 50-5, para 3-29)

c. A nuclear duty position roster was not being maintained. (AR 50-5, para 3-3)

d. During the nuclear accident/incident (simulated) the following were observed:

(1) The emergency procedures in HDL Pamphlet 70-5 and the Interservice Agreement between the Cdr, WRAMC and the Cdr, HDL were in conflict concerning Public Affairs Activities during reactor accident/incidents, (HDL Pam 70-5, 1 Dec 69, para 4.4.10 and Interservice Agreement between Cdr, WRAMC and the Cdr, HDL, undated, para 13)

(2) The accident was not reported to the Chief of Engineers. (AR 385-40, Aug 72, para 6-2)

(3) Notification of emergency condition was not in accordance with approved procedures, i.e., the person designated as being in charge notified the fire department, the military police and the hospital for an ambulance, rather than notifying the military police only. (HDL Pam 70-5, para 4.4.3)

e. The Americium-Beryllium start-up source had not been leak tested at six-month intervals. (USAEC License 08-02534-09, 12 Jul 72, Condition 13 A(1))

f. Several basic facility documents were in need of revision and updating, for example:

(1) The technical specifications for the facility did not reflect the changes required by the installation of the Instrumentation and Control Modification. For instance, ARCHS approved the SSAR which allows Mode I steady state operation to a maximum of 250KW, yet the technical specifications limit the Mode I steady state operation to 100KW. (AR 385-80, 16 Oct 70, para 3-1c)

(2) The facility emergency plan contained in HDL Pamphlet 70-5 does not meet the requirements of Appendix B, AR 385-80 nor had it been approved by ARCHS. (AR 385-80, para 3-1f and 3-2c; Table 3-1 and App B)

g. Modifications were being performed without the approval of the Chief, Nuclear Vulnerability Branch. For example, on 1 March 1973 a 220K Ohm resistor was replaced in the console system with a 100K Ohm resistor and on 21 June 1972 a 1 K Ohm resistor in the area monitoring system was changed to a 3 K Ohm resistor. (HDL Pam 70-4, 31 Jan 69, w/Ch 1, para 5.8J)

h. A malfunction which occurred on 17 July 1972 to the Transient Rod Indicator, was not recorded in the Reactor Log Book. (HDL Pam 70-5, 1 Dec 69, part 2, page 2-2)

i. The operating procedures for the recently installed reactor console had not been reviewed by the Reactor Safeguard Committee or approved by the Chief of Nuclear Vulnerability Branch 280. (HDL Pam 70-4, paras 5.5 and 5.6)

j. Emergency reporting instructions were not conspicuously posted and some of the emergency telephone numbers listed thereon were incorrect. (EM 385-1-1, para 13 G03, 1 Mar 67)

k. The velometer used to verify the stack flow rate had not been calibrated at the prescribed intervals. (TB 750-236, Feb 72, Sec II, page 365)

4. Comments.

a. In at least two locations along the perimeter fence of the facility, vines had grown and entangled within the chain link to the extent that the view of the area outside the fence was completely obstructed.

b. Indications were that the screening procedures required by Chapter 3, AR 50-5 were not being accomplished in a timely manner; i.e., several instances were noted where DA Form 2706's had been executed as late as 16 April 1973.

c. The hot water generators located in the 1st floor "machinery room" of the basement "warm room" were installed at a height that prevented visual inspection or operation of controls from the floor level.

d. Drills of Facility Emergency Procedures were not conducted on a quarterly basis. Dates of drills recorded were 28 March 1972, 8 August 1972, 27 February 1973, and 6 April 1973. (Technical Specifications for the DORF Reactor, 24 Nov 71, para VI-4-2d)

e. There was a need for a more effective facility preventive maintenance program. For instance:

(1) Maintenance card DA Form 2409, on oscilloscope 316 Serial #263, was not up to date. Last entry was 26 February 1969.

(2) Micro-Microameter type 413 Serial #662515713-01 had not been calibrated since 20 January 1970.

(3) Micro-Microameter type 410c Serial #19681 was past calibration due date. Records (DA Form 2409) indicated last calibration performed 8-71. Instrument was supposed to be calibrated on 3092.

f. The facility had not developed adequate guidance to facilitate recording of malfunctions. For instance the facility had not specified:

(1) What constitutes a malfunction.

(2) Reporting requirements to include where information is entered, (i.e., malfunction log, station log) what information is required, and what corrective action was taken.

(3) Review of malfunctions by facility personnel to check for health and safety considerations. (Para VI-5-8, page 63, Technical Specifications for DORF Reactor)

g. A wire leading to the reactor pool water-level alarm system was bare and exposed.

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL
WASHINGTON, D.C. 20310

REPLY TO
ATTENTION OF:

S-2 2 APR 1974

20 FEB 1974

DAIG-TI

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 74)

Commander
Harry Diamond Laboratories
Washington, D. C. 20438

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Diamond Ordnance Radiation Facility, Washington, D.C., was made during the period 28 - 30 January 1974, by LTC Thomas A. Epperson, Office of The Inspector General. The facility was rated UNSATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The facility was rated UNSATISFACTORY because one individual, assigned and listed on the nuclear duty position roster, was disqualified for assignment to or retention in a nuclear duty position. (TAB A, para 2a)

3. Immediate action was taken to correct the unsatisfactory condition. A reinspection was then made covering the previously deficient area, and the facility was rated SATISFACTORY.

4. The rating applied to the technical, health physics, and safety aspects of the operation of the facility, and to the administrative and logistical support essential for its safe and efficient operation.

5. Evaluation was accomplished by questioning personnel; reviewing procedures, records and reports; observing reactor operations; and inspecting the facility. Detailed results of the inspection appear in TAB A.

6. The team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' informal notes and a memorandum of minor irregularities and deficiencies. The Chief, Nuclear

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Effects Simulation Technology Branch, attended the critique, and the Commander, Harry Diamond Laboratories was briefed on the results of the inspection.

II. SUMMARY OF OBSERVATIONS

7. Additional command emphasis was needed on the facility's personnel reliability program. (TAB A, para 2a)

8. Additional training in emergency procedures was required. (TAB A, paras 2b, 3c and 3d)

III. RECOMMENDATIONS

9. It is recommended that:

a. Commander, Harry Diamond Laboratories, correct and ensure nonrecurrence of the observations noted in TAB A, paragraphs 2, 3b, 3c and 3d.

b. Commander, Health Services Command, correct and ensure nonrecurrence of the observation noted in TAB A, paragraph 3a.

IV. PROCESSING REPORT

10. AR 20-1 prescribes the procedures for processing this report.

FOR THE ACTING THE INSPECTOR GENERAL:

SIGNED

ROWLAND B. SHRIVER, JR.
Colonel, IG
Chief, Technical Inspections Division

1 Incl
TAB A - DORF

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SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (FY 74)

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12 and 13.	Dir, Div of Reactor Development, AEC, ATTN: Asst Dir for Army Reactors	" " " "
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DAIG-TI

TAB A

DIAMOND ORDNANCE RADIATION FACILITY
WASHINGTON, D. C. 20438

1. Mission. The mission of the facility was to:

a. Investigate and determine the susceptibility of electronic materiel to nuclear weapons radiation environment, mechanisms of those effects, and ways and means of developing less susceptible materiel.

b. Conduct research and development in the various physical science and engineering fields directed toward meeting the military characteristics for fuzing, or of related items.

2. Deficiencies.

a. One individual, assigned to duty and listed on the Nuclear Duty Position Roster, was disqualified for assignment to or retention in a nuclear position. (AR 50-5, 8 Nov 73, para 3-4a(2))

b. During the simulated emergency drill, no one was assigned to direct emergency personnel. (HDL Pam 70-5, 1 Dec 69, Part 4, Annex D, para 1b)

c. A stayplex high volume air sampler located in the emergency cache was not calibrated at the required six month intervals. (MEDEC, SOP No 2-3, 19 Dec 72, para)

d. Eleven self-reading pocket dosimeters maintained for use in emergency situations were not inspected and certified at the required six month intervals. (TB 750-242-3, 21 Nov 69, para 4)

e. Two compressed gas cylinders, which were not in use, did not have the regulator removed and the cylinder cap installed. (EM 385-1-1, 1 Mar 67, para 21.D.11)

3. Comments.

a. In at least four locations along the perimeter fence of the facility, vines had grown and entangled within the chain link to the extent that the view of the area outside the fence was completely obstructed. The situation will be compounded when the summer growing season arrives. (Recurring from FY 73)

b. There were no standard procedures for recording the results of the maintenance performed on the electrical, instrumentation and mechanical systems.

c. During the simulated emergency drill it was noted that:

(1) Precautions were taken to prevent the spreading of contamination.

(2) Once emergency personnel were inside the reactor facility building, action was not taken to control the movement of personnel.

(3) Personnel responding to the emergency were not adequately briefed on the conditions that existed within the facility.

d. The actions of the reactor facility staff during the simulated emergency drill indicated a need for additional training.

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WASHINGTON, D.C. 20310

REPLY TO ATTENTION OF:

DAIG-TI

S-18 JUN 1975

21 APR 1975

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (FY 75).

Commander
Harry Diamond Laboratories
2800 Powder Mill Road
Adelphi, Maryland 20783

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Diamond Ordnance Radiation Facility, Washington, DC was made during the period 24 - 27 March 1975, by Colonel Otto C. Doerflinger, Jr., Office of The Inspector General. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility and to the administrative and logistical support essential for its safe and efficient operation.

3. The evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations and a simulated emergency drill; and inspecting the facility. Detailed results of the inspection appear in TAB A.

4. The team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' informal notes and a memorandum of minor irregularities and deficiencies. The Chief, Nuclear Radiation Effects Laboratory and the Associate Director for Administration attended the critique.

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DAIG-TI
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (FY 75)

II. SUMMARY OF OBSERVATIONS

5. The ENRADMON plan required updating. (TAB A, para 3a)

6. The facility was not fully complying with USAMC physical security requirements. (TAB A, paras 2, 3b and 3c)

III. RECOMMENDATIONS

7. It is recommended that:

a. Commander, Harry Diamond Laboratories, correct and ensure nonrecurrence of the observations noted in TAB A, paragraphs 3 and 4.

b. Commander, United States Army Materiel Command:

(1) Assist the facility in correcting the observations noted in TAB A, paragraphs 3b and 3c.

(2) Comment on the observation noted in TAB A, paragraph 2.

IV. PROCESSING REPORT

8. AR 20-1 prescribes the procedures for processing this report.

FOR THE INSPECTOR GENERAL AND AUDITOR GENERAL:

SIGNED

ROWLAND B. SHRIVER, JR.
Colonel, IG
Chief, Technical Inspections Division

1 Incl
TAB A - DORF

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DAIG-TI
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (FY 75)

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12	and 13. Dir, Div of Reactor Research & Development, US Energy, Research & Development Admin, ATTN: Asst Dir for Army Reactors	" " " "
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TAB A

DIAMOND ORDNANCE RADIATION FACILITY
WASHINGTON, DC 20438

1. Mission. The mission of the facility was to:

a. Investigate and determine the susceptibility of electronic materiel to nuclear weapons radiation environment, mechanisms of those effects, and ways and means of developing less susceptible materiel.

b. Conduct research and development in the various physical science and engineering fields directed toward meeting the military characteristics for fuzing or of related items.

2. Factor Affecting Operations. The lock and key control procedures prescribed by higher headquarters do not take into consideration the physical design of the research reactor nor the limited periods during which the reactor is manned.

3. Deficiencies.

a. The ENRADMON plan needs immediate updating in order to evaluate the impact, if any, on the current operating procedures. Due to the significant changes in man-made terrain features, plant life and population density, this updating should include a detailed reevaluation of the diffusion model for gaseous effluents and of the necessary monitoring requirements to determine the impact of the effluent releases. (AR 385-80, 20 Dec 73, para 4-3c)

b. The clear zone was not being monitored to detect the presence of individuals or vehicles within the zone so as to allow response by members of the security force at the time of penetration of the area. (AMCR 190-3, 8 Nov 73, w/C1, para 9-3c)

c. The reactor building which contains highly radioactive special nuclear material was not locked with two separate padlocks. (AMCR 190-3, para 9-9a)

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DAIG-TI

4. Comments.

a. The interpretation of data and the conclusions contained in the Draft Summary of Environmental Radiation Levels for the year 1974 were not adequately supported by the information (e.g., unusual trends presented were not discussed, ambient radioactive levels were not properly monitored and radiological impact on the local population was not considered).

b. Procedures for and method of notification of security personnel when a duress situation exists should be reviewed.

c. Maintenance of thermocouple wiring on the reactor head needed additional emphasis.

TAB A

DEPARTMENT OF THE ARMY
OFFICE OF THE INSPECTOR GENERAL AND AUDITOR GENERAL
WASHINGTON, D.C. 20310

REPLY TO
ATTENTION OF:

DAIG-TI

S: 24 MAY 1976

24 MAR 1976

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (FY 76)

Commander
Harry Diamond Laboratories
2800 Powder Mill Road
Adelphi, Maryland 20783

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Diamond Ordnance Radiation Facility, Washington, DC was made during the period 9 - 12 March 1976, by Colonel Otto C. Doerflinger, Jr., Office of The Inspector General. The facility was rated SATISFACTORY. (Only ratings of SATISFACTORY and UNSATISFACTORY were considered.)

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility and to the administrative and logistical support essential for its safe and efficient operation.

3. The evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations and a simulated emergency drill; and inspecting the facility. Detailed results of the inspection appear in TAB A.

4. The team chief held a critique of the inspection and gave the Physicist in Charge a copy of the inspectors' draft notes and a memorandum of minor irregularities and deficiencies. The Chief, Nuclear Effects Simulation Technology Branch, attended the critique.

Copy No. 10

DAIG-TI
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (FY 76)

II. SUMMARY OF OBSERVATIONS

5. The ENRADMON plan required updating. (Recurring from FY 75.) (TAB A, para 2a)

6. An effective plan to apprehend armed intruders had not been developed. (TAB A, para 2b)

III. RECOMMENDATIONS

7. It is recommended that:

a. Commander, Harry Diamond Laboratories, correct and ensure non-recurrence of the observations noted in TAB A, paragraphs 2 and 3.

b. Commander, United States Army Materiel Development and Readiness Command, assist the facility in correcting the observation noted in TAB A, paragraph 2b.

IV. PROCESSING REPORT

8. AR 20-1 prescribes the procedures for processing this report.

FOR THE INSPECTOR GENERAL AND AUDITOR GENERAL:

SIGNED

1 Incl
TAB A - DORF

OTTO C. DOERFLINGER, JR.
Colonel, IG
Chief, Technical Inspections Division

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TAB A

DIAMOND ORDNANCE RADIATION FACILITY
WASHINGTON, D. C. 20438

1. Mission. The mission of the facility was to:

a. Investigate and determine the susceptibility of electronic materiel to nuclear weapons radiation environment, mechanisms of those effects, and ways and means of developing less susceptible materiel.

b. Conduct research and development in the various physical science and engineering fields directed toward meeting the military characteristics for fuzing or of related items.

2. Deficiencies.

a. The ENRADMON plan needed immediate updating in order to insure that the data being collected were adequate to determine the amount and type of radioactive material being released. (Recurring from FY 75.) (AR 385-80, 20 Dec 73, para 4-3c)

b. An effective plan to apprehend an organized group of armed intruders had not been developed. (DA message 032350Z Jul 75)

c. Visitors were not issued or required to wear identification badges. (AMCR 190-3, paragraph 9-6 and Physical Security Plan, Diamond Ordnance Radiation Facility, 25 Sep 1975, paragraph 3a(3))

3. Comments.

a. During the conduct of an emergency response exercise, the following were noted:

(1) A more responsive method of notifying the WRAMC health physics personnel was needed.

(2) Procedures used did not control adequately all personnel entering and leaving the controlled area.

(3) Greater participation by medical personnel was needed to insure timely medical support in all emergency situations.

b. Two compressed gas cylinders had not been hydrostatically tested within five years. (Recurring from FY 75)

c. Approximately 25% of all end-of-wearing period readings for pocket dosimeters had not been recorded.

d. Action levels had not been defined for monitoring of pool water for radioactive contamination.

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S-7 JUN 1977

REPLY TO ATTENTION OF:

DAIG-TI

6 APR 1977

SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (RFI 3-77)

Commander
Harry Diamond Laboratories
2800 Powder Mill Road
Adelphi, Maryland 20783

I. GENERAL

1. In accordance with AR 20-1 and AR 385-80, a reactor facility inspection of the Diamond Ordnance Radiation Facility, Washington, DC, was made during the period 21 - 24 March 1977, by Colonel Otto C. Doerflinger, Jr., Office of The Inspector General. The facility was rated SATISFACTORY (SUPPORT UNSATISFACTORY) because of failure to provide a secure environment for the nuclear reactor as noted in paragraph 6 below. Immediate action was taken to correct the Unsatisfactory condition and a reinspection rated the deficient area as SATISFACTORY.

2. The rating applied to the technical, health physics, and safety aspects of the operation of the facility and to the administrative and logistical support essential for its safe and efficient operation.

3. The evaluation was accomplished by questioning personnel; reviewing procedures, records, and reports; observing reactor operations and a simulated emergency drill; and inspecting the facility.

4. The mission of the facility was to:

a. Investigate and determine the susceptibility of electronic materiel to nuclear weapons radiation environment, mechanisms of those effects, and ways and means of developing less susceptible materiel.

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b. Conduct research and development in the various physical science and engineering fields directed toward meeting the military characteristics for fuzing or of related items.

5. The team chief held a critique of the inspection and provided the facility a copy of the inspectors' draft notes and a memorandum of minor irregularities and deficiencies. The Chief, Nuclear Radiation Effects Laboratory, attended the critique.

II. SUMMARY OF OBSERVATIONS

6. Factor Affecting Operations. One health physics operator was assigned to a controlled nuclear duty position for a period in excess of 180 days with a SECRET security clearance based on an Entrance National Agency Check. No action had been taken to upgrade the investigation. (AR 50-5, 15 Jul 76, para 3-3b(1) and DARCOM Supplement 1 to AR 50-5, 10 Sep 76, addition to para 3-3b(3)(e)) (Attributed to Walter Reed Army Medical Center)

7. Deficiencies.

a. A drainage pipe under the perimeter fence had an opening greater than 96 square inches and was not protected by securely fastened metal grills or bars. (AMC Reg 190-3, 8 Nov 73, w/Ch 1, para 3-9b(5))

b. The emergency equipment storage shed was not posted with a "Radiation Area" sign when the measured level of radiation outside the shed was in excess of 10 mr/hr. (AR 385-30, 18 Nov 71, para 3-1b)

III. RECOMMENDATIONS

8. It is recommended that:

a. Commander, Harry Diamond Laboratories, correct and ensure nonrecurrence of the observation noted in paragraph 7.

b. Commander, United States Army Health Services Command, ensure nonrecurrence of the observation in paragraph 6.

IV. PROCESSING REPORT

9. AR 20-1 prescribes the procedures for processing this report.

DAIG-TI
SUBJECT: Reactor Facility Inspection of the Diamond Ordnance Radiation Facility (DORF) (RFI 3-77)

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OTTO C. DOERFLINGER, JR.
Colonel, IG
Chief, Technical Inspections Division

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U 8-70

TECHNICAL SPECIFICATIONS FOR THE

DIAMOND ORDNANCE RADIATION FACILITY REACTOR

JANUARY 1970

W. L. GIESELER

J. M. McGARRITY

HARRY DIAMOND LABORATORIES

U.S. ARMY MATERIEL COMMAND

WASHINGTON, D.C.

Incl 7

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DEFINITIONS

A. Safety Limits are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

B. Limiting Safety System Settings are settings for automatic protective devices related to those variables having significant safety functions. The setting shall be chosen such that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded.

C. Limiting Conditions for Operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

D. Surveillance Requirements are the requirements relating to test, calibration or inspection to assure that operation will be within the Technical Specifications.

E. Operable means a component or system is capable of performing its intended function in a normal manner.

F. Operating means a component or system is performing its intended function in its normal manner.

G. Measuring Channel is a combination of sensors, cables, amplifiers and output devices which are connected for the purpose of measuring the value of a reactor variable.

H. Scram is the interruption of magnet holding current of the standard control rods or the air holding pressure of the transient rod such that the control rods fall by gravity into the reactor core.

I. Safety Interlock System is that combination of measuring channels and associated circuitry which forms the automatic protective system for the reactor or provides information which requires manual protective action to be initiated.

J. Administrative Controls are the provisions relating to organization and management, procedure, record keeping, review and audit, and reporting that are considered necessary to assure operation of the reactor in a safe manner.

K. Steady State Mode is the reactor condition when the reactor mode selector switch is in the steady state or automatic position. In this mode, the reactor power may be held constant or changed either manually or by servo-action of a control rod.

L. Square Wave Mode is a form of steady state operation when the reactor mode selector switch is in square wave mode position. In this mode the transient rod is ejected from the core and held out until scrammed by operator action. The reactor power is increased on periods less than one second and is held at a constant power by servo-action of the transient rod as determined by a power demand setting.

M. Pulse Mode is the reactor condition when the reactor mode switch is in the pulse mode position. In this mode, the transient rod is ejected from the core and is scrammed automatically. The reactor power increases on periods less than one second. The power excursion is self quenching by the prompt negative temperature coefficient of reactivity.

N. Standard Control Rod is a rod having a neutron absorber upper section and an aluminum or fuel follower lower section. It has an electric motor gear drive and has scram capability.

O. Transient Control Rod is a rod having a neutron absorber upper section and an aluminum or air followed lower section. It can be driven or ejected from the core by an electromechanical drive and has scram capability.

I GENERAL

The Diamond Ordnance Radiation Facility (DORF) is owned and operated by the Harry Diamond Laboratories (HDL). The Radiation Facility utilizes a TRIGA Mark F Reactor as the principal research tool in the study of the effects of neutron and gamma radiation on electrical and electronic components. Various radiation effects can be studied with complete safety under controlled laboratory conditions.

The DORF TRIGA Mark F Reactor was designed and built by Gulf General Atomic, San Diego, California. It is an inherently safe reactor designed for both steady state and pulsed operation. It has the design capability of:

(1) Steady-state operation of 100 kW.

(2) Square-wave operation of 1000 kW for a maximum of one hour per day. change

(3) Pulsed operation resulting in a prompt energy release of up to 40 MW-seconds per pulse. The corresponding peak power is 5000 MW with a pulse width of 7.0 msec. at half maximum. change

SITING

(a) The DORF is located within the metropolitan area of Washington, D.C., at the Forest Glen Section of the Walter Reed Army Medical Center (WRAMC), which is eight miles from the center of Washington, D.C., and approximately two miles south of Kensington, Maryland. The Foreset Glen site is an area of approximately 190 acres of rolling, partially wooded and cleared areas, on which are located both research laboratory facilities and hospital facilities for patients. The site is located in a commercial and residential area.

(b) The reactor is located near the southern border of the Forest Glen Section (FGS) about 600 feet from the nearest research laboratories and about 500 feet northwest of Brookville Road, which bounds the property on the southeast. The 4.2 acre DORF site is surrounded by woods, except to the northeast, where a large open field separates it from the research laboratories. Change

The reactor building is encircled by an exclusion fence with a radius of approximately 240 feet. Access to the exclusion area is controlled at the single entrance gate.

BUILDING

The building containing the DORF TRIGA is 65 ft. by 50 ft. and 25 ft. high. It has been designed to operate at a slight negative pressure. The ~~air-conditioning system~~ ventilation system exhausts all air from the reactor building through absolute filters and out a stack 45 feet above ground level.

REACTOR

(a) The core assembly is located at the bottom of a 15,000 gallon aluminum reactor tank which is approximately 13 feet in diameter and 19.5 feet deep. The core assembly is suspended by a support structure from a motor-driven carriage. The carriage is mounted on rails at the top of the tank and is capable of traversing the tank. A pair of 10-ton lead shield doors are located in the tank to allow access to an exposure room while the reactor is operating at the opposite end of the pool.

(b) The reactor core forms a compact cylinder and consists of a lattice of about 85 cylindrical fuel-moderator elements (maximum 87 elements), four control rods and a neutron source holder. Core components are contained between the upper and lower aluminum grid plates and surrounded by an aluminum shroud, 19 inches in diameter, which supports the grid plates. Holes approximately 1.5 inches in diameter in the upper grid plate space the fuel elements and control rods and allow water to flow out of the core and through the plate. Holes in the lower grid plate permit coolant passage through the plate and receive the fuel element lower end fixtures.

(c) The DORF-TRIGA uses solid fuel elements, developed by Gulf General Atomic, in which a zirconium hydride moderator is homogeneously combined with enriched uranium fuel. The active part of each element consists of a cylindrical rod of uranium-zirconium hydride containing 8 weight % of uranium enriched to 20 % in 235 uranium. The hydrogen-zirconium ratio is about 1.7. A 0.25 inch diameter zirconium hydride rod runs through the center of the active section. Graphite slugs act as top and bottom neutron reflectors. The active region and graphite slugs are clad with 0.02 inch stainless steel. The overall dimensions of the fuel element are 28.37 inches long by 1.47 inches in diameter. In addition to the standard fuel moderator elements, thermocouple instrumented elements are used to provide fuel temperature measurements and scrams.

(d) In generating a pulse, the DORF reactor period may be as short as 2 milliseconds. The electro-mechanical scram system is too slow to terminate this excursion. The shutdown results from the large prompt negative temperature coefficient of reactivity, an inherent property of the uranium-zirconium hydride fuel. This coefficient arises primarily from the hardening of the neutron spectrum associated with the heating of the uranium-zirconium hydride. Other contributions to the shutdown are the Doppler broadening of the U-238 absorption resonances and the expulsion of water from the core due to thermal expansion of the fuel element.

A more detailed description of the reactor and the facility is given in the Hazard Summary Report and Special Safety Analysis Reports.

II SAFETY LIMIT: MAXIMUM FUEL TEMPERATURE

A. FUEL TEMPERATURE

APPLICABILITY

This specification applies to the maximum temperature of the fuel attained during all modes of operation.

OBJECTIVE

To prevent undue release of radioactivity or risk to personnel.

SPECIFICATION

The safety limit for both the pulse and steady state modes of operation is that the fuel temperature at the hottest point shall not exceed 1050°C. The maximum temperature shall be determined by multiplying the measured temperature by the calculated peak-to-measured temperature ratio.

DESIGN BASIS (1)

Fuel elements in the DORF reactor core are stainless steel clad $U\text{-}ZrH_{1.7}$. The uranium loading (enriched to 20% U 235) in the fuel is between 8 and 8.5 wt-%. Having a H:Zr ratio approximately 1.7 is advantageous because these higher-hydride compositions are single phase beyond 1050°C and are not subject to phase separation on thermal cycling. Lower hydrides, which are used in many lower power TRIGA cores, have a substantial volume change associated with phase transformations at approximately 530°C. This phase and volume change coupled with temperature gradients through the fuel element can cause distortion or bending in fuel elements if the lower hydride is used at temperatures above

(1) GGA-7882 Kinetic Behavior of TRIGA Reactors, G.B. West, et al, (March 1967)

approximately 530°C. Hydrides of 1.65 to 1.7 are substantially removed from the phase boundary at approximately 1.5 (61.4 atom percent) and were chosen for the original high performance TRIGA elements to assure single phase composition even under extreme temperature gradient and heating rate conditions.

A potential limitation on the maximum fuel temperature is the possible diffusion of hydrogen from the fuel with a resulting pressure within microscopic voids in the fuel body and in the gap in the fuel can. However, in recent experiments at Gulf General Atomic (GGA), measurements have been made of the internal pressure in the fuel can for pulses yielding a local peak temperature of approximately 1100°C (calculated). The resulting rise was approximately 24 psi compared with 575 psi predicted from the equilibrium steady state data. Out of core (laboratory) experiments on U-$ZrH_{1.7}$ fuel sections have confirmed the lower hydrogen pressures for transient temperatures. They further show that the transient hydrogen pressure measured in the fuel can is a function of heating rate and that with high heating rates, pressures much reduced from the equilibrium values are obtained. Upon cooling of the alloy rehydriding will occur so that no residual hydrogen is left in the fuel clad gap.

To demonstrate the high temperature capability of the TRIGA element GGA has pulsed the ATPR (Advanced TRIGA Prototype Reactor) to peak fuel temperatures of 1050°C. The ATPR, fueled with elements of the same type used in the DORF TRIGA, has been pulsed to $5 reactivity insertions where a 100 element core released an energy of 54 MW sec., achieved a peak power of 8400MW, a pulse width of 5.5 msec. and a peak fuel temperature of 1050°C. Since the energy release in a pulse is a function of the prompt reactivity insertion, the $5 experiments in the ATPR have increased the tested performance of this system to a factor of about 1.5 over $4 pulse characteristics of the DORF TRIGA.

	CORE TEMPERATURE AT HOTTEST POINT
SAFETY LIMIT	1050 °C
	350 °C
LIMITING SAFETY SYSTEM SETTING	700 °C
	415 °C
MAXIMUM AVAILABLE PULSE (2.8% $\Delta k/k$)	$\approx$585 °C
1 MEGAWATT STEADY STATE	$\approx$430 °C
ROUTINE PULSE (2.1% $\Delta k/k$)	$\approx$385 °C 2

FIGURE 1. DORF REACTOR OPERATING LEVEL LIMITATIONS

III LIMITING SAFETY SYSTEM SETTINGS

III-1 PULSE OPERATION

APPLICABILITY

This specification applies to the maximum fuel temperature and peak power attained during pulse operation.

OBJECTIVE

To assure that an automatic protective action is initiated before a safety limit is exceeded.

SPECIFICATION

1. The maximum measured fuel temperature shall not exceed 700°C.

2. The maximum pulse peak power shall not exceed 5,500 MW. (110% of full scale).

DESIGN BASIS

With an aluminum follower on the A-ring control rod a measured temperature of 700°C corresponds to a calculated peak temperature of 890°C which is well below the safety limit.(1) This setting is based on a maximum measured fuel temperature of 600°C for a $4.00 reactivity insertion reported by Gulf General Atomic (GGA) and a maximum pool temperature of 60°C for the DORF reactor (III-2).

The calculated peak power for a $4.00 step reactivity insertion is 4900 MW. For a $5.00 reactivity insertion at GGA the measured peak power was 8400 MW and the peak temperature was 1050°C.(2) Therefore, a 110% of full scale limiting peak power setting of 5,500 MW corresponding to a fuel temperature of 650°C is conservative and safe.

The pulse limiting safety systems settings are therefore dualized for both peak fuel temperatures and peak power.

(1) GGA-7882 Kinetic Behavior of TRIGA Reactors, (March 1967).
(2) GGA-6489 Stability Tests on Stainless Clad TRIGA Fuel Elements Subjected to High Reactivity Insertions, (July 1965).

III-2 STEADY STATE OPERATION

APPLICABILITY

This specification applies to the maximum reactor power attained during steady state operation.

OBJECTIVE

To assure that the limiting safety system settings are not exceeded and that automatic protective action is initiated before the safety limit is reached.

SPECIFICATION

1.1%   1.1 MW ?   1.2 %   1.2 MW

1. The measured value of steady state power shall not exceed 110% ± of full power of 1 megawatt.

2. The integrated power at steady state power shall not exceed 1 megawatt hour per day.

BASIS

Selection of the limiting safety system setting for steady state operation is based upon operation of other TRIGA type reactors at both GGA and other facilities which use the same type fuel elements as DORF. Experience at GGA has shown that for reactor power levels up to 1.5 megawatts the maximum fuel temperature is 510°C and is well below the safety limits (1). At the DORF reactor the maximum steady state power is 1 megawatt and the corresponding fuel temperature is about 430°C.

(1) GGA-5786 Research in Improved TRIGA Reactor Performance, (October, 1964).

A steady state power limitation (scram) of 110% of full power is considered a reasonable limitation.

A loss of coolant calculation [2] made for the high hydride stainless steel clad element shows that the fuel temperature reaches a maximum of 780°C and the internal stress on the cladding would be 5070 psi if the loss of coolant occurs after an infinite run at 1 MW. The yield stress of 304 stainless steel at 780°C is 12,300 psi and the ultimate strength is 24,800 psi. Consequently, it is concluded that loss of coolant after prolonged operation at 1 MW will not result in the rupture of the fuel cladding.

If the operation is limited to one hour at 1 MW then the pool temperature will reach a maximum of 40°C or 104°F and it will take about ten hours to cool the pool to room temperature with the existing 100 KW heat exchanger.

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(2) Safety Analysis Report, Part 8, U. S. Geological Survey Reactor, Denver, Colorado.

IV MINIMUM CONDITIONS FOR OPERATION

IV-1 REACTOR INSTRUMENTATION

APPLICABILITY

This specification applies to steady state, square wave and pulse modes of reactor operation.

OBJECTIVE

To specify the minimum reactor instrumentation and associated scrams which must be operational in order to monitor and control reactor core power.

SPECIFICATION

The reactor shall not be operated without the following instrumentation being operable.

a. Modes I, IA, and II--Steady State (manual and automatic) and Square Wave.

	Measuring Channel	No.	Scrams
Power Level	Ion Chambers		
0-100 kW		2	1
0-1000 kW		2	1
Fuel Temperature	Fuel Element Thermocouples	2	2

b. Mode III--Transient Operation

	Measuring Channel	No.	Scrams
Power Level	Ion Chamber		
0-5000 MW		1	1
Fuel Temperature	Fuel Element Thermocouples	2	2

DESIGN BASIS

The linear power and log power measuring channels are required for the establishment of delayed critical during pulse preparation procedure. The linear and log power level channels are used from subcritical source level to the operating level in steady state and square wave power operation.

A transient (pulse) operation channel which uses a separate pulse detector for monitoring pulse peak and integrated power is required.

Two temperature measuring channels which monitor the core fuel temperatures during steady state, square wave and pulse modes of operation are required.

The scrams are the automatic protective devices which limit the reactor power and consequently the reactor fuel temperature. For steady state, square wave and pulse modes of operation the initiation of a scram will drop all control rods into the core.

The thermocouple response is slower than the ion chamber response and the core temperature reflects the integral power rather than instantaneous power. For these reasons the fuel temperature scrams are not used to initiate scrams during pulse operation and the scram limits are set above the maximum measured temperature expected for the largest pulse. After a pulse, the reactor is scrammed by either an automatic timer or manually.

One power level channel is required to provide a scram which is set at 110% of each range of operation in steady state, or square wave and pulse mode. In each case all control rods are dropped into the core upon initiation of a power level scram.

IV MINIMUM CONDITIONS FOR OPERATIONS

IV-2 LIMITING CORE OPERATING CONDITIONS AND PARAMETERS

APPLICABILITY

This specification applies to the core conditions and parameters during reactor operation.

OBJECTIVE

To assure that the reactor will be operated with core conditions and parameters within the bounds used to establish safety limits.

SPECIFICATION

1. The reactor core shall consist of standard TRIGA Mark F reactor fuel elements and a minimum of two(2) thermocouple instrumented TRIGA Mark F reactor fuel elements. The fuel shall consist of an alloy of uranium zirconium hydride, 8.5% (nominal) uranium enriched to 20% U235 and a Zr:H of 1.7 (nominal). The fuel elements shall be clad with 0.02 inches of stainless steel and be placed in a close packed cylindrical array.

2. The core shall contain three (3) standard control rods and one (1) transient control rod.

3. The reactor core shall be loaded so that the excess reactivity above cold critical does not exceed $6.25 in infinite water.

4. The shutdown margin shall not be less than 50 cents with the most reactive rod stuck out of core. Why

5. The core shall be cooled by natural convective water flow. why

6. The reactor shall not be operated if the maximum fuel temperature as measured with a standard thermocouple instrumented fuel element in the B ring of the core, exceeds 700°C. why

7. The reactor shall not be operated with the bulk reactor water temperature in excess of 60°C. why

8. When the initial daily reactor startup is accomplished from a power level less than 0.1 watt, neutron source level multiplication shall be indicated on the linear channel on withdrawal of the first control rod. why

9. The time from the initiation of the scram signal to full insertion shall not exceed one (1) second for the standard control rods and two (2) seconds for the transient control rod.

BASIS

The TRIGA Mark F type elements have demonstrated to be safe and reliable fuel elements in research reactors for many years. The safety limits are based on this type of fuel element.

Experience at DORF has demonstrated the reliability of the standard and transient control rod system. No control rod has ever stuck in any position. However, if the most reactive control rod becomes stuck in the fully withdrawn position, the necessary shutdown margin will be available in the other ocntrol rods.

The maximum allowed excess reactivity of $6.25 provides sufficient reactivity to accommodate fuel burn-up, xenon and samarium poisoning, experiments and control requirements. With $6.25 excess reactivity the reactor can be brought to critical with the transient control rod down and the core positioned against the exposure room tank wall.

A 50 cent shutdown margin is adequate for shutdown of the reactor in the event that the most reactive rod does not fall back into the core.

Holes in the bottom grid plate provide sufficient coolant flow through the core and past the fuel element triflute and through the upper grid plate.

The expected measured fuel temperatures for the maximum allowable pulse of $4.00 is about 600°C above ambient pool temperature. The fuel temperature scrams are not normally used to initiate scrams during pulse operation. Therefore, the temperature scram limits are set above the maximum measured temperature expected for the largest pulse reactivity insertion.

Experiments at Gulf General Atomic(1) indicate that for a steady state power of 1.5 MW and an ambient pool water temperature of 60°C the maximum water temperature at the top grid plate could not exceed about 90°C. Therefore, no bulk boiling would occur under these conditions. Since the maximum steady state power of the DORF reactor is only 1 megawatt, the 60°C pool water temperature is a conservative temperature limit.

The linear channel measurements provide adequate assurance of neutron source level subcritical multiplication during daily intitial reactor check-out and reactor start-up.

The rod drop or scram is not the primary shutdown mechanism for the TRIGA type fuel because of the prompt negative temperature coefficient of reactivity. Therefore, the scram time is not critical to the safe operation of the reactor. The scram time is measured to assure that the rods drop and that there is no restriction which may increase the ejection time of the transient control rod.

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(1) Public Document File, Docket 50-163, License R-67, Letter to AEC-Reactor Operations from Gulf General Atomic, dated 25 July 1967.

IV MINIMUM CONDITIONS FOR OPERATION

IV-3 REACTOR AND FACILITY SAFETY INTERLOCK SYSTEM

APPLICABILITY

This specification applies to reactor interlocks which must be satisfied during reactor operations.

OBJECTIVE

To assure that the reactor will be operated within the bounds of approved written procedures and to assure against the radiation exposure of operating personnel.

SPECIFICATION

1. Reactor operation control shall be restricted by the reactor safety interlock system if any of the following conditions exist.

	Modes in Which Effective		
Action Prevented	Steady State	Square Wave	Pulse
a. Control rod withdrawal with less than 10% full scale neutron source multiplication on the two most sensitive linear channel ranges.	x	NA	NA*
b. Simultaneous manual withdrawal of two or more standard control rods.	x	NA	NA

* Not Applicable

Action Prevented	Modes in which Effective		
	Steady State	Square Wave	Pulses
c. Initiation of a pulse from below 1 watt or above 1 kW steady state power or on a period of less than 40 seconds.	NA	x	x
d. Application of air to transient control rod unless drive cylinder is fully down.	x		
e. Withdrawal of any standard control rod.		x	x

2. The Facility Interlock system shall be provided such that:

a. the reactor cannot be operated unless the lead shielding doors within the reactor pool are either fully opened or fully closed.

b. the exposure room plug door cannot be opened unless the lead shield doors in the reactor pool are fully closed and are between the reactor core and the exposure room.

c. the reactor core cannot be moved from one end of the pool to the opposite end unless the lead shield doors are fully open.

d. the lead shield doors in the reactor pool cannot be opened unless the exposure room plug door is closed and a warning horn in the exposure room has sounded.

BASIS

The reactor standard control rod drive interlock system is designed to prevent standard control rod movement unless sufficient neutron source multiplication is available.

Although not a safety restriction, the manual withdrawal of not more than one standard control rod restricts the rate of reactivity insertion during steady state operations.

In square wave mode of operation, the regulating standard control rod will automatically ramp-out of the core after the transient control rod drive has received the initial drive down signal after being ejected from the core.

The transient drive cylinder interlocks are intended to prevent an inadvertant pulse on top of steady state power levels. In preparing for square wave and pulse operation, the power and period interlocks are to assure the reactor is near critical at a low power level.

The restriction of withdrawal of a standard control rod in pulse mode or square wave is to prevent a change in the reactor critical power level after switching from steady state mode and during the time of adjustment of the transient rod cylinder prior to initiation of a pulse or square wave.

The Facility Interlock System is designed to prevent the exposure of individuals who may be preparing experiments in the exposure room or adjacent areas. The exposure room door must be closed before the reactor can be moved to the exposure room end of the pool. Administrative procedures require the inspection of the exposure room by two persons prior to closing the plug door.

IV-4 RADIATION MONITORING EQUIPMENT

APPLICABILITY

This specification applies to the availability of ionizing radiation monitoring equipment in proper operational condition.

OBJECTIVE

To assure that radiation monitoring equipment is available for evaluation of radiation hazards to operating personnel and to the public.

SPECIFICATION

1. There shall be two fixed area monitoring instruments operational in the reactor building during all periods of reactor operation.

2. An air particulate monitoring instrument continuously sampling air above reactor pool shall be operational and when in an alarmed state capable of automatically closing building isolation dampers.

3. A gas stack monitoring instrument continuously sampling air exhausted from the facility stack shall be operational and when in an alarmed state capable of automatically closing the building isolation dampers.

BASIS

It is necessary that a continued evaluation of the radiation levels within the Facility Building be made to assure the safety of personnel. This is accomplished by the remote area monitoring system of the type described in the HSR.

The air particulate monitor (APM) system is necessary for the detection of possible radioactive particulates within the facility building and is also used as the primary instrument for the detection of fission product activity.

The air-conditioning system exhausts the air from the reactor building through absolute filters and out the stack. Since a major portion of the air exhausted from the facility building passes through the exposure room where it may become activated, a gas stack monitor is necessary to measure the activity released to the atmosphere. The evaluation of this activity assures that these releases are in accordance with the standards of protection prescribed by 10CFR20.

When alarmed, the gas and particulate monitors supply signals to close the building isolation dampers to prevent the escape of radioactive effluents to the atmosphere.

V SURVEILLANCE REQUIREMENTS

V-1 REACTOR INSTRUMENTATION AND SAFETY SYSTEMS

APPLICABILITY

This specification applies to the requirement for surveillance of reactor instrumentation.

OBJECTIVE

To insure that the reactor instrumentation which is required for safe operation is operable.

SPECIFICATION

The following measuring channels in the reactor instrument and safety systems shall be checked for proper operation before the first steady state or pulse operation of the day:

a. Linear channel
b. Log N channel
c. 110% power level scram
d. Temperature level scrams
e. Safety interlock system

BASIS

The operational capability of reactor instrumentation and safety systems must be verified prior to their use in reactor operation to insure proper control and safety of operation. A daily instrument checklist is used to establish that all instrumentation channels are operable and important scrams and interlocks have been tested prior to reactor start-up. The daily instrument checklist is augmented by a weekly instrument checklist. The checklists and procedures discussed in the Operating Procedures Manual (HDL 70-5) will be used.

V SURVEILLANCE REQUIREMENTS

V-2 RADIATION MONITORING EQUIPMENT

APPLICABILITY

This specification applies to the surveillance of radiation monitoring systems.

OBJECTIVE

To assure that the minimum required number of radiation monitors are maintained in operable condition.

SPECIFICATION

1. Fixed area radiation monitors shall be tested daily for operability and calibrated at intervals not exceeding 6 months.

2. The air particulate monitor calibration shall be checked with a radioactive source weekly. The monitor shall be checked for operability daily.

3. The gas stack monitor calibration shall be checked with a radioactive source weekly. The monitor shall be checked for operability daily. A record of all releases of radioactivity to the atmosphere shall be kept.

4. The closure of building isolation dampers in the air conditioning system by air particulate and gas stack monitoring systems shall be checked daily.

BASIS

Experience at DORF has indicated that a 6 month calibration of the fixed area monitoring system is sufficient to provide reasonable assurance of the accuracy of the system. Although two fixed area monitors are the minimum requirement, six detectors are located throughout the facility building. The fixed area monitors are backed up by portable radiation survey instruments.

The air particulate monitor has been in operation since 1961 and experience has demonstrated that weekly check of the calibration with a radioacive source assures that the system is operating and the system sensitivity is maintained.

The gas stack monitor calibration is checked with a check source at weekly intervals. The calibration of the system has been related to a primary Argon 41 calibration. The primary calibration will be performed as deemed necessary by the Physicist-in-Charge.

V-3 REACTOR FUEL ELEMENT MEASUREMENTS

APPLICABILITY

This specification applies to the requirements for surveillance of reactor fuel elements.

OBJECTIVE

To assure that the integrity of the fuel elements is maintained.

SPECIFICATION

1. The fuel elements shall be measured for elongation and traverse bend after every 500 pulses, or at least once annually.

2. The fuel element elongation shall not exceed 0.1 inch; the traverse bend over the element cylindrical section shall not exceed 1/16 inch.

3. The fuel follower section of the control rods shall not exceed 1/16 inch traverse bend and shall be measured after every 500 pulses or at least once annually.

BASIS

The results of a fuel stability test program at Gulf General Atomic[1] indicate that the stainless steel clad U-Zr $H_{1.7}$ TRIGA fuel elements maintain good dimensional stability when subjected to step reactivity insertions up to $4.70. Experience at DORF indicates no appreciable elongation or bend

(1) C.O. Coffer, et al. Stability Tests on Stainless Clad TRIGA Fuel Elements Subjected to Large Reactivity Insertions, GGA-6489 (July 1965).

for reactivity insertions up to \$3.00 in over 2800 pulses. A 500 pulse measurement criteria is considered a reasonable interval for surveillance. The fuel follower control rod traverse bend criteria is based upon the bend limitations for the fuel elements.

V-4 AREA WARNING SYSTEMS

APPLICABILITY

This specification applies to the requirement for surveillance of area warning systems.

OBJECTIVE

To assure that the area warning system required for reactor operations is operable.

SPECIFICATION

1. The evacuation siren shall be checked weekly for operability.

2. The coincident radiation alarm to WRAMC-Forest Glen Military Police Station shall be checked weekly for operability.

3. The air particulate and stack monitor warning horn and indicator lights shall be checked for operability daily.

DESIGN BASIS

The evacuation siren is used to evacuate personnel from the facility in the event of a hazard or emergency which may affect their health and safety. Weekly operational check of this item is considered a reasonable interval for surveillance.

The coincident alarm system notifies the Military Police at the Forest Glen Section of WRAMC of a radiation release at the facility. Two out of three radiation monitors in the coincident system must be in an alarmed state to actuate the coincident alarm. A weekly operational check is considered to be a reasonable surveillance interval.

A daily operational alarm check of the air particulate and stack gas monitors provide a reasonable check of these items.

V-5 CONTROL RODS

APPLICABILITY

This specification applies to the surveillance requirements for the reactor control rods.

OBJECTIVE

The objective is to assure the integrity of the control rods.

SPECIFICATION

1. The reactivity worth of each control rod shall be determined annually with the core in infinite water reflector and at a position adjacent to the tank wall at the exposure room end of the pool.

2. Control rod scram times shall be determined with a stop watch weekly and by electronic measurement annually.

3. The control rods shall be visually inspected at intervals not exceeding one year.

DESIGN BASIS

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available, to provide a means for determining the step reactivity insertions in pulse mode, and to determine the reactivity worth of materials placed in the reactor environment.

Experience since 1961 has indicated that stop watch times are acceptable for routine measurement of control rod drop times. A more accurate electronic measurement is performed annually.

A visual inspection of the control rods and acceptable rod drop times are sufficient for confidence of proper operation.

V-6 OPERATOR PERSONNEL

APPLICABILITY

This specification applies to the certification of personnel required to operate the reactor.

OBJECTIVE

To assure that the operators have appropriate training and that all certified operators maintain proficiency.

SPECIFICATION

The DORF Qualifications Board shall review the qualifications of all reactor operating personnel and shall recommend approval or disapproval for assignment as a reactor operator.

DESIGN BASIS

The DORF Reactor Operator Qualifications Board is responsible for review of the training and qualification of reactor operators as specified in AR 385-80.

VII ADMINISTRATIVE CONTROLS

A. ADMINISTRATION

The following are the key administrative and organizational elements established for the protection of health and for the safe operation of the DORF (In accordance with AR 385-80, Nuclear Reactor Systems Health and Safety).

1. Organization

a. Operation

The direct line of responsibility and authority for reactor operation and the protection of the health and safety of both the public and on-site personnel is as follows:

Commanding Officer, Harry Diamond Laboratories (HDL)
Technical Director, Harry Diamond Laboratories (HDL)
Chief, Systems Research Laboratory (HDL)
Chief, Radiation Facilities Branch (HDL)
Physicist-in-Charge, DORF
Assistant Physicist-in-Charge, DORF
Reactor Operator

b. Health Physics

A Host-Tenant Agreement has been established between the Commanding Officer of HDL and the Commanding General of Walter Reed Army Medical Center (WRAMC) wherein, in part, the Health Physics activities for DORF are supplied by WRAMC Health Physics Office.

The Commanding General, WRAMC is responsible for all aspects of Health Physics, both on-site and off-site, at the DORF. The Health Physics Officer, WRAMC assigns Health Physics Personnel to provide coverage at the reactor site. These personnel are under direct control of the WRAMC Health Physics Officer. The Health Physics Officer has final and absolute authority with regard to matters pertaining to radiological safety at DORF.

c. Reactor Safeguards Committee

The Reactor Safeguards Committee (RSC) is appointed by and reports to the Commanding Officer. The Committee shall consist of personnel who collecively provide a broad spectrum of experience in reactor technology. The RSC serves as an independent committee to review all matters pertaining to reactor safety, operating procedures, reactor modifications and proposed tests and experiments.

d. Reactor Test Planning Committee

The Reactor Test Planning Committee (RTPC) is appointed by the Commanding Officer to review radiation tests to be performed at the Diamond Ordnance Radiation Facility. The Committee has the responsibility insure the desirability, validity and the use of proper radiation test procedures of experiments.

e. Physicist-in-Charge

The Physicist-in-Charge (PIC) of the Diamond Ordnance Radiation Facility has the direct and immediate responsibility for the facility, its operation and safety.

f. Chief, Radiation Facilities Branch

The Chief, Radiation Facilities Branch, has the ultimate responsibility for the safe, competent and efficient operation and use of the facility. He is responsible for the approval or disapproval of proposed radiation exposure experiments.

2. Records and Reports

a. Written records including log books are maintained on reactor operations, reactor maintenance, reactor modifications, and radiation exposures of personnel.

b. Written summaries of review and recommendations by the RSC and the RTPC are maintained.

c. A status report summarizing DORF operations will be issued at least annually and submitted to the Chief of Engineers.

d. Accidents and incidents are reported as required in AR 385-40.

e. Radiation exposures of personnel are reported as required in AR 40-14 (Occupational Exposure to Ionizing Radiation).

B. ACTION IN RESPONSE TO VIOLATION OF SAFETY LIMIT

The following criteria is established to define action in the event that the safety limit in Section II is violated.

Whenever it is determined that the safety limit has been exceeded, the reactor shall be shut down and a report submitted through channels to the Chief of Engineers in accordance with AR 385-40. Resumption of operations requires Chief of Engineers approval.

C. ACTION IN RESPONSE TO VIOLATION OF LIMITING SAFETY SYSTEM SETTINGS

The following criteria are established to define action in the event that any of the limiting safety system settings given in Section III are violated.

Whenever it is determined that a limiting safety system setting has been exceeded, an evaluation shall be made by the Physicist-in-Charge, who shall submit a report of the violation and proposed corrective action to the Chief of Radiation Facilities Branch and the Reactor Safeguards Committee (RSC) for review. The Chief of the Radiation Facilities Branch will authorize corrective action and report through channels to Chief of Engineers as required by AR 385-40 and AR 385-80.

D. PROCEDURES

The operations listed below shall be conducted in accordance with approved written administrative and operating procedures:

(a) Normal startup, operations and shutdown of the complete facility and all systems and components involving nuclear safety of the facility.

(b) Actions to be taken to cope with emergency conditions involving possible or actual release of radioactive materials, including provisions for evacuation of the facility.

(c) Preventive or corrective maintenance operations.

E. EXPERIMENT REVIEW

A radiation experiment requires completion of a Proposal for Radiation Experiment Form by the experimenter with a written description of the experiment plan. All radiation experiments require a review by the Reactor Test Planning Committee (RTPC), the Reactor Safeguards Committee (RSC), and the reactor Physicist-in-Charge (PIC). Approval for the performance of an experiment is the responsibility of the Chief of the Radiation Facilities Branch.

F. MODIFICATIONS

Modifications to the DORF reactor must be performed in accordance with approved procedures. These procedures must as a minimum have the approval of the Reactor Safeguards Committee, the Chief, Radiation Facilities Branch, and the Physicist-in-Charge, DORF, and may require other approvals by high authority depending upon the nature of the modifications involved.

SAFETY EVALUATION REPORT

DIAMOND ORDNANCE RADIATION FACILITY
TECHNICAL SPECIFICATIONS

BY THE

ARMY REACTOR SYSTEMS HEALTH
AND SAFETY REVIEW COMMITTEE
(ARCHS)

NOVEMBER 1971

SAFETY OFFICE
OFFICE OF THE CHIEF OF ENGINEERS
DEPARTMENT OF THE ARMY
WASHINGTON, D. C. 20314

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SUMMARY

In accordance with provisions of Army Regulation 385-80[1], the Army Reactor Systems Health and Safety Review Committee (ARCHS) reviewed Technical Specifications prepared for the Diamond Ordnance Radiation Facility (DORF) reactor to define those significant design features, operating procedures, and operating limitations which were considered important to provide reasonable assurance that the facility would be operated without undue hazard to the public health and safety.

The DORF Technical Specifications were prepared in accordance with standards and guidance published by the Atomic Energy Commission in Part 50 of Title 10, Code of Federal Regulations. Also, the format and content of the SM-1A Technical Specifications were compared to Technical Specifications recently approved by the Atomic Energy Commission for the TRIGA reactor at the University of California at Irvine.

The ARCHS concluded that the DORF Technical Specifications provided reasonable assurance that the DORF reactor would be operated without undue hazard to the health and safety of the general public or operating crew and that they were consistent in format and content with Technical Specifications presently being approved by the Atomic Energy Commission.

It was recommended that the Chief of Engineers approve the DORF Technical Specifications dated 24 November 1971.

SECTION I

FACILITY DESCRIPTION

The Diamond Ordnance Radiation Facility (DORF) nuclear reactor is located within the metropolitan area of Washington, D. C. at the Forest Glen Section of the Walter Reed Army Medical Center. The DORF reactor is a TRIGA Mark F designed and built by General Atomic Division of General Dynamics Corporation (GGA). It has operational characteristics consistent with the other reactors of the same class.

The mission of the DORF is (1) to investigate and determine the susceptibility of electronic materiel to nuclear weapons radiation environment mechanisms and effects, and to determine ways and means of developing less susceptible materiel, (2) to conduct research and development into various physical sciences and engineering fields directed toward meeting the military characteristics for fuzing or related items, and (3) to provide a source of radiation for experiments aligned toward medical research and nuclear medicine.

SECTION II

BACKGROUND

For a number of years, reactors operated by the Army under provision of Section 91b of the Atomic Energy Act used either the Hazards Summary Report or the later Safety Analysis Report as the basic operating document. As a result of recent changes made by the U. S. Atomic Energy Commission in its rules and regulations constituting Title 10 of the Code of Federal Regulations,[2] it was decided that Technical Specifications, as established in Title 10, would be prepared for all Army nuclear reactor facilities.

The Harry Diamond Laboratories TRIGA Mark F research reactor was authorized under Section 91b and began operations in 1961. Since that time, it has operated under the original Hazards Summary Report[3] as changed by an Addendum[4,5] and three Special Safety Analysis Reports (SSAR)[6,7,8] which were reviewed and approved by the Army Reactor Systems Health and Safety Review Committee (ARCHS) for the Chief of Engineers.[9, 10, 11]

Based upon the above documentation, the Harry Diamond Laboratories submitted Technical Specifications to define those significant design features, operating procedures, and operating limitations which were considered important to provide reasonable assurance that the facility would be operated without undue hazard to the public health and safety. In preparing the DORF Technical Specifications, the Dorf staff used the guidance published by the AEC in the Code of Federal Regulations 10 CFR 50 and the format and content of recently approved Technical Specifications for a similar TRIGA reactor at the University of California at Irvine.[12]

The following paragraphs present the results of the review conducted by ARCHS to insure that the DORF Technical Specifications conformed to the above guidance and standards.

SECTION III

SAFETY EVALUATION

The Design Basis Accident (DBA) for the DORF was a defect in the cladding of a fuel element either prior to or simultaneously with a 2.2% $\Delta k/k$ transient. From operating experience with TRIGA reactors it was concluded that this DBA caused no significant hazard to individuals near the reactor. The Committee review was therefor organized to insure protection of the facility from an accident more severe than the DBA. Accordingly, it was necessary to insure fuel clad integrity and limit reactivity insertion.

Fuel Clad Integrity

Fuel clad integrity was assured by (1) defining damage criteria, (2) expressing the damage criteria in terms of measurable system variables (safety limits) and (3) providing administrative controls or safety systems which acted automatically to prevent damage.

The limiting damage criterion for the fuel clad was the possible diffusion of hydrogen from the fuel with a resulting pressure within microscopic voids in the fuel body and in the gap in the fuel can. In recent experiments at Gulf General Atomic (GGA), measurements were made of the internal pressure in the fuel can for pulses yielding a local peak temperature of approximately 1100°C (calculated). The resulting rise was approximately 24 psi compared with 575 psi predicted from the equilibrium steady state data. Out-of-core (laboratory) experiments on $U\text{-}ZrH_{1.7}$ fuel sections confirmed the lower hydrogen pressures for transient temperatures. They further showed that the transient hydrogen pressure measured in the fuel can was a function of heating rate and that with high heating rates, pressures much reduced from the equilibrium values were obtained. Upon cooling of the alloy, rehydriding occurred so that no residual hydrogen was left in the fuel gap.

GGA has pulsed the Advanced TRIGA Prototype Reactor (ATPR) to peak fuel temperatures of 1050°C. This temperature of 1050°C corresponded to the hottest point in a fuel element and was calculated from a measured temperature of 750°C. The ATPR, fueled with elements of the same type used in the DORF TRIGA, has been pulsed to $5.00 reactivity insertions where a 100 element core released an energy of 54 MW sec., achieved a peak power of 8400 MW, a pulse width of 5.5 msec. and a peak fuel temperature of 1050°C.

Harry Diamond Laboratories (HDL) selected a maximum fuel element temperature of 1000°C as the safety limit for the DORF reactor. The Committee concurred with the basis and conservatism of this selection.

It was noted that no safety limits were discussed with regard to the prevention of departure from nucleate boiling during steady state operation. The Committee considered this acceptable for three reasons. First, a maximum fuel temperature of 1000°C with a stainless steel clad was a sufficient condition to preclude clad damage since the clad must necessarily be less than the threshhold of damage under this condition. Secondly, there were no other variables which affected DNB, (such as flow or pressure) over which the facility maintained variable control and for which limiting safety system settings could be specified with automatically acting scram devices. A minimum pool level was specified from other considerations although its overall effect on DNB was negligible for any reasonable submersion of the core. Thirdly, specifying only a temperature limit was consistent with AEC practice in the Technical Specifications for the TRIGA reactor at the University of California at Irvine.

In reality DNB ratios were very large (at least 7) for all DORF operations and bulk boiling was prevented by subsequent limiting conditions for operation. If steady state power levels at the DORF were not so low and safety margins correspondingly high, the Committee would want to reconsider its approach in this matter.

To assure that an automatic protective action was initiated before the safety limit was exceeded during pulse operation, HDL specified limiting safety system settings as follows:

1. The maximum fuel temperature, as measured with a standard thermocouple instrumented fuel element in the B-ring of the core, shall not exceed 500°C.

2. The maximum measured pulse peak power shall not exceed 2,200 MW (110% of full scale).

With an aluminum follower on the A-ring control rod, a measured temperature of 500°C corresponds to a calculated peak temperature of 650°C which was well below the safety limit[13]. In practice, reactivity insertions at DORF were limited to \$3.00 by a mechanical stop on the pulse rod. A reactivity insertion of \$3.00 resulted in a maximum, measured fuel temperature of 460°C and a corresponding, calculated peak fuel temperature of 580°C for a pool temperature of 25°C.

A \$5.00 reactivity insertion of GGA with identical fuel elements resulted in a measured peak power of 8400 MW and the measured peak temperature of 750°C.[13] The measured peak power for a \$3.00 step reactivity insertion was 1860 MW.[4] Therefore, a limiting peak power setting of 2,200 MW (110% of full scale), corresponding to a measured fuel temperature of about 500°C, was conservative and safe.

The pulse limiting safety systems settings were, therefore, dualized for both peak fuel temperatures and peak power.

The Committee concurred with the basis and conservatism of these limiting safety system settings for pulse operation.

To assure that automatic protective action was initiated before the safety limit was reached during steady state operation, HDL specified limiting safety system settings as follows:

1. The maximum measured fuel temperature during steady state operation shall not exceed 500°C.

2. The measured value of steady state power shall not exceed 110% of full power of 250 kilowatts.

3. The integrated power at steady state power shall not exceed 1 megawatt hour per day.

Selection of the limiting safety system setting for steady state operation was based upon operation of other TRIGA-type reactors at both GGA and other facilities which use the same type fuel elements as DORF. Experience at GGA has shown that for reactor power levels up to 1.5 megawatts, the maximum fuel temperature was 510°C, and was well below the safety limits.[14] At the DORF reactor the maximum steady state power was 250 kilowatts and the corresponding fuel temperature was about 240°C. Based on fuel temperature limitations, a steady state power of 110% of full power was conservative by approximately a factor of six.

An analysis[15] of the peak heat flux, where there is a departure from nucleate boiling (DNB) and a transition to film boiling, has been performed for other TRIGA reactors similar to the configuration, fuel elements, and core spacing of the DORF reactor. The analysis showed that the maximum channel heat flux for a DNB ratio of 1 and bulk pool temperature of 90°F (core inlet temperature) was 380,000 BTU/hr $ft.^2$ This heat flux corresponds to a maximum reactor power of 1800 KW for a 70 element core with a peak-to-average power density ratio of 2.0. Based on DNB ratio considerations, the 250 KW steady state power limitation for the DORF reactor was conservative by a factor of approximately seven.

The Committee concurred with the basis and conservatism of these limiting safety system settings for steady state operation.

HDL specified limiting conditions for operation which insured that for any mode of operation there were three automatic protective devices (two on fuel temperature and one on core power) any one of which could scram the reactor.

Reactivity Limitations

As discussed earlier a mechanical stop on the pulse rod limited reactivity insertions at the DORF to \$3.00. In addition limiting conditions for operation restricted core excess reactivity to less than \$5.00 in infinite water and required a minimum shutdown margin of 50 cents with the most reactive rod stuck out of the core and the pool temperature less than 60°C. Also experiment reactivity worth was limited to \$2.00 and experiments with a reactivity worth greater than \$1.00 were required to be securely fastened or properly located to prevent movements which could cause inadvertent reactivity changes during reactor operation.

Maximum rod drop times were specified and a reactor and facility safety interlock system was provided to assure that the reactor would be operated within the bounds of approved written procedures.

Appropriate surveillance requirements were specified to assure adherence to these limitations.

The Committee concluded that potential reactivity changes were adequately controlled and that there was reasonable assurance that a step reactivity greater than 2.2% $\Delta k/k$ would not be inserted into the DORF reactor core.

Additional Safety Considerations

The Committee considered the following additional areas of sufficient importance to warrant individual consideration:

Emergency Power

Ventilation

Fuel Storage

Environmental Monitoring

Facility Radiation Level Control

Organization

Review of Reactor Operations

Review of Experiments

Operating Procedures

Modification Procedures

Records

Reports

Quality Assurance

Compliance

Following is a summary of the Committee's review in each of these areas.

Emergency Power: Electrical power supplied by batteries was required to be available to operate emergency building lights and radiation monitoring equipment.

The equipment operated by the emergency electrical battery power was considered adequate to assure the detection of radiation levels, and to provide lighting so that the reactor core, including the control rod positions, could be monitored visually. Normally, two additional sources of power were available: (1) a 100 KW, diesel driven generator; (2) a gasoline powered, 5 KW, portable generator. In view of the adequacy of the battery operated emergency system, these units were not required to be operational to conduct reactor operations.

Ventilation: The reactor was not permitted to operate unless the facility ventilation system was in operation. The entire facility was air-conditioned. Normal air recirculation was used in the building, except in the exposure room and in the warm-storage room. All exhaust air from the exposure room and warm-storage room passed through absolute filters and was exhausted through a stack that extended approximately 45 feet above the ground level. The flow of air was controlled by maintaining a lower pressure within the exposure room and the warm-storage room than in the remainder of the building. The remainder of the building had, in turn, a slightly lower pressure than that found outside.

The ventilation system was required to be equipped with automatically closing dampers which isolated the air in the building from the outside environment when radioactivity was detected within the facility. Air particulate and gas stack monitoring systems activated alarms if undesirable quantities of radioactivity were detected within the building or in the stack. Concurrently with the alarms, control signals were provided to actuate positive sealing dampers in the ventilation ducts to isolate the building from the outside environment.

Fuel Storage: Arrangements were specified for fuel to be stored in quantities and geometrical arrays which precluded criticality and adequate cooling water was specified for irradiated fuel elements to prevent temperatures in excess of design limits.

Environmental Monitoring: Specifications required sampling stations and frequency of sampling to be in accordance with the latest Environmental Radiation Monitoring Plan approved by the Army Reactor Systems Health and Safety Review Committee.

Plant Radiation Level Control: Specifications required:

1. There shall be two fixed area monitoring instruments operating in the reactor building during all periods of reactor operation.

2. An air particulate monitoring instrument, continuously sampling air above the reactor pool shall be operating and, when in an alarmed state, capable of automatically closing isolation dampers.

3. A gas stack monitoring instrument continuously sampling air exhausted from the facility stack shall be operating and, when in an alarmed state, capable of automatically closing the building isolation dampers.

There were appropriate surveillance specifications to insure proper operation of these monitors and the isolation dampers.

Organization: The operational chain of command and support elements were established as follows:

Commanding Officer, Harry Diamond Laboratories (Responsible Command)

Technical Director, Harry Diamond Laboratories

Chief, Nuclear Radiation Effects Laboratory

Chief, Nuclear Vulnerability Branch (Reactor Commander)

Physicist-in-Charge, DORF

Reactor Operator

Two levels of independent safety review were provided (Reactor Safeguards Committee and the Army Reactor Systems Health and Safety Review Committee)

A reactor operator qualification board was established for the purpose of certification and re-certification of reactor operators. The qualification board was convened by the Reactor Commander. The criteria for reactor operator certification established in AR 385-80 were to be followed.

The Reactor Test Planning Committee (RTPC was appointed by the Commanding Officer, HDL, to review radiation tests to be performed at the facility. The Committee has the responsibility to insure the desirability, validity, and the use of proper radiation test procedures during experiments.

A Host-Tenant Agreement was established between the Commanding Officer of HDL and the Commanding General of Walter Reed Army Medical Center (WRAMC). Therein, it is specified that the Health Physics activities for DORF were to be supplied by WRAMC Health Physics Office. The Commanding General, WRAMC, was responsible for all aspects of Health Physics personnel to provide coverage at the reactor site. These personnel were under direct control of the WRAMC Health Physics Officer. The Health Physics Officer has final and absolute authority regarding matters pertaining to radiological safety at DORF.

Review of Reactor Operations:

The Reactor Safeguards Committee was required to meet periodically to review reactor operations. The scope of the Committee reviews were specified and included:

1. Review and approval of all proposed changes to the facility procedures and Technical Specifications.

2. Review of abnormal performance of facility equipment and operating anomalies.

3. Review of unusual or abnormal occurrences and incidents which are reportable under AR 385-40.

4. Meeting annually at the facility for the purpose of refreshing knowledge of operation and reviewing changes which have occurred.

5. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or change in the Technical Specifications.

6. Review and recommendation of approval for experiments utilizing the reactor facilities based on review criteria specified.

Review of Experiments:

All radiation experiments require a review by the Reactor Test Planning Committee (RTPC), the Reactor Safeguards Committee (RSC), and the reactor Physicist-in-Charge (PIC). Approval for the performance of an experiment was the responsibility of the Reactor Commander.

All experiments conducted at the DORF were to be described by a test plan. The test plan included:

1. The purpose of the experiment

2. A procedure for the performance of the experiment

3. The reactivity worth of the experiment

4. The integrity of the experiment, including the effects of changes in temperature, pressure, and chemical composition

5. Any physical or chemical interaction that could occur with the reactor components.

6. Any radiation hazard

Operating Procedures: Written reactor operating and administrative procedures were required for startup, operation, and shutdown of the reactor and all systems and components involving the nuclear safety of the facility, emergency situations, and modification and maintenance operations which could have an effect on reactor safety.

These procedures required as a minimum the approval of the Chief, Nuclear Vulnerability Branch.

Periodic emergency drills were required and special procedures were specified for interlock bypass and entry into the facility during non-operational hours.

Modification Procedures:

Modifications of the DORF reactor were to be performed in accordance with specific written procedures and accomplished in accordance with the provisions of AR 385-80. These procedures required as a minimum the approval of Chief, Nuclear Vulnerability Branch and require other approvals depending upon the nature of the modification involved.

Records:

The following records were required to be kept in a manner convenient for review by facility management and higher headquarters.

1. An operational log which included

   a. Pertinent data regarding system operation

   b. Operator actions

   c. Details of any abnormalities occurring

2. Records of maintenance activities which include:

   a. Routine maintenance and component replacement

   b. Equipment failures

   c. Replacement of principal items of equipment

3. Records of periodic checks, inspections, and/or calibration performed to verify that requirements specified under surveillance standards were being met. All equipment failing to meet surveillance requirements and the corrective action taken were to be included.

4. Routine facility radiation and contamination surveys

5. Environmental monitoring surveys

6. Radiation exposure for all facility personnel, including all visitors to the facility who enter controlled areas

7. Records of the concentration of radioactivity in liquid and airborne effluents released to the environment and solid waste shipped

8. Malfunction reports

9. Records of changes made to the facility as described in the safety analysis report

10. Records of reactor tests and measurements

11. Records of changes made in the operating procedures

12. Records of new and spent fuel inventory

13. Safety review of facility modifications, changes, tests, or experiments including recommendations by the RSC.

Reports:

The following reports were to be prepared and submitted for review in accordance with appropriate Army regulations.

1. An Annual Operating Report, as required by AR 385-80

2. Accidents and incidents as required in AR 385-40

3. Radiation exposures of personnel as required in AR 40-14

(Records were to be kept indefinitely)

Quality Assurance Program:

A quality assurance program was specified which established design and test controls and provided records that components, systems structures, operation and maintenance of the facility conformed to design specifications for the facility.

Compliance:

Under Department of Army regulations, the DORF was subject to an annual general inspection on health and safety by qualified representatives of the Department of the Army Inspector General.

An annual technical, health physics and safety review was also to be conducted by the Safety Office, AMC, as prescribed by USAMC Supplement 1 to AR 385-80.

SECTION IV

CONCLUSIONS

The Committee concluded that there was reasonable assurance that the DORF could be operated within the limitations of these Technical Specifications without endangering the health and safety of the general public or operating personnel. Specifically, it was noted that:

1. Operating experience with TRIGA reactors had demonstrated no significant hazards to individuals near the reactor when the reactor experienced a 2.2% $\Delta k/k$ transient with a defect in the cladding of a fuel element (the design basis accident for the DORF).

2. DORF Technical Specifications were organized to insure clad integrity and limit reactivity insertion to less than 2.2% $\Delta k/k$ for all operational conditions.

3. Fuel clad integrity was assured by (1) defining damage criteria, (2) expressing the damage criteria in terms of measurable system variables (safety limits) and (3) providing administrative controls or safety systems which acted automatically to prevent damage.

4. The safety limits established for pulse and steady state operation provided appropriate allowance for uncertainty in onset of damage.

5. Limiting safety system settings had been established which provided adequate safety margins during steady state and pulse operation. Reactor and Facility interlock systems and administrative

controls had been provided to prevent conceivable hazardous reactivity changes in those instances where automatically operating safety systems were not feasible.

6. In addition to providing protection for the barrier to the release of radioactive fission products, the Technical Specifications provided appropriate limiting specifications and surveillance requirements in the following areas which were considered to warrant special consideration:

Emergency Power

Ventilation

Fuel Storage

Environmental Monitoring

Facility Radiation Level Control

Organization

Review of Reactor Operations

Review of Experiments

Operating Procedures

Modification Procedures

Records

Reports

Quality Assurance

Compliance

7. The format and content of the DORF Technical Specifications were comparable to Technical Specifications recently approved by the

Atomic Energy Commission and other Technical Specifications being prepared for Army nuclear reactor facilities.

SECTION V

RECOMMENDATION

It is recommended that the Chief of Engineers approve the DORF Technical Specifications dated 24 November 1971.

REFERENCES

1. Army Regulation 385-80, Nuclear Reactor Health and Safety Program, 16 October 1970.

2. Code of Federal Regulations, Title 10, revised as of 1 January 1971

3. Hazard Summary Report (for the TRIGA Mark F), Harry Diamond Laboratories, October 1962.

4. Addendum, Hazards Summary Report, DORF TRIGA-MARK F Stainless Steel Clad Core, Harry Diamond Laboratories, 3 January 1967.

5. ARCHS Report 66-5, April 1966, Operations with Stainless Steel Clad Core.

6. Special Safety Analysis Report, Reactor Automatic Scram Timing System, Diamond Ordnance Radiation Facility, Harry Diamond Laboratories, 3 July 1969.

7. Special Safety Analysis Report,"Replacement of the DORF Reactor Poison Followed Transient Rod with a Transient Rod with an Aluminum Follower," Diamond Ordnance Radiation Facility, Harry Diamond Laboratories, 1 October 1969.

8. Special Safety Analysis Report, Proposed DORF Fuel Follower Control Rod Modification, Diamond Ordnance Radiation Facility, Harry Diamond Laboratories, 10 February 1971.

9. Memorandum from Chief, Safety Office, AMC to Commanding Officer, Harry Diamond Laboratories, Subject,Reactor Automatic Scram Timing System, dated 26 August 1969.

10. OCE Letter, ENGSO, 21 October 1969, Subject: Replacement of the DORF Poison-followed Transient Rod with a Transient Rod with an Aluminum Follower, with USAMC, AMCSF-N 1st Indorsement, 24 October 1969.

11. ARCHS Report 71-2, March 1971, Diamond Ordnance Radiation Facility Fuel-Follower Control Rods

12. Technical Specifications for the University of California at Irvine TRIGA Reactor, November 24, 1969.

13. GGA-7882 Kinetic Behavior of TRIGA Reactors (March 1967).

14. GGA-5786 Research in Improved TRIGA Reactor Performance, (October 1964).

15. Gulf General Atomic Report, GA-9638, Safeguards Summary Report for the New York Hall of Science TRIGA Mark II Reactor, (1 September 1969).

SAFETY EVALUATION REPORT

DIAMOND ORDNANCE RADIATION FACILITY

TECHNICAL SPECIFICATIONS

ARCHS COMMITTEE

James H. Edgerton
JAMES H. EDGERTON
Dir Trans., DCSLOG (MTMTS) Member

Jerry D. Griffith
JERRY D. GRIFFITH
Chairman, ARCHS

WILMER F. NOSER
OCE Member

Bobby R. Adcock
BOBBY R. ADCOCK
LTC, MSC
TSG Member

Homer D. Musselman
HOMER D. MUSSELMAN
OCE Alternate Member

Harold E. Wells
HAROLD E. WELLS
AMC Member

DORF POOL WATER

Report from Rockwell International (Frank Badger) concerning the DORF Pool water results from Teledyne Isotopes

Gross β : < 1.0 pico Curies/liter
^3H : 1.41×10^{-3} pico Curies/liter $\pm$ 0.13

HEALTH PHYSICS
WALTER REED ARMY MEDICAL CENTER
Washington, D. C. 20012

HSWP-QHP
*SOP Number 3-D-1

7 March 1979

HEALTH PHYSICS SOP FOR DIAMOND ORDNANCE RADIATION FACILITY

1. Purpose: To establish routine Health Physics procedures and to provide guidance to the DORF staff in the implementation of the Health Physics Program at DORF.

2. Radiation Control Areas. Radiation Control Areas are as described in WRAMC Regulation 40-10. The entire area located within the DORF perimeter fence is a restricted area.

3. Personnel Monitoring

a. Personnel assigned to DORF will be placed on film badge service in accordance with AR 40-14.

b. Personnel entering radiation areas will be placed on film badge service in accordance with AR 40-14.

c. Visitor personnel not entering radiation areas and accompanied by DORF personnel at all times need not be placed on film badge service.

d. Personnel placed on film badge service will wear pocket dosimeters. Dosimeters will be read as required, at least daily and results monitored on the dosimeter log.

4. Radiation Protection Standards

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* This SOP supersedes HP SOP's 3-D-1 thru 3-D-8, dated March 1975

a. Personnel exposure to radiation, permissible levels of radiation permissible levels of contamination will be in accordance with WRAMC NRC License requirements.

b. Radioactive waste disposal procedures will be in accordance with current WRAMC NRC License requirements.

c. Release of liquid radioactive waste from detention tanks will be in accordance with current WRAMC NRC License regulations and DORF ENRADMON Plan.

5. Radiation Work Permits (RWP)

a. Radiation work permits will be obtained prior to performing any of the following procedures.

1. Shielding, brazing, soldering, or grinding or radioactive or contaminated material.

2. Work in radiation or high radiation areas.

3. Any procedure requiring respiratory protection.

4. Any procedure which in the opinion of the HPO WRAMC or PIC, DORF requires an RWP.

5. The RWP will be used to specify working conditions in order to issue radiation safety and will be submitted by person performing procedure and will be approved by HPO WRAMC AND PIC, DORF.

6. Environmental Monitoring

Environmental monitoring will be in accordance with current WRAMC ENRADMON Plan.

7. Surveys

Periodic radiation & contamination surveys will be performed in accordance with current WRAMC License requirements.

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* This SOP supersedes HP SOP's 3-D-1 thru 3-D-8, dated March 1975

b. Decontamination procedures will be performed, as required, in accordance current NRC License.

8. Health Physics Operational Checks

a. Monitor checks. The air particu;ate monitor, gas stock monitor, auxiliary rotemeters, water monitor. Coincidence monitor, and area monitors will be checked weekly for proper operation. A calibration check will be performed semiannually on the area monitors.

b. A weekly pool water sample will be drawn and analyzed for 3-H and long lived activity. If positive results are encountered, further analysis will be performed as required.

9. Receipt & Shipment of Radioactive Materials

a. Receipt and shipment of radioactive materials will be in accordance with current WRAMC NRC License and will be coordinated with the Radioactive Materials Control Branch, WRAMC.

10. Spent Demineralizer Resins

a. Removal and disposal of spent demineralizer resins will be performed by Health Physics as necessary in accordance with current WRAMC NRC License.

b. Resins will be analyzed prior to shipment to determine radioactive inventory.

11. Air Filter

a. Removal and disposal of used air filters will be performed by Health Physics as necessary.

12. Portable Radiation Survey Instruments

a. Portable radiation survey instruments will be furnished, calibrated and repaired by Health Physics in accordance with current WRAMC NRC License.

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* This SOP supersedes HP SOP's 3-D-1 thru 3-D-8, dated March 1975

DEPARTMENT OF THE ARMY
HARRY DIAMOND LABORATORIES
2800 POWDER MILL ROAD
ADELPHI, MD. 20783

DELHD-N-RBI

28 December 1979

SUBJECT: DORF Dismantling Plan

THRU: ~~Commander~~
US Army Materiel Development
and Readiness Command
ATTN: DRCSF-P (Mr. D. Taras) — D. Taras 14 Jan 80
5001 Eisenhower Avenue
Alexandria, VA 22333

TO: Commander
Office of the Chief of Engineers
ATTN: DAEN-MPZ-E/Mr. J. Wagoner II
Washington, D.C. 20314

1. Reference: Letter dated 26 Nov 1979, DAEN-MPZ-E to DELHD-RBI, Subject: DORF Dismantling Plan.

2. Inclosed are six copies of the revised DORF Dismantling Plan (prepared by Rockwell International) incorporating the changes recommended in paragraph b of the referenced letter.

FOR THE COMMANDER:

Walter L. Gieseler
WALTER L. GIESELER
DORF Physicist-in-Charge

Incl
as

DAEN-MPZ-E   26 NOV 1979

SUBJECT: DGRF Dismantling Plan

THRU: Commander
US Army Materiel Development and Readiness Command
ATTN: DRCSF-P (Mr. D. Taras)
5001 Eisenhower Avenue
Alexandria, VA 22333

TO: Commander
Harry Diamond Laboratories
ATTN: SELHD-RBI (Mr. Giesler)
2800 Powder Mill Road
Adelphi, MD 20783

Subject plan has been reviewed by ARCHS. Approval is hereby granted subject to:

a. Submission to ARCHS prior to commencement of the physical dismantlement operations of six (6) copies of procedures for emergency medical treatment of workers.

b. Incorporation of the following comments in the plan:

(1) Page 17, 3d paragraph - should read US Army Environmental Hygiene Agency.

(2) Page 35, paragraph J.1. - direct reading dosimeters shall be required during work in high radiation areas.

(3) Page 35, last paragraph - Surgeons gloves may not withstand direct handling operations and should be physically protected by an over glove.

OPERATIONS REPORT OF
THE DIAMOND ORDNANCE RADIATION FACILITY
NUCLEAR REACTOR

REPORT NO. 1

1 FEBRUARY 1962 TO 30 DECEMBER 1966

U.S. ARMY MATERIEL COMMAND
HARRY DIAMOND LABORATORIES
WASHINGTON, D.C. 20438

CONTENTS

I. INTRODUCTION

The Harry Diamond Laboratories (HDL) operates an irradiation test reactor with associated experimental equipment which is called the Diamond Ordnance Radiation Facility (DORF). The facility occupies a single remote building on the grounds of the Forest Glen Annex of the Walter Reed Army Medical Center (WRAMC) near Silver Spring, Maryland. It is used to conduct experiments: (1) to investigate and determine susceptibility of ordnance electronic materiel to nuclear weapons radiation environment, mechanisms of those effects and ways and means of developing less susceptible materiel; (2) to conduct research and development in the various physical science and engineering fields directed toward meeting the military characteristics for fuzes or of related items.

The DORF, designed and constructed by the General Atomic Division of General Dynamics Corp., incorporates a reactor which is designated the DORF-TRIGA. General Atomic started up the reactor which attained criticality on 18 September 1961 and operated the facility until 13 January 1962. During this period, the reactor was checked out and fundamental calibration tests were completed. A full report for this period is found in "HDL-TRIGA Reactor Acceptance Test Report", GA-2995, General Atomic, March 1962. HDL accepted the facility on 1 January 1962 and assumed full administrative and operational responsibility on 1 February 1962.

The purpose of this report is to describe, in accordance with letter AMCAD-SN, subject: Reporting Requirements for Army Nuclear Reactors dated 30 September 1966, the DORF operations and modifications. Subsequent reports will be submitted annually.

II. OPERATIONAL PROBLEMS

A. Fuel Elements - Aluminum Clad

The original fuel elements used in the DORF reactor were aluminum clad. The fuel portion of each element consisted of an alloy of uranium-zirconium hydride containing 8 wt-% uranium enriched to 20% in U^{235}. The hydrogen-to-zirconium atomic ratio was approximately 1.0. The fuel meat temperatures were limited to 500°C.

The problems associated with the aluminum clad fuel elements were (a) longitudinal growth and (b) phase transition at 530°C. The growth phenomena resulted from a ratcheting effect between the expanding fuel meat and the aluminum clad. The reactivity insertions were limited to $2.65 to insure that the maximum fuel temperature did not exceed 500°C. The maximum allowable dimension change was a longitudinal growth of 100 mils and/or a bend of 1/16 inch. Thirty (30) fuel elements were removed from reactor service because of growth in a period from reactor startup in September 1961 to June 1964 when the aluminum clad fuel elements were discarded in preparation for the installation of a stainless steel clad core. The total power generated was 40596 KW-HR including power generated in 2450 transient insertions (pulses).

B. Fuel Element Fission Product Leak - Aluminum Clad, 28 October 1963

A leaking fuel element was detected, identified, isolated and placed in storage in a non-hazardous condition and eventually shipped from the reactor facility for disposal. The problem was noticed during a 250 KW operation at steady state reactor operation. The air particulate monitor detected unusual high readings approximately 15 minutes after initial start of the power run. The faulty element was detected and removed from the core and placed in storage. Subsequent pulsing of the reactor showed normal conditions. The fission product monitor or gamma area monitors did not detect the fission product leak. No over-exposures occurred to any personnel at the reactor facility.

C. Fuel Elements - Stainless Steel Clad

It was well known that some of the problems in elongation were associated with the use of aluminum clad fuel elements in a pulse type Triga reactor. In order to correct and significantly improve the dimensional stability of the fuel elements, General Atomic developed a stainless steel clad U-$ZrH_{1.7}$ fuel element. The chief advantages of this type of fuel element are (a) the phase structure of the higher-hydride material is unaffected by temperatures in excess of 700°C whereas the fuel element with a U-$ZrH_{1.0}$ experienced a phase transition at 530°C and (b) the stainless steel cladding increases the strength, integrity and reliability of the fuel element. The stainless steel clad fuel elements presently used at the DORF are identical to those used by General Atomic to pulse their Mark F prototype reactor to 6800 MW peak with a corresponding peak temperature in excess of 750°C with no physical effect on the integrity or reliability of these fuel elements.

The DORF reactor installed a new core composed at the above improved fuel elements and the reactor attained criticality on 15 June 1964. The dimensional stability of the stainless steel clad fuel elements has been extremely good. No appreciable growth has occurred in over 1900 transient reactivity insertions.

D. Excess Reactivity Loss

A good indication of the nuclear condition and/or changes of reactor characteristics, if any, is the excess reactivity measurements. The excess reactivity measurements of the reactor are made each day the reactor is operated. Since the installation of the stainless steel high hydride fuel element core, there has been a gradual decline of excess reactivity. The loss of excess reactivity is not noticeable from day-to-day since the measurements may vary as much as ±5 cents, depending on the previous reactor operating history, experiments near core, and the particular control rods used for measurement, however, long term trends are clearly evident. The total loss of excess reactivity since the installation of the stainless steel core is approximately 40 cents which, at present, prevents the reactor from being pulsed with the core in its "full-in" position against the exposure room tank wall. Therefore, the reactor core is moved about

5 to 7 millimeters toward the center of the reactor pool in order to go critical with the transient rod down. The transient rod must be fully in the core (down position) before the pulsing interlock system is satisfied. The present excess reactivity of the reactor in infinite water reflector and loaded with 85 elements is approximately \$3.63. The exposure room worth is approximately a negative 70 cents.

The initial loss of excess reactivity was noticed after one hundred \$3.00 pulse program when approximately 12 cents were lost. Since that time there has been a gradual decrease in excess reactivity to a rather constant value of \$3.63 for the past 9 months. Although it is recognized that there is no safety hazard, the maximum reactivity insertion has been administratively limited to \$2.65, since the conclusion of the \$3.00 pulse program, so that certain reactor parameters can be measured before the reactivity insertions are increased to \$3.00 pulses.

It appeard from an analysis that the loss of excess reactivity occurred after large reactivity insertions (>\$2.65). This indicates that the reactivity loss is associated with dimensional changes in the fuel. General Atomic has observed this as reactivity losses with steady state reactor power level but ascribes it to decreased heat transfer after pulsing. Since the center fuel elements are in a more severe temperature environment than those in the outer regions of the core, fuel elements from the inner rings were interchanged with those in the outer ring. The results show that although there was less U^{235} in the inner rings than previously, the excess reactivity increased a measurable amount (approximately 8 cents).

The loss in reactivity is not considered a safety hazard since the rate of change is small and is not a large step change. The DORF can be presently pulsed to \$3.00. If the 40 cents should magically reappear as a positive insertion during a pulse, this would mean only a \$3.40 reactivity insertion. While this insertion would be slightly above the operating limits of the DORF reactor, there will be no safety hazard involved since the DORF type fuel elements have been pulsed to \$4.70 for hundreds of pulses with no adverse effects (G.A. Reports 5786 and 6849).

A definitive physical cause for the reactivity loss is not yet known. The DORF staff is continuing to pursue, as the reactor schedule permits, possible mechanisms to identify the particular cause for the loss phenomena. General Atomic and other Triga reactor facilities are being advised and consulted concerning the reactivity loss.

III. CHANGES

A. Concident Radiation Alarm System

On 4 May 1962, a coincident radiation alarm system was installed to notify the WRAMC MP station or a radiation release within the DORF facility. A radiation alarm from any two of the following radiation detectors will actuate the WRAMC MP radiation alarm: (1) Air Particulate Monitor, (2) Victoreen Gamma Monitor above reactor tank and (3) mezzanine Victoreen Gamma Monitor.

B. Removal of Radiation Monitor

In August 1963, the remote area monitor in the Forest Glen Annex Bldg. 500 area was removed and the detector unit installed directly in the DORF stack as a backup monitor of stack effluent.

C. Console Area Inclosure

In November 1963, the reactor console area was inclosed to reduce building noise level, provide an isolated reactor operating area for the reactor operator, and to eliminate outside interference from the rest of the building.

D. Change in Lead Shield Door Interlock System

In January 1964, the interlock system was modified to allow reactor operation in Position 1 (against exposure room tank wall) with the pool lead shield door open.

E. Removal of Plug Door Wood Timber Facing

On 10 January 1964, the wood timbers on the exposure room plug door were removed because of an industrial safety hazard.

F. Modification of Interlock System

In March 1964, the interlock system was modified to allow reactor operation in any position within the reactor pool (Positions 1, 2, and 3) providing the exposure room plug door is closed.

G. Reactor Carriage Position Stops

In March 1964, carriage stops installed for experimental use of a beam tube.

H. Fuel Storage Pits

In May 1964, the construction of the dry storage pits for activated fuel elements was completed and approval was received for their use.

I. Adjustment of Transient Rod Drive Up Limit

14 August 1964 - Mechanical rod and electrical transient rod drive up limit adjusted to prevent insertion of more than $3.00 reactivity.

J. Change in Fuel Element Measurement Interval

8 January 1965 - Approval granted for change in fuel element measurement interval from 100 pulses to 500 pulses.

K. Installation of Emergency Generator

In March 1965, an instantaneous start emergency generator installed and checked out. Electrical distribution modified for connections to emergency generator output.

L. Removal of Transient Rod Drive Up Limit Stops

On 18 May 1965, the $3.00 limit mechanical and electrical stops on transient rod drive mechanism were removed for the purpose of reactivity rod calibration. Mechanic and electric stops reinstalled at completion of calibration.

M. Removal of Fission Products Monitor

On 4 June 1965, approval received for the deletion of the requirement for the fission product monitor as a necessary piece of reactor instrumentation.

N. Ion Chamber Power Supply

On 10 January 1966, additional ion chamber power supply installed as a part of reactor instrumentation. Appropriate scram features of reactor safety system retained.

DEPARTMENT OF THE ARMY
HEADQUARTERS US ARMY MATERIEL DEVELOPMENT AND READINESS COMMAND
5001 EISENHOWER AVENUE. ALEXANDRIA. VA. 22333

D. Taras/seb/AUTOVON 284-9340

DRCSF-P

17 April 1980

SUBJECT: Decontamination of Diamond Ordnance Radiation Facility

THRU: Commander
US Army Electronics ~~Research~~ and Development Command
ATTN: ~~DRDEL-SS~~
Adelphi, MD 20783

TO: Commander
Harry Diamond Laboratories
ATTN: DELHD-N-RBI
Adelphi, MD 20783

1. Reference is made to the following report: Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980.

2. On 10 April the Army Reactor Committee for Health and Safety reviewed the referenced report and concluded that decontamination is consistent with the criteria in NRC Regulatory Guide 1.86 and is as low as reasonably achievable. In PHONECON, 17 April 80, LTC Quillin, WRAMC Radiation Protection Officer, stated these acheived levels are acceptable to WRAMC. Based on the above, the facility is suitable for unrestricted use and occupancy.

DARWIN N. TARAS
Member, Army Reactor Committee
for Health and Safety

CF:
HQDA(DASG-PSP-E); (DAPE-HRS)
DRCIS
DRCSG

DRCSG-E (3 Apr 80) 1st Ind
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Headquarters, US Army Materiel Development and Readiness Command
5001 Eisenhower Avenue, Alexandria, VA 22333 9 Apr 80

TO: Commander, US Army Electronics Research and Development Command
ATTN: DELHD-N-RBI, Adelphi, MD 20783

Subject report has been reviewed by this office and is forwarded for information and appropriate action.

FOR THE COMMANDER:

for Carl W. Johnson MAJ, MS

ROBERT T. CUTTING, M.D.
Colonel (P), MC
Command Surgeon

1 Incl
nc

CARL W. JOHNSON
MAJOR, MSC
MEDICAL ENTOMOLOGIST
OFFICE OF THE SURGEON

CF:
DRCSF-P
DRCSA-NS
DRXOS-ES
DRCIS-A

DEPARTMENT OF THE ARMY
U. S. ARMY ENVIRONMENTAL HYGIENE AGENCY
ABERDEEN PROVING GROUND, MARYLAND 21010

Mr. Lodde/cw/AUTOVON
584-3526

HSE-RH/WP

3 APR 1980

SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Commander
US Army Materiel Development and
Readiness Command
ATTN: DRCSG
5001 Eisenhower Avenue
Alexandria, VA 22333

1. AUTHORITY. Letter, DELHD-N-RBI, Harry Diamond Laboratories, 2 November 1979, subject: Request for a Radiological Health Special Study, and indorsement thereto.

2. PURPOSE. This special study was performed to determine the presence and extent of radioactive contamination and whether the facility met the radioactive contamination levels stated in Nuclear Regulatory Commission, Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974, following decontamination.

3. GENERAL.

a. This radiation protection special study was conducted by Mr. Gordon M. Lodde, Health Physicist, and 2LT Roger M. Davis, Jr., Health Physics Division, this Agency, during the period 25-28 February 1980.

b. An entrance interview and an exit briefing were provided to Mr. Charles Ware, Contracting Officer's Representative, Harry Diamond Laboratories.

4. FINDING.

a. The results of smear surveys are provided in Inclosure 1.

b. The results of concrete analysis are provided in Inclosure 2.

c. Surveys by direct radiation measurements indicated that the highest radiation values were obtained on the north, south, and west walls of the exposure room. The values ranged from 20-400 microroentgen per hour (μR/h) on contact as measured with an Eberline, Model PRM-7, Micro-R-Meter and up to 350 μR/Hr when measured with a Victoreen, Model 440, Ionization Chamber. These two methods of radiation measurements are in close agreement.

5. DISCUSSION.

a. Samples were taken from the wastewater holding tanks and the sewage system down stream from the holding tanks.

b. Core samples were taken off site and soil and vegetation samples were taken both on and off site.

c. The final report will be forwarded in about 60 days following analysis of the samples.

6. CONCLUSION. A review of the findings indicated that after decontamination the facility conformed to the requirements of Regulatory Guide 1.86.

7. RECOMMENDATION. None

FOR THE COMMANDER:

2 Incl
as

FRANK E. McDERMOTT
COL, MSC
Director, Radiation and
Environmental Sciences

CF:
Cdr, ERADCOM
Cdr, HSC (HSPA-P)

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

RESULTS OF ANALYZING WIPE TEST SAMPLES

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 c Gross Alpha Activity	Gross Beta Activity
1	L244	< 1.4	4.4 ± 2.5
2	L245	< 1.4	< 2.5
3	L246	< 1.4	< 2.5
4	L247	< 1.4	< 2.5
5	L248	< 1.4	< 2.5
6	L249	< 1.4	< 2.5
7	L250	< 1.4	< 2.5
8	L251	< 1.4	2.8 ± 2.0
9	L252	< 1.4	< 2.5
10	L253	< 1.4	6.0 ± 2.7
11	L254	< 1.4	2.6 ± 2.0
12	L255	< 1.4	< 2.5
13	L256	< 1.4	< 2.5
14	L257	< 1.4	< 2.5
15	L258	< 1.4	< 2.5
16	L259	< 1.4	3.6 ± 1.9
17	L260	< 1.4	< 2.5
18	L261	< 1.4	< 2.5
19	L262	< 1.4	14.6 ± 3.7
20	L263	4.7 ± 2.4	14.0 ± 3.6
21	L264	< 1.4	< 2.5
22	L265	< 1.4	6.2 ± 2.3
23	L266	< 1.4	7.0 ± 2.6
24	L267	3.2 ± 1.9	< 2.5
25	L268	< 1.4	5.2 ± 2.4
26	L269	< 1.4	< 2.5
27	L270	< 1.4	3.0 ± 2.0
28	L271	< 1.4	< 2.5
29	L272	< 1.4	< 2.5

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm^2 Gross Alpha Activity	Gross Beta Activity
30	L273	< 1.4	3.2 ± 2.2
31	L274	< 1.4	9.8 ± 3.2
32	L275	< 1.4	3.2 ± 2.3
33	L276	< 1.4	< 2.5
34	L277	< 1.4	< 2.5
35	L278	< 1.4	3.2 ± 2.4
36	L279	< 1.4	3.2 ± 2.1
37	L280	< 1.4	5.0 ± 2.4
38	L281	< 1.4	4.8 ± 2.3
39	L282	< 1.4	< 2.5
40	L283	< 1.4	< 2.5
41	L284	< 1.4	3.4 ± 2.1
42	L285	< 1.4	< 2.5
43	L286	< 1.4	< 2.5
44	L287	< 1.4	< 2.5

Alphus L. Jones

ALPHUS L. JONES, Chief
Radl & Biol Chem Div, USAEHA

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

INTERIM RESULTS OF ANALYZING CONCRETE SAMPLES

Sample Identification	RCB Lab No.	Microcurie per Gram ±2 Standard Deviations Europium-152 Activity	Europium-154 Activity	Cobalt-60 Activity
EX-N	RC1	$3.5 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.8 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$1.0 \times 10^{-5} \pm 0.4 \times 10^{-6}$
EX-S	RC2	$5.9 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$4.5 \times 10^{-6} \pm 0.8 \times 10^{-6}$	$3.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
ES In Pool	RC3	$1.6 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$1.4 \times 10^{-6} \pm 0.4 \times 10^{-6}$	$5.4 \times 10^{-6} \pm 0.3 \times 10^{-6}$
ES-W	RC4	$2.8 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.2 \times 10^{-6} \pm 0.5 \times 10^{-6}$	$1.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
EX LIFT-S	RC5	$1.1 \times 10^{-4} \pm 0.2 \times 10^{-5}$	$7.9 \times 10^{-6} \pm 0.9 \times 10^{-6}$	$3.0 \times 10^{-5} \pm 0.1 \times 10^{-5}$

Alphus L. Jones
ALPHUS L. JONES, Chief
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APPENDIX IV

ACTIVATION ANALYSIS OF RADIOACTIVE MATERIAL IN THE DORF STRUCTURE BEFORE DECOMMISSIONING

INTRODUCTION

This report documents information of the amount and type of radioactive material that will be present in the structure and building of the Diamond Ordnance Radiation Facility after removal of the reactor fuel in the spring of 1978. Such information is required for decommissioning plans and must be supplied to the Army Reactor Committee for Health and Safety (ARCHS) prior to their approval of such plans. The information is also needed by the waste-disposal area directorate who must budget for specific volumes and radioactive levels. Finally, the isotopic composition of the radioactive waste is necessary for labeling containers at the time of shipment.

The first section of this report is a summary for those who need only the final results on type, location and amount of residual radioactivity. Section two describes the investigative procedures, discusses the possible sources of radioactivity and the properties of the radioactive isotopes found. Graphs of isotopic analyses and calculations, which convert detector response to specific activities, are included in this section. The second section also provides the detailed calculations of volumes, weights and total radioactivity in the various sections of DORF. The final section contains recommendations based on things discovered during this study.

SUMMARY

The radioactivity that will remain at DORF after the fuel removal in the spring of 1978 has been carefully estimated based on criteria, measurements and necessary assumptions documented in this report. A concise summary of that radioactivity is given in Table I. The most predominant radioactive isotopes in the concrete are cobalt-60 and europium-152 and -154. The most predominant isotopes in lead are antimony-124 and silver-110^{m}. The wood and steel (mainly in the lead-shield hoist) are not very radioactive and easy to dispose of. The aluminum itself is almost non-radioactive but there is a radioactive Phenoline

liner which tends to stick to the aluminum. Its radioactivity comes from cobalt-60 and zinc-65. All of these readioactive isotopes have half-lives in excess of 60 days.

TABLE I. Summary[1/] of total radioactivity to be expected from materials in the DORF structure after core removal.

Material	Mass (lbs)	Volume (ft^3)	Radioactivity (millicuries)
CONCRETE	82,170	412	36.24
(If whole plug door included).	(170,050)	(850)	(36.24)
LEAD	55,753	112	13.34
ALUMINUM	2,288	15	75.71
WOOD	34,944	1344	0.33
STEEL	2,662	5.5	0.03
GRAND TOTAL	177,817 lbs (89 tons)	1889 ft^3	0.126 Curies

[1/] This represents a summation of the values given in Table X.

IDENTIFICATION OF THE RADIOACTIVITY

Isotope Identification:

The principal method of identification was gamma-radiation spectroscopy with a germanium lithium-drifted detector, or Ge(Li) crystal. The crystal is housed inside a very low-activity-lead cave lined with wood. Numerous background analyses confirm that for photons with energies greater than 140 keV, samples with low activities (two to three times background) can be successfully analyzed for specific photon energies. A plot of a multichannel analyzer spectrum of the background is given in Fig. 1. The principal higher energy peaks in the background spectrum are the 511-keV gammas associated with annihilation radiation and the 1461-keV peak from ^{40}K, a radioactive isotope which is found naturally in almost all "non-radioactive" materials.

The method of analyses provides for very good resolution of the photon energies in the range 140 keV to 2500 keV at approximately 9.7 keV per channel of 256 total channels. The electronic equipment is sufficiently stable over counting periods of 50,000 seconds to permit energy assignment within two percent. Graphs of the gamma spectra of the various materials investigated are shown. (See Fig. 2 through 5).

The method does not provide for the ultimate in accuracy for determining specific activity. The crystal efficiency (disintegrations per count as a function of energy) can only be accurately assigned for a well-defined geometry. The samples in the present situation varied in size and shape. Therefore, they were suspended above the crystal so that their centers of mass were approximately three centimeters from the active volume of the detector and efficiencies were determined with calibrated point sources. The error associated with this procedure is estimated to be no greater than 50%, based on a volume integration of point-source response at points in space representative of the sample size. For the task at hand such accuracy is sufficient.

Rational of Sample Selection

The job was to identify the radioactive content and quantity of materials that will have to be removed from the DORF site so that it can be certified, by post-decommissioning radioactive survey, as an unrestricted area for possible public use. This survey, to be conducted by the Army Environmental Health Agency (AEHA), must be accomplished prior to any filling, sealing or burying activities. This presented two problems. How can we identify the radioactivity in presently inaccessible areas, such as below the reactor pool, before the reactor fuel and higher-level radioactive structures have been removed? What amount of material will have to be removed from walls and floors to reach an acceptable AEHA level?

The first problem was attacked as follows. Representative samples of all the material types are accessible in the exposure-room area. Because of the significantly larger thermal-neutron cross sections of materials and the fact that the DORF-TRIGA reactor is zirconium-hydride moderated and water-cooled reactor, the thermal component of the spectrum is the dominate source of induced radioactivity. As will be discussed later, the predominance of radioactive europium confirms this. Therefore, isotopic analyzes of exposure room samples are representative of those in presently inaccessible areas. Furthermore, with facility dosimetry data for the various locations, we can estimate the residual radioactivity in remote locations with significantly different flux exposure levels.

The second problem of how much material to remove is more complex because we do not have good guidance on the amount of radioactivity in volume that can remain. NRC Regulation 1.86, the current guide, clearly specifies levels for removal surface contamination but is, at best, vague on volume activity and how to detect it. The criteria set for the analysis in this report are as follows:

(1) Once the reactor support structure has been removed there will be no high-level radioactive waste remaining in the DORF structure. Our analyses confirm this.

(2) Based on existing allowable concentrations of radioactive materials in water and a specific activity proportional to material density, we can set an allowable specific activity of 2×10^{-5} microcuries per gram as the maximum permissible concentration of radionuclides in water when it is known that Sr 90, I 129, (I 124, I 126, I 131, Table II only)*, Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present. Since the density of water is one g/cm^3 and there are 28317 cm^3/ft^3, 2×10^{-5} µCi/g corresponds to 0.57 µCi/ft^3 of water.

(3) It is assumed that the radioactivity is distributed in the material to be removed in proportion to the incident thermal fluence (flux-time product) and attenuated exponentially according to thermal-neutron relaxation lengths, (i.e., the inverse of microscopic removal cross sections for broad beams). Half-life decay is taken into consideration for the period until spring 1978. Therefore, the depth of material to be removed, D in centimeters, is determined

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* 10 CFR20, note to Appendix B

by relative fluence level at the surface, ϕ/ϕ_o, and

$$D = L \ln \frac{\phi/\phi_o \times A}{0.57 \mu Ci/ft^3} \quad (1)$$

where A is the activity in $\mu Ci/ft^3$ estimated from this study. The values of relaxation length are given in Table II.

TABLE II. Material densities and relaxation lengths, L

Material	Density	Relaxation Length
Concrete	2.35 g/cm^3	1.6 cm
Lead	11.0 g/cm^3	4.2 cm
Wood	0.42 g/cm^3	2.9 cm

Measured Radioactivity

Samples taken from the DORF exposure room were concrete, wood, aluminum, lead and a tar-paper-like liner installed between the aluminum pool tank and the concrete pool base. Although the aluminum itself has very little residual radioactivity (less than $8 \times 10^{-6} \mu Ci/gm$ for the sections counted), the Phenoline paper (i.e., the tar-paper liner) has the highest specific activity of all the materials examined. Since this liner tends to stick to the aluminum, for all practical purposes the aluminum tank exhibits this activity.

Tables IV through IX give a breakdown of the isotopic composition of the radioactivity in the various samples. Tables IV and V are composed of several additional pages that serve as detailed examples of the methods of analyses and are self explanatory when reference is made to the graphs of the multichannel-analyzer output. Figs. 1 through 5 are the multichannel-analyzer gamma spectra for the various types of samples. The energy of the photopeaks is related to

TABLE III. GAMMA* SPECIFIC ACTIVITY AND THE NUMBER OF MICROCURIES PER UNIT OF MATERIAL FOR VARIOUS RADIOACTIVE MATERIALS FROM THE DORF EXPOSURE ROOM

Type of Material	Location in Exposure Room	Specific Activity (d/s·g)	Activity per unit of material
1. PHENOLINE PAPER	On aluminum tank near exposure room end of pool	636	3.6 μCi/sq ft
2. CONCRETE	From front part of room about 4 feet from reactor	78	140 $\mu Ci/ft^3$
3. CONCRETE	Very near reactor at exposure room end of tank	141	252 $\mu Ci/ft^3$
4. LEAD	From curtain above the movable lead shield	72	0.62 μCi/lb
5. LEAD	From brick in middle of the movable lead shield	205	1.30 μCi/lb
6. WOOD	From very near reactor and concrete sample #3, above.	1.3	0.40 $\mu Ci/ft^3$

*From gross beta plus gamma analyses, the beta-to-gamma activity of all these different materials is approximately 1.8.

spectra for the various types of samples. The energy of the photopeaks is related to start the channel number (abcissa) by the following equation:

$$E(\text{keV}) = (\text{channel} + 2.5) \times 9.69 \quad \pm 2\% \tag{2}$$

For clarification, the gamma-ray peaks are indentified by isotope and their energies in keV (and in parenthsis) are given for most of the peaks.

The specific activity (d/s·g) for each measured sample is compared in Table III. This table also provides the number of microcuries per unit most practical for that type of material. This latter information is used in Table X to determine the total radioactivity in the volumes of radioactive materials at DORF.

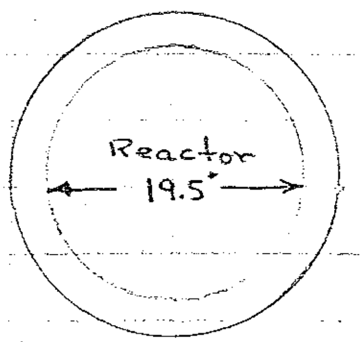
CONCRETE EXCAVATION

The maximum observed concrete-volume activity was 252 $\mu Ci/ft^3$. If we assume that the maximum fluence at any location is 100 times larger than at this monitored exposure room location, Eq. (1) ... the excavation-depth formula ... requires

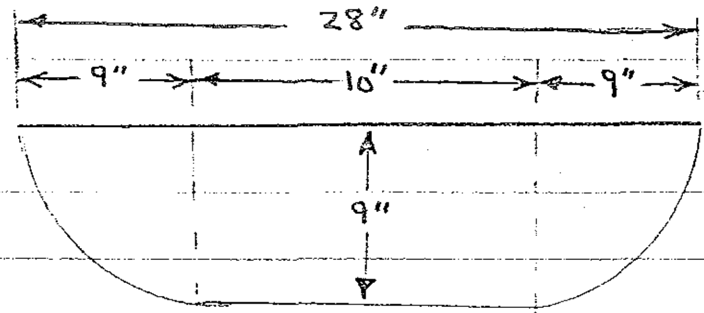
$$D = 1.6 \ln\left(\frac{100 \times 252}{0.57}\right) = 17.1 \text{ cm or } 6.7 \text{ inches}$$

Of course, fast neutrons will also penetrate and thermalize but the factor of 100 is already very conservative so excavation depths of 9 to 10 inches in the immediate vicinity of the reactor and at the end positions of the pool should adequately remove the radioactivity.

Excavation below reactor in Pos #3 & radiography position:



Excavation diameter
28"



Volume: $\pi r^2 l + \frac{1}{2}\left(\frac{4}{3}\pi r^3\right)$

$\pi\left[(10^2)9 + \frac{2}{3}(9^3)\right]$

$V = \pi(900 + 486) \div 1728 = 2.52 \, ft^3$

<u>Activity in concrete below Pos. 3</u>: Activity will not exceed measured maximum of 252 $\mu Ci/ft^3$ by more than a factor of two. Assign 500 $\mu Ci/ft^3$.

$2.52 \times 500 = 1260 \, \mu Ci$

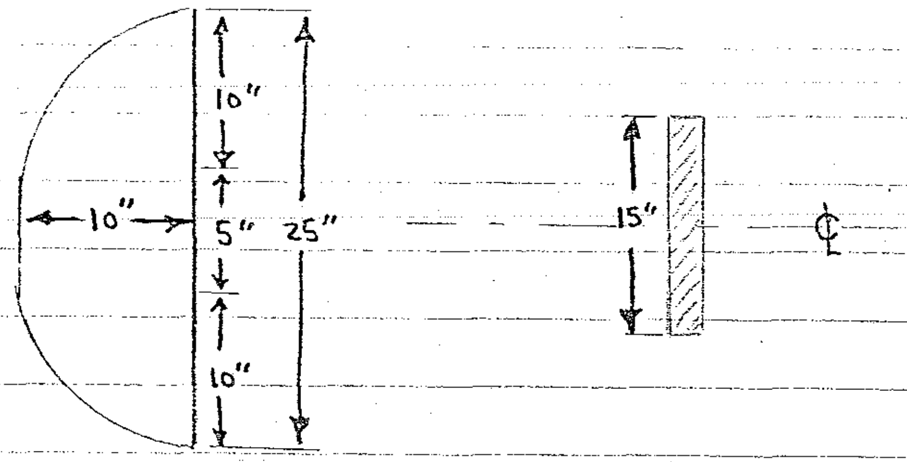
<u>Activity in concrete below radiography position:</u>

Assign 250 $\mu Ci/ft^3$ (conservative)

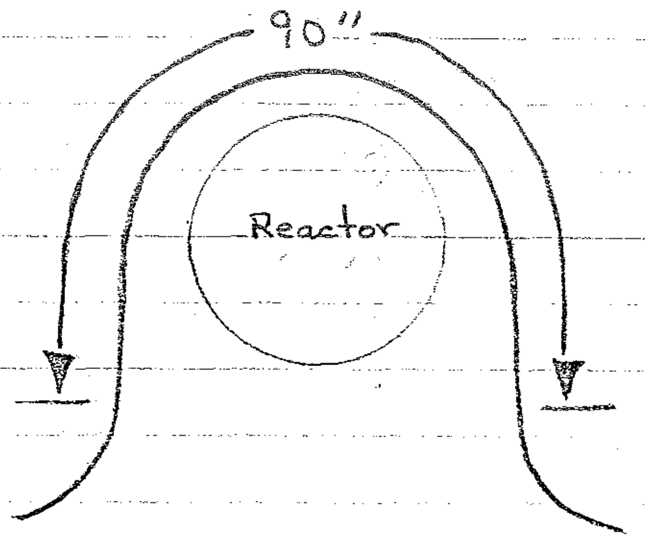
$2.52 \times 250 = 630 \, \mu Ci$

Excavation from tank-support walls in Pos. #1 & #3:
(active fuel length is 15 inches)

FOR 1&3:



Volume for Pos. 3 end of pool:



Volume $L \times (\frac{1}{2}\pi r^2 + t \times d)$

$90\left(\frac{\pi}{2}(10^2) + 5 \times 10\right)$

$V = 90(157 + 50) \div 1728 = 10.8\ ft^3$

Volume for Pos. 1 end of pool: Because tank projects into the exposure room, L for pos. 1 in taken as 40" rather than 90" and $V = \frac{4}{9}(10.8) = 4.8\ ft^3$

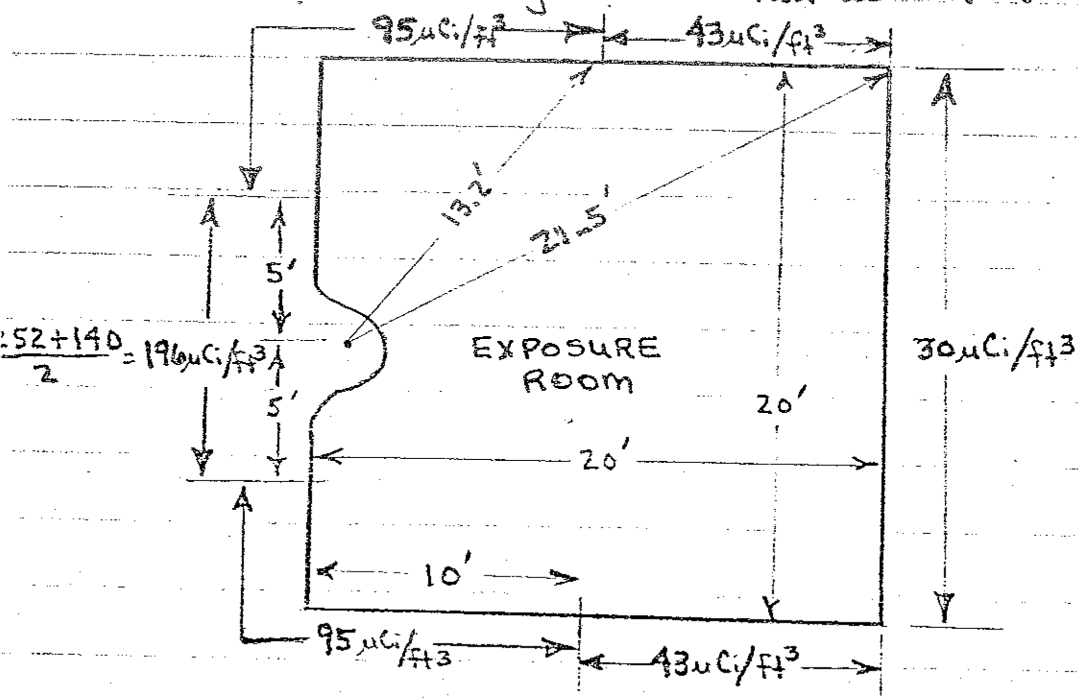
Activity for wall excavation in Pos. 3 & 1: Consistant with conservative estimate for concrete below reactor in Pos. 3, assign $500\ \mu Ci/ft^3$

Activity for Pos. 3 end: $500 \times 10.8 = 5400\ \mu Ci$

Activity for Pos. 1 end: $500 \times 4.8 = 2400\ \mu Ci$

Excavation of concrete from walls, floor and ceiling of the DORF exposure room.

The following diagram shows the estimated volume activity in the concrete of the exposure room walls. Because of thermalization of fast flux by the wood and scatter from the walls, the effective thermal flux is assumed to fall off more nearly as 1/R than as 1/R².



room heighth 8 feet.

Excavation depths:

No.	D (inches)	Assign Activity/ft³
1	3.7"	196 µCi/ft³
2	3.2	95 "
3	2.7	43 "
4	2.5	30 "

Volumes of exposure room section 1 thru 4:

No. 1: Area: $8' \times 10' = 80\ ft^2$
Volume: $80 \times 3.7/12 = 24.6\ ft^3$
Activity: $24.6 \times 196 = 4822\ \mu Ci$

No. 2: Wall Area: $2(5' \times 8') + 2(10' \times 8') = 240\ ft^2$
Ceiling/Floor Area: $2(20' \times 10') = 400\ ft^2$
$\Big\}\ 640\ ft^2$
Volume: $640 \times 3.2/12 = 171\ ft^3$
Activity: $171 \times 95 = 16245\ \mu Ci$

No. 3: Wall Area: $2(10' \times 8') = 160\ ft^2$
Ceiling/Floor Area: $2(20' \times 10') = 400\ ft^2$
$\Big\}\ 560\ ft^2$
Volume: $560 \times 2.7/12 = 126\ ft^3$
Activity: $126 \times 43 = 5418\ \mu Ci$

No. 4: Area: $20' \times 8' = 160\ ft^2$
Volume: $160 \times 2.5/12 = 33\ ft^3$
Activity: $33 \times 30 = 1000\ \mu Ci$

Exposure Room Totals:
Volume: $354.6\ ft^3$
Weight: 70,920 lbs.
Activity: 27.49 mCi

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IV. CONCRETE SAMPLE FROM "FRONT" OF EXPOSURE ROOM

$M = 2.47\,g$ $T = 54,252$ seconds $(MT = 1.34 \times 10^5)$

ISOTOPE	Half Life $T_{1/2}$	Activity dis/s.g	% of Total Activity
^{60}Co	5.27y	31.66	40.55 %
$^{152}Eu - ^{154}Eu$	12.2y - 16y	30.15	38.62 "
^{46}Sc	84 days	5.44	6.97 "
^{134}Cs	2.07 y.	5.06	6.48 "
^{65}Zn	245 d	2.29	2.93 "
^{182}Ta	115 d	1.19	1.52 "
^{124}Sb	60 d	0.63	0.81 "
* Annihilation	—	1.65	2.12 "
		TOTAL: 78.07 d/s.g	(100.00)

Nat. Bkg in concrete
(^{226}Ra series)
^{214}Bi 1622y 0.84 d/s.g.
(i.e., concrete is 93 times its own natural background)

Activity per cubic foot of concrete

$(12 \times 2.54)^3 = 28317\ cm^3/ft^3$
ρ concrete $= 2.35\ g/cm^3$ } $6.65 \times 10^4\ g/ft^3$

$$\text{Activity} = \frac{6.65 \times 10^4 \times 78.07}{3.7 \times 10^{10}\ d/s.ci} = 1.40 \times 10^{-4}\ Ci \text{ or } \boxed{140\,\mu Ci/ft^3}$$

Estimate per "ml" of H_2O $(78.07)/3.7 \times 10^{10} \times 2.35 = 8.98 \times 10^{-10}$ or $8.98 \times 10^{-4}\ \mu Ci/g$

* Note: All activities except annihilation radiation (511 keV) were determined from sample with background substracted.

(2 pag

TABLE IV-A CONCRETE SAMPLE FROM FRONT OF EXPOSURE ROOM

WEIGHT = 2.47 g (BKG. SUBSTRACTED FOR EQUAL TIME: 54,252 seconds)

* Indicated more that one isotope has a peak in this channel 9.71±0.05 kev/ch+2.5

SOURCE ISOTOPE & $T_{1/2}$	keV ENERGY	CHANNEL	PEAK GROSS COUNTS	COMPTON BKG COUNTS	PEAK NET COUNTS	ϵ c/d	Total d
^{152}Eu – ^{154}Eu (12.2y – 16year)	$122^{152,154}$	10	23500	3364	20136	2.50×10^{-2}	805440
	$245 \& 248^{154}$	23	4889	2030	2859	1.16×10^{-2}	246465
	344^{152}	33	8441	1756	6685	8.10×10^{-3}	825308
	$411^{154,152}$	40	1894	1370	524	6.66×10^{-3}	78797
	488^{152}	47.8	1658	1340	318	5.58×10^{-3}	56989
	689^{154}	68.8	1366	1230	136	3.90×10^{-3}	34872
	$720 \& 723.1^{154}$	72 *	1357	1240	117	3.63×10^{-3}	32231
	779^{152}	78	2562	1250	1313	3.35×10^{-3}	391940
	$867 \& 873^{154}$	87	1788	1325	463	2.95×10^{-3}	156949
	966^{152}	97	2457	1150	1307	2.66×10^{-3}	491353
	1006^{154}	101	1130	940	190	2.57×10^{-3}	73930
	~~1092^{152}~~	~~110~~	~~1728~~	~~975~~	~~753~~	~~2.45×10^{-3}~~	~~307347~~ delete because c Zinc
	1277^{154}	129	321	135	186	1.95×10^{-3}	95385
	1412^{152}	143	1220	50	1170	1.74×10^{-3}	672913
	1457^{152}	146.7	73	32	41	1.67×10^{-3}	24551
	1528^{152}	155	59	18	41	1.60×10^{-3}	25625
	1595^{154}	162.5	60	17	43	1.52×10^{-3}	28289
						TOTAL =	4.046×10^{6}
^{60}Co (5.27 years)	1173.1	119	4857	589	4268	2.11×10^{-3}	2.023×10^{6}
	1332.4	135	4187	126	4061	1.86×10^{-3}	2.219×10^{6}
						TOTAL =	4.242×10^{6}
^{134}Cs (2.07 year)	563&569	56	1580	1300	280	4.72×10^{-3}	5.932×10^{4}
	605	60 *	1612	1293	319	4.36×10^{-3}	7.317 "
	795&802	80	2101	1300	801	3.25×10^{-3}	24.646 "
	1040	105 *	870	840	30	2.50×10^{-3}	1.200 "
	1168	119 *	(370)	120	250	2.15×10^{-3}	11.628 "
	1365	139	370	60	310	1.81×10^{-3}	17.127 "
						TOTAL =	6.785×10^{5}
^{65}Zn (245 day)	1115.4	110	1728	975	753	2.45×10^{-3}	307347
						TOTAL	3.073×10^{5}

(2 page table)

TABLE IV-A. (continued)

ISOTOPE	kev ENERGY	channel	GROSS	Bkgs	Net	ε c/d	Total d
^{46}Sc (84 days)	889.4	89 *	2491	1400	1091	2.90×10^{-3}	3.762×10^{5}
	1120.3	113	2330	(1550)	780	2.21×10^{-3}	3.529×10^{5}
						TOTAL	7.291×10^{5}
^{124}Sb (60 day)	602.6	60 *			200	4.40×10^{-3}	45.454×10^{3}
	646	65 *			25	4.17×10^{-3}	5.995 "
	722.8	72 *			25	3.63×10^{-3}	6.887 "
	968	97 *	1120	1100	20	2.66×10^{-3}	7.518 "
	1045	105 *	870	853	17	2.50×10^{-3}	6.800 "
	1691	172	41	28	13	1.43×10^{-3}	9.091 "
	2088	209	25	22	3	1.15×10^{-3}	2.609 "
						TOTAL =	8.435×10^{4}
^{182}Ta (115 day)	100.3	8	4577	4102	475	3.10×10^{-2}	15.322×10^{3}
	152.4 & 156	13.6	3054	2700	354	1.42×10^{-2}	24.930 "
	198.4	17.5 *	2620	2450	170	1.50×10^{-2}	11.333 "
	223	20.5	2348	2210	138	1.35×10^{-2}	10.222 "
	1121	113 *	870	810	60	2.21×10^{-3}	27.149 "
	1189	119	(710)	650	60	2.09×10^{-3}	28.708 "
	1221.6	123	337	252	85	2.05×10^{-3}	41.463 "
						TOTAL =	1.591×10^{5}
Annihilation	511	50	4100	3100	1,000	4.50×10^{-3}	2.22×10^{5}
						TOTAL	2.22×10^{5}

Additional analysis of "natural background in concrete:

^{214}Bi (1622 years*)

* from ^{226}Ra series

	kev ENERGY	channel	GROSS	Bkgs	Net	ε c/d	Total d
	1731	177	36	27	9	1.39×10^{-3}	6.475×10^{3}
	1767	180	39	29	10	1.24×10^{-3}	8.064 "
	2119	217	23	14	9	1.13×10^{-3}	7.965 "
	2204	226	32	23	9	1.08×10^{-3}	8.333 "
	2293	234	28	12	16	1.03×10^{-3}	15.534 "
	2447	250	36	25	11	9.40×10^{-4}	11.702 "
						TOTAL	5.807×10^{4}

Actual about 2x this → TOTAL 5.807×10^{4}

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE V. TAR-PAPER* SAMPLE FROM FRONT OF EXPOSURE ROOM

* CALLED PHENOLINE COATING

$M = 3.69$ g $T = 20{,}183$ s ($MT = 7.45\times10^4$)

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s·g	% of Total Activity
^{65}Zn	245 day	185.2	29.1%
^{60}Co	5.27 year	363.8	57.2%
^{152}Eu	12.2 year	86.8	13.7%
	TOTAL:	635.8 d/s·g	100.0%

From gross counts, this sample was $423 \div 11 = 38.5$ times more active than the natural background in the Ge(Li) crystal cave. NOTE: It is $635.8/78.1 = 8.1$ times more radioactive than the concrete from the front part of the room.

Activity per square foot of paper:
(assuming a "1/8-inch thickness")

$(12\times2.54)^2\times0.125\times2.54 = 295\ cm^3$ per square foot

$\rho = 0.7\ g/cm^3$

Activity $\dfrac{635.8\ d/s\cdot g\times206.5g}{3.7\times10^{10}\ d/s\cdot Ci} = \boxed{3.55\ \mu Ci/sq.foot}$

TABLE VI A. TAR-PAPER SAMPLE FROM POS #1 END OF ALUMINUM TANK (IN EXPOSURE ROOM).

TIME = 20,183 sec.

M = 3.6875 grams

423 sample/10c
11 ileg

20183 000004 002000

		1	2	3	4	5	6	7	8	9
000000	020183	000004	065213	016877	013805	014030	014273	013571	013285	013616
000010	$\boxed{048361}$	011699	011727	011230	011878	010214	010054	009864	009489	009693
000020	009953	009577	009525	$\boxed{013469}$	008729	008657	008524	008433	008402	008066
000030	008837	008377	007927	$\boxed{019884}$	007346	007546	007144	006983	006859	006789
000040	007690	006823	006536	007492	006603	006469	006496	006540	006508	006670
000050	$\boxed{011253}$	006750	006323	006526	006497	006455	007013	006515	006682	006515
000060	007831	006514	006633	006741	006702	006907	006732	006904	007032	006989
000070	006557	006954	007301	007106	007305	007392	007381	007520	$\boxed{009923}$	007635
000080	008622	008056	008383	008368	008992	008968	009128	010116	009901	012019
000090	010301	009768	008668	008394	008263	008354	008177	009248	006260	005804
000100	005583	005638	004958	004642	004680	004475	004422	004334	004357	004701
000110	005809	004901	009792	$\boxed{037642}$	002790	002421	002438	002329	007215	$\boxed{028249}$
000120	001547	001386	001187	001140	000822	000787	000644	000546	000530	000822
000130	000530	000744	000621	000538	000840	$\boxed{027671}$	000259	000203	000178	000189
000140	000194	000178	000271	$\boxed{002053}$	000157	000170	000186	000296	000244	000122
000150	000132	000137	000141	000132	000135	000193	000148	000144	000137	000121
000160	000126	000137	000143	000130	000094	000106	000124	000107	000116	000104
000170	000114	000116	000092	000124	000107	000099	000118	000084	000094	000087
000180	000094	000080	000089	000100	000093	000115	000099	000102	000088	000080
000190	000082	000107	000089	000090	000090	000090	000095	000092	000078	000081
000200	000093	000089	000093	000095	000085	000116	000081	000082	000079	000068
000210	000090	000064	000065	000062	000069	000070	000053	000076	000058	000074
000220	000066	000064	000068	000052	000064	000070	000059	000076	000054	000062
000230	000072	000062	000049	000079	000056	000047	000041	000045	000043	000034
000240	000033	000027	000023	000019	000027	000016	000012	000013	000012	000016
000250	000013	000006	000009	000011	000016	000019				

TABLE V-B.

ISOTOPIC ANALYSES OF TAR-PAPER SAMPLE FROM ON TANK WALL AT EXPOSURE-ROOM END AT APPROXIMATELY CORE CENTERLINE.

COUNT TIME = 20,183 sec. ; WEIGHT = 3.6875 g

SOURCE ISOTOPE/ $T_{1/2}$	ENERGY* (keV)	CHANNEL #	PEAK GROSS COUNTS	COMPTON BKG. COUNTS	PEAK NET COUNTS	XTAL EFF. (c/d)	TOTAL d
^{65}Zn (245 day)	1115.4	113	37642	3700	33942	2.45×10^{-3}	1.38×10^{7}
						TOTAL =	(1.38×10^{7})
^{60}Co (5.27 year)	1173	119	28249	1700	26549	2.11×10^{-3}	1.26×10^{7}
	1332	135	27671	750	26921	1.86×10^{-3}	1.45×10^{7}
						TOTAL =	(2.71×10^{7})
^{152}Eu (12.2 year)	121.8	10	48361	12600	35761	2.50×10^{-2}	1.43×10^{6}
	244.6	23	13469	9140	4329	1.16×10^{-2}	0.30 "
	344.2	33	19884	7700	12184	8.10×10^{-3}	1.50 "
	411.0	40	7690	6790	900	6.65×10^{-3}	0.14 "
	443.9	43	7492	6500	992	6.60×10^{-3}	0.15 "
	778.6	78	9932	7550	2382	3.35×10^{-3}	0.71 "
	964.1	97	9248	7200	2048	2.66×10^{-3}	0.77 "
	1086.0	110	5809	4800	1009	2.45×10^{-3}	0.41 "
	1407.5	143	2053	210	1843	1.74×10^{-3}	1.06 "
						TOTAL =	(6.97×10^{6})

* Energy is approximately (±1.5%) 9.69 × (CHANNEL + 2.4) keV/channel

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VI. CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE ROOM

$M = 11.63$ g $T = 40{,}000$ sec $(MT = 4.65 \times 10^5)$

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s·g	% of Total Activity
^{60}Co	5.27 y	63.2	44.5 %
^{152}Eu-^{154}Eu	12.2 y – 16 y	55.4	39.0 "
^{46}Sc	84 day	10.5	7.4 "
^{65}Zn	245 day	4.0	2.8 "
^{182}Ta	115 day	3.1	2.2 "
^{134}Cs	2.07 y	1.3	0.9 "
^{124}Sb	60 day	0.2	1.2 "
Annihilation	—	2.8	2.0 "
	TOTAL	140.5 d/s·g	(100.0)

Note: This concrete sample is 1.8 times more radioactive than the sample from the front of the room.

Activity per cubic foot of concrete:

Density: 6.65×10^4 g/ft³

$$\text{Activity} = \frac{6.65\times10^4 \times 140.5}{3.7\times10^{10}\ d/s\cdot Ci} = 2.52\times10^{-4}\ Ci \text{ or } \boxed{252\,\mu Ci/ft^3}$$

Estimate per "ml" of H_2O: $1.62 \times 10^{-3}\ \mu Ci/g$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VII. LEAD FROM BRICK IN MOVABLE LEAD SHIELD

M = 2.531 g   T = 40,000 sec   (MT = 1.01×10^5)

ISOTOPE	HALF-LIFE ($T_{1/2}$)	ACTIVITY d/s·g	% of TOTAL ACTIVITY
^{124}Sb	60 days	181.8	71.0%
^{110m}Ag	253 day	67.3	26.3%
^{152}Eu-^{154}Eu	12.2y – 16 y	1.5	0.6%
Annihilation	—	5.4	2.1%
	TOTAL	256.0 d/s·g	(100.0%)

Activity per pound of lead:
454 g/lb

$$\text{Activity} = \frac{454 \times 256}{3.7 \times 10^{10}} = 3.14 \times 10^{-6} \text{ or } 3.14\,\mu Ci/lb$$

NOTE: About 44% of this radioactivity will decay by spring 1978 because of the ^{124}Sb contribution. Therefore Eff. Activity = 1.76 µCi/lb.
Futhermore, the top and bottom quarter sections of the lead shield should have activities more like that of the lead curtain (Eff. Act = 0.62 µCi/lb) so a better value for the whole shield is $0.6 \times 1.76 + 0.4 \times 0.62 = \boxed{1.30\,\mu Ci/lb}$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VIII. LEAD FROM CURTAIN

M = 0.602 T = 40,000 sec ($MT = 2.41 \times 10^4$)

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY (d/s.g)	% of TOTAL ACTIVITY
^{124}Sb	60 days	4.96	62.1%
^{110m}Ag	253 days	2.28	28.5%
^{46}Sc	84 days	0.45	5.6%
$^{152}Eu-^{154}Eu$	12.2y - 16y	0.05	0.7%
Annihilation	——	0.25	3.1%
		7.99 d/s.g	(100.0)

Activity per pound of lead:
(454 g/lb)

$$\text{Activity}: \frac{454 \times 7.99}{3.7 \times 10^{10}} = 0.98 \times 10^{-7} = 1.0\,\mu Ci/lb$$

NOTE: About 38% of this activity will have decayed by spring 1978.
Therefore: Eff Activity = $\boxed{0.62\,\mu Ci/lb}$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IX. WOOD FROM FRONT OF EXPOSURE ROOM

M = 36.58 g T = 8967 sec. (MT = 3.28×10^5)

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY d/s.g	% of TOTAL ACTIVITY
$^{152}Eu-^{154}Eu$	12.2y-16y	0.588	46.7 %
^{60}Co	5.27 y	0.534	42.4 %
46 Sc	84 day	0.055	4.4 "
^{129m}Te	41 day	0.040	3.2 "
^{127m}Te	110 day	0.026	2.1 "
Annihilation	——	0.017	1.2 "
	TOTAL	1.26 d/s.g	(100.0)

Activity per cubic foot of wood

$$28317 \ cm^3/ft^3 \times 0.42 \ g/cm^3 = 1.189 \times 10^4 \ g/ft^3$$

$$\text{Activity} : \frac{1.26 \times 1.189 \times 10^4}{3.7 \times 10^{10}} = 4.04 \times 10^{-7} \text{ or } \boxed{0.40 \mu Ci/ft^3}$$

Less than 10% of this activity will decay by spring 1978.

NOTE: From WRAMC Health Physics survey of this sample we find there is a β/γ ratio of approximately two.

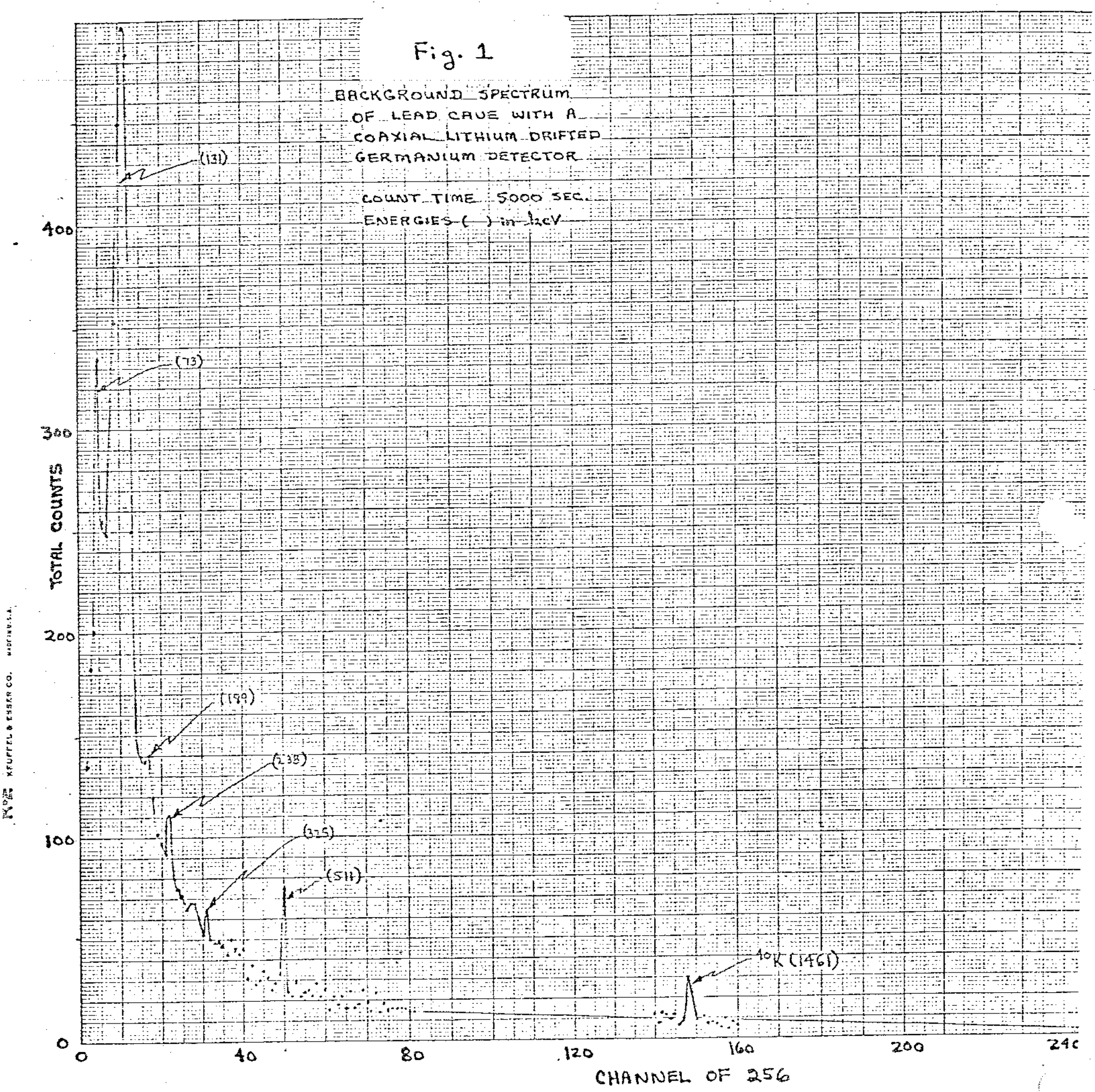


Fig. 1
BACKGROUND SPECTRUM OF LEAD CAVE WITH A COAXIAL LITHIUM DRIFTED GERMANIUM DETECTOR
COUNT TIME 5000 SEC.
ENERGIES ( ) in keV
(131)
(73)
(159)
(238)
(325)
(511)
^{40}K (1461)
TOTAL COUNTS
CHANNEL OF 256
0
100
200
300
400
40
80
120
160
200
240

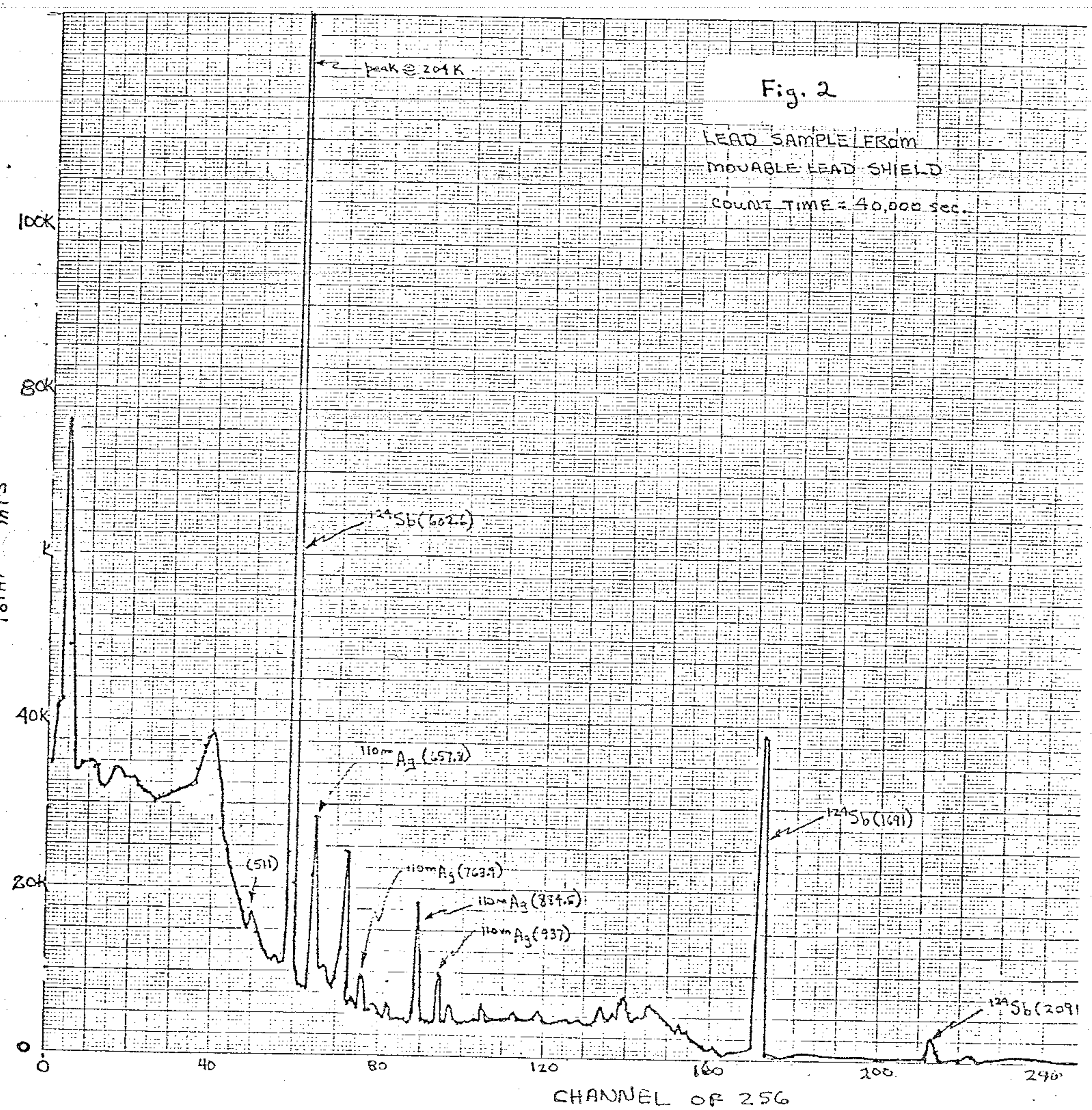


Fig. 2
LEAD SAMPLE FROM
MOVABLE LEAD SHIELD
COUNT TIME = 40,000 sec.
peak @ 204 K
100K
80K
40K
20K
0
Total
ints
^{124}Sb(602.6)
^{110m}Ag(657.8)
(511)
^{110m}Ag(763.9)
^{110m}Ag(884.5)
^{110m}Ag(937)
^{124}Sb(1691)
^{124}Sb(2091
0
40
80
120
160
200
240
CHANNEL OF 256

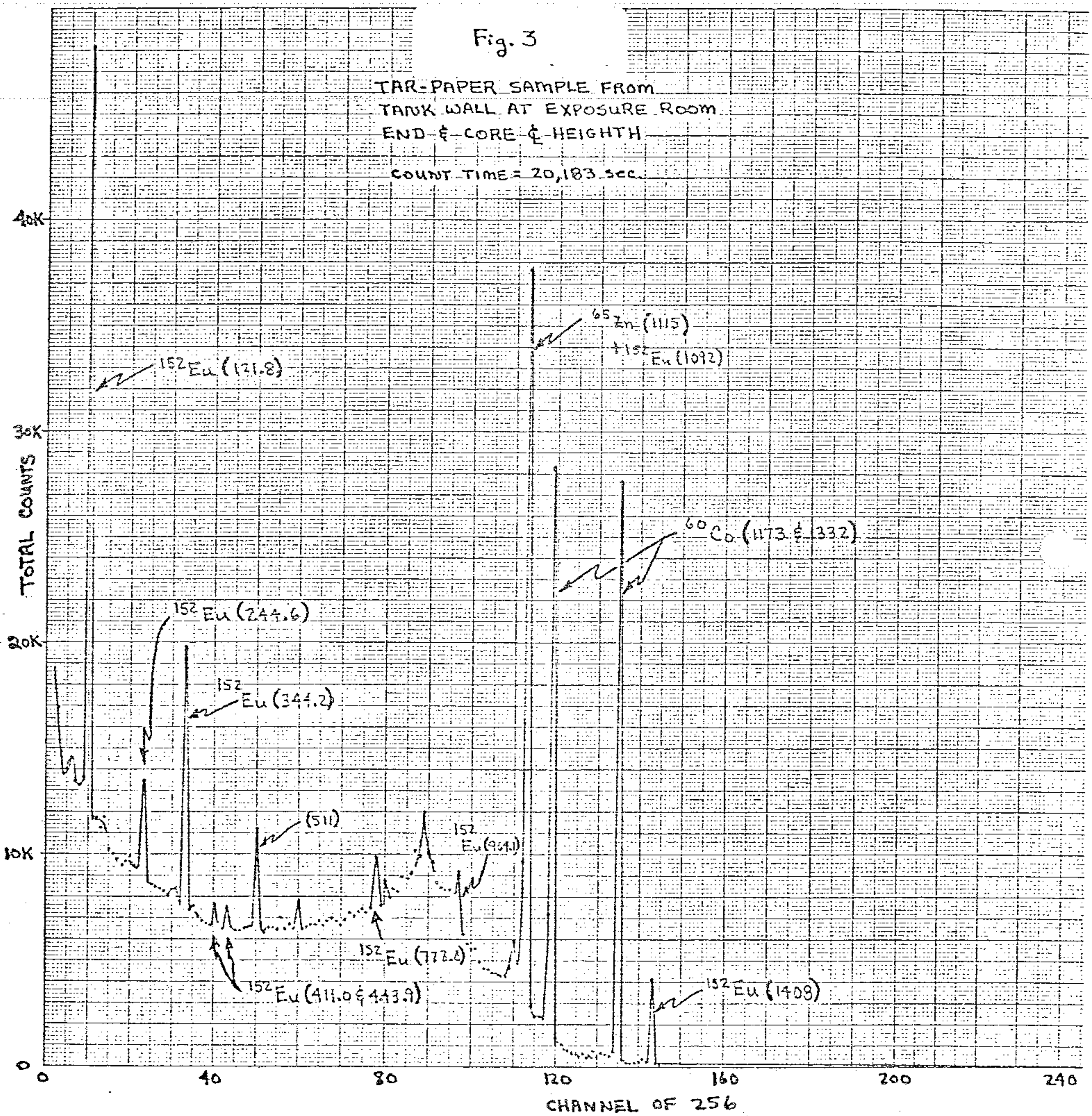


Fig. 3
TAR-PAPER SAMPLE FROM
TANK WALL AT EXPOSURE ROOM
END-&-CORE-℄-HEIGHTH
COUNT TIME = 20,183 sec.
TOTAL COUNTS
40K
30K
20K
10K
0
0
40
80
120
160
200
240
CHANNEL OF 256
152Eu (121.8)
152Eu (244.6)
152Eu (344.2)
(511)
152Eu (964.1)
152Eu (778.0)
152Eu (411.0 & 443.9)
65Zn (1115)
+152Eu (1092)
60Co (1173 & 1332)
152Eu (1408)

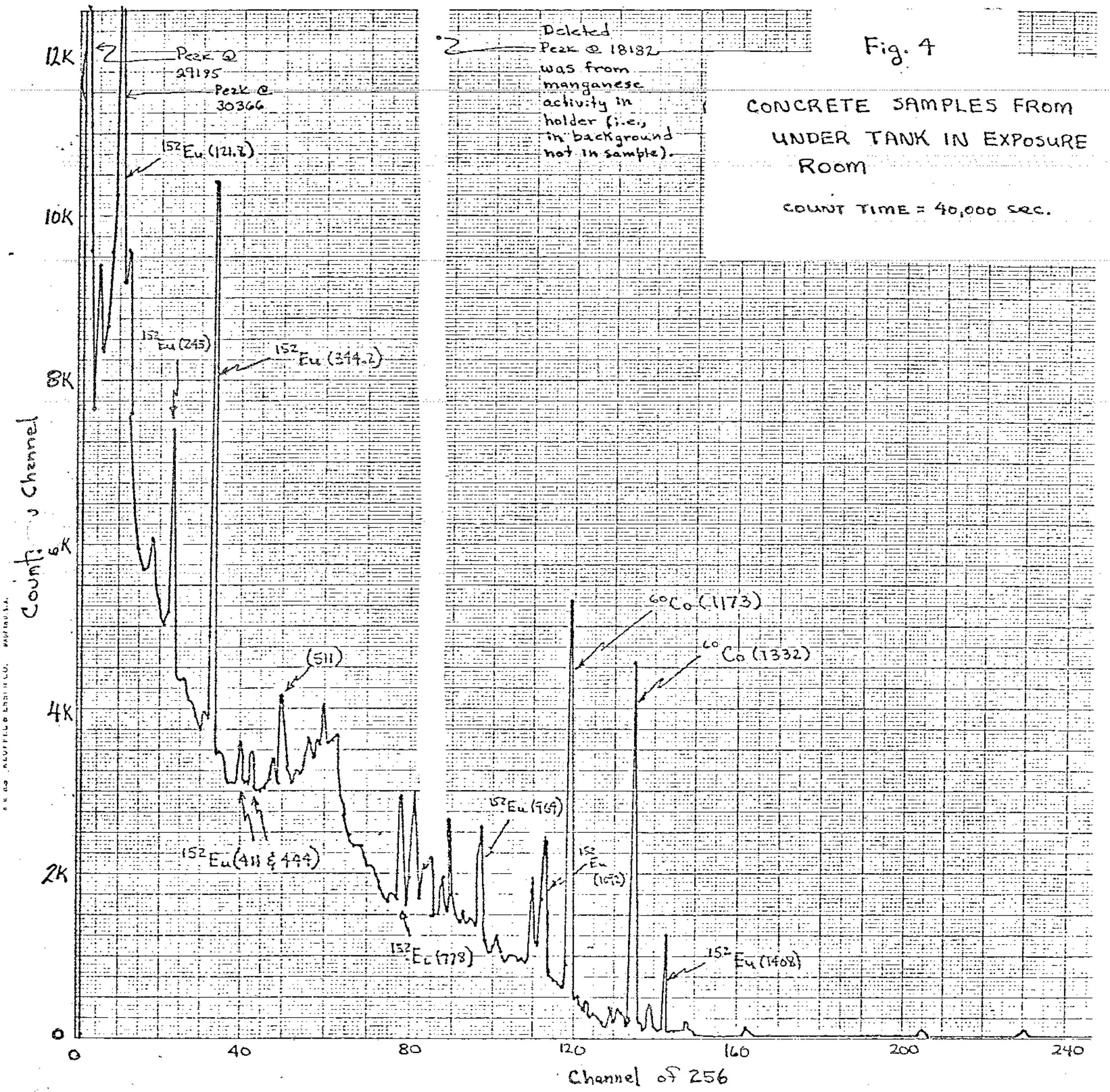


Fig. 4
CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE Room
COUNT TIME = 40,000 sec.
Peak @ 29195
Peak @ 30366
Deleted Peak @ 18182 was from manganese activity in holder (i.e., in background not in sample).
152Eu (121.8)
152Eu (245)
152Eu (344.2)
(511)
152Eu (411 & 444)
152Eu (779)
152Eu (964)
152Eu (1092)
152Eu (778)
60Co (1173)
60Co (1332)
152Eu (1408)
12K
10K
8K
6K
4K
2K
0
Counts / Channel
0
40
80
120
160
200
240
Channel of 256

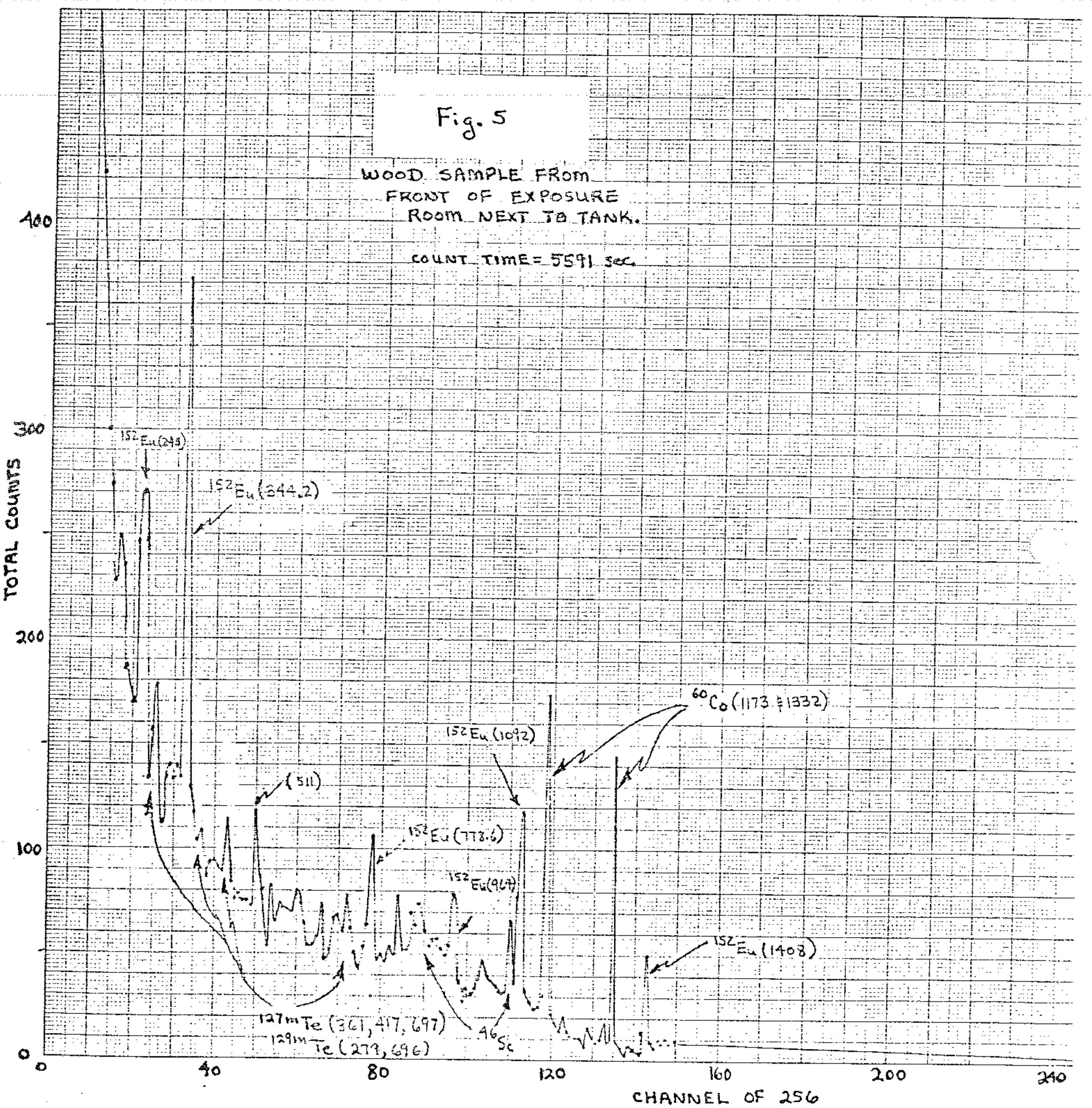


Fig. 5
WOOD SAMPLE FROM
FRONT OF EXPOSURE
ROOM NEXT TO TANK.
COUNT TIME = 5591 sec.
TOTAL COUNTS
CHANNEL OF 256
152Eu(245)
152Eu(344.2)
(511)
152Eu(778.6)
152Eu(964)
152Eu(1092)
60Co(1173 & 1332)
152Eu(1408)
127mTe(361, 417, 697)
129mTe(277, 696)
46Sc
0
100
200
300
400
0
40
80
120
160
200
240

CONCLUSIONS & RECOMMENDATIONS

The principal conclusion from this study is that once the reactor-grid support structure has been removed there is very little radioactivity remaining at DORF. Unfortunately the levels are definitely above background, but only by factors of several hundred, and the radioactivity is mainly distributed throughout concrete walls and floors. Deep excavations will not be necessary. However, this is of little consequence if one still has to remove several inch-thick layers from large areas. This is the situation in the exposure room. In fact, the exposure-room decontamination is by far the major problem and several possible methods of attack come to mind.

(1) Excavate and remove the three 5000-gallon waste-water holding tanks, cut off part of the tops and use them as shipping containers for the radioactive debris from DORF. For example, the wood has suffered radiation damage and dry rot so that it crumbles rather easily. It is a big volume (1200 ft^3) but relatively light in weight so it can easily be tossed or shoveled into the tanks and they could then be closure welded for shipment. There will also be much dust, dirt, paper and small concrete chips of radioactive waste, all of which could be put into the tanks.

(2) Mechanically cut, DO NOT CUT WITH A TORCH, the aluminum because of the radioactive "tar-paper" liner which could easily catch on fire and produce contaminated smoke. However by reference to the excavation-of-concrete details in this report, the places where the aluminum liner will be radioactive are easily identified. It does not appear that the liner will produce a problem in other than these areas.

(3) Thought should be given to the possibility of transferring some of the lead to AFRRI or APRF because its radioactivity is really not a serious hazard and these facilities need it for shielding in neutron fields. This could save a few dollars on transportation and disposal costs.

(4) Survey activities are going to be a problem because there just isn't much activity to survey right now. For example, depending on what is going to be done with the exposure room, it may not be necessary to excavate concrete from the rear wall of the room. In any event, thought should be given to how much the survey reading "from the rear wall only", before excavation, must be decreased by material removal to provide an "acceptable" survey level. In view of the expense to breakup and ship concrete, it is prudent to be practical about sealing up or burying very small, but detectable, amounts of radioactivity.

(5) Almost all of the materials exhibit one or two predominant and characteristic photopeaks. Therefore, survey activities could be determined by a sodium-iodide scintillation detector. It is suggested that a portable detector with a 3/4-inch-thick cylindrical lead shield around the sides would be practical. Calibration could be accomplished in a crude, but adequate, manner by measuring the response of a variety of sources simultaneously positioned over a square-meter plane area behind about 1/4-inch thick aluminum. This approximates the following situation. The dose rate to tissue in rads per hour in an infinite medium, of density ρ, uniformly contaminated by a gamma emitter, of energy E (MeV), is

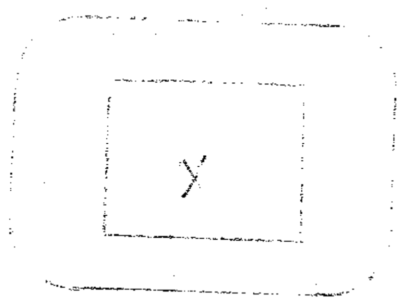
$$2.12\ EC/\rho \qquad (3)$$

where C is in microcuries per cm^3. At the surface, the dose rate is about one half of this and for air a one-centimeter-from-the-surface survey is an adequate representation of the surface rate. By then surveying the "calibration setup at one meter" and correcting for $1/R^2$ to one centimeter, one can estimate the rads per hour efficiency of the scintillation detector. A variety of sources, respositioned should be used and the results averaged.

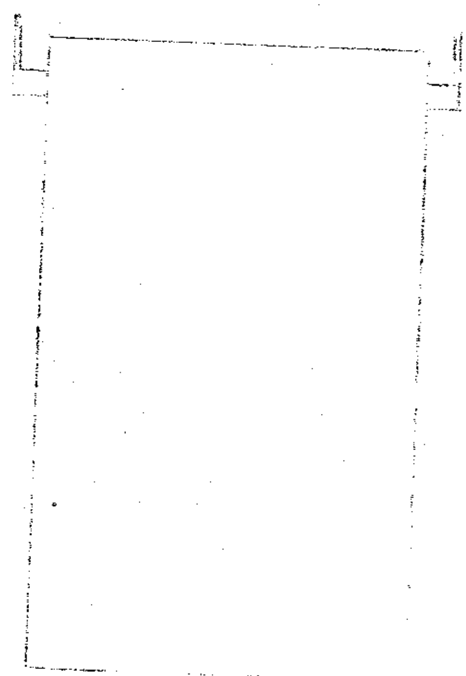
Full Xfer Cask / Affix Xfer Instrument Used: VICTOREEN 740 F 1708

7 total 7	7 mR 8	Lapsed
7 14	9 14	6:50
4 18	15 18	5:00

MEASUREMENTS



x 2.5 R No Shield



HEALTH PHYSICS SURVEY WORK COPY

HP LOG #: DD 8

PRINCIPAL USER: HDL

SURVEYOR: CRAFTON

SECTION:

SAMPLE I.D.:

METER USED:

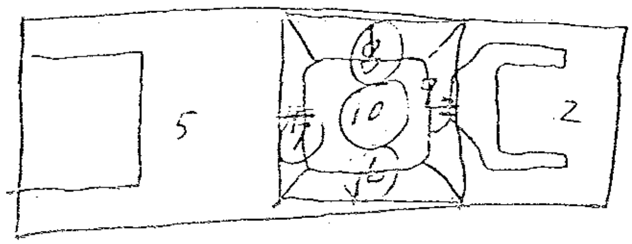
LOCATION: DORF FUEL XFER TRUCK

AUTHORIZATION:

PHONE: 7-5168

TIME & DATE: 14 MAY 79 1600

SAMPLE DESCRIPTION



LABORATORY CHECK LIST
ANNEX P WR 40-10

_____ Airborne hazard

_____ Ventilation

_____ Storage Areas

_____ Waste

_____ Labelling

_____ Monitoring Equipment

SUSPECT ISOTOPE

ANALYSIS DESIRED

CHECK POINT	dpm/100cm^2	CHECK POINT	dpm/100cm^2	REMARKS
1	159932 ± 876	11	< 100	1 UNDER PLUG
2	< 100	12		11 ON FLOOR AT LEAK FROM
3		13		DRAIN LINE
4		14		
5		15		
6	152 ± 28	16		
7	106 ± 24	17		
8	351 ± 42	18		
9	< 100	19		
10	462 ± 48	20		

REPORT OF ANALYSIS APPROVED BY: J Crafton

TIME AND DATE OF COMPLETION:

ANALYSIS PERFORMED AT:

Radioactivity Analysis Laboratory
Health Physics, Bldg 188, Forest Glen
Walter Reed Army Medical Center
Washington, D.C. 20012

WRAMC FORM 708

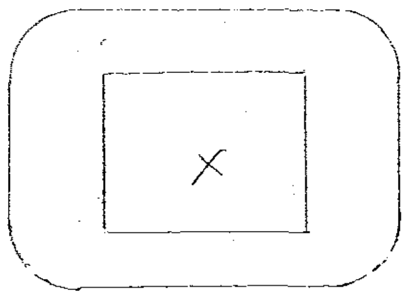
DOES RATE MEASUREMENTS DURING FUEL TRANSFER

A. AT HATCH COVER DURING TRANSFER TO SHIPPING CASK

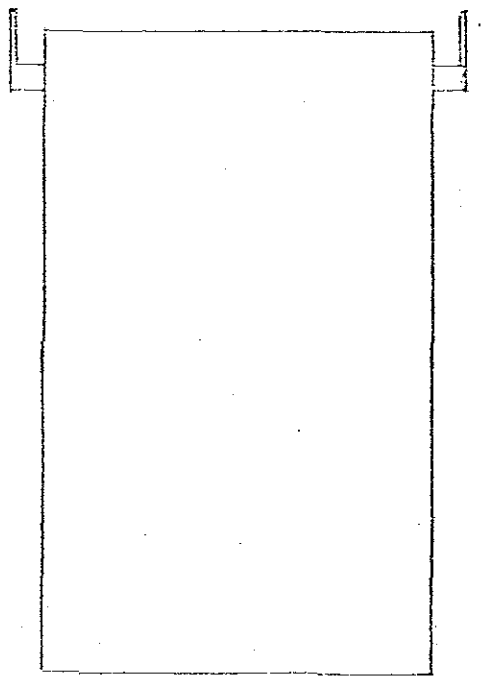
Instrument Used: TELETECTOR

No of Elements	Dose Rate (mr/hr)	Lapsed Time for Transfer
2	<1 / <1	2 MIN
1 TOTAL 3	20 / ≤1	1 MIN 20 SEC
2 5	7 / 2	2 MIN
7 12	11 / 6	5 MIN
7 19	35 / 17	4 MIN 20 SEC
6 25	70 / 35	4 MIN 40 SEC

B. CASK DOSE RATE MEASUREMENTS



× 1.5 R PLUG OUT
0.7 mR PLUG IN



RADIOACTIVE MATERIALS MOVEMENT

☒ SHIPMENT ☐ RECEIPT

For use of this form, see AR 55-55; the proponent agency is Office of the Deputy Chief of Staff for Logistics.

(See instructions on reverse.)

DETAILS OF SHIPMENT

1. TO: (Include ZIP Code)
PENNSYLVANIA STATE UNIVERSITY
COLLEGE OF ENGINEERING
ATTN RE. TOTENBEIR
UNIVERSITY PARK PA 16802

2. FROM: (Include ZIP Code)
HARRY DIAMOND LABS
ADELPHI MD 20783

3. SHIPMENT NUMBER: 4-24-79-1

4. SECURITY CLASSIFICATION: UNCLASSIFIED

5. MODE OF SHIPMENT (i.e., Railway Express): COMMERCIAL CARRIER

6. COMMODITY DESCRIPTION			7. RADIOACTIVITY	8. LEVEL	
CONTAINERS a	NUMBER OF ITEMS b	NOMENCLATURE c	QUANTITY, ISOTOPE AND FORM	AT SURFACE	AT ONE METER
1	18	TRIGA FUEL ELEMENTS	TOTAL ELEMENT U WGT - 3436.71 gms TOTAL ISOTOPE 235U WGT - 687.35 gms 3-83 NCS ACTIVATION PRODUCTS ~2 Ci	015 mR/hr	0.1 mR/hr

...ENT
...E ABOVE DESCRIBED ARTICLES ARE PROPERLY CLASSIFIED, PACKAGED, MARKED, AND LABELED. THE ARTICLES ARE IN ...CONDITION FOR TRANSPORTATION AND THE SPREADABLE ACTIVITY AND DOSE RATES ARE WITHIN THE SPECIFIED LIMITS, ...CRIBED BY APPLICABLE REGULATIONS OF THE DEPARTMENT OF TRANSPORTATION AND DEPARTMENT OF THE ARMY.

...KS
...LE CLASS III SHIPMENT
...OACTIVE YELLOW III LABELS ATTACHED
...CCESSABLE SURFACES OF CASK & TRAILERS
...HAN 150 dpm/100 cm² (See Survey)

...TIONS
...Y SHOWS No spreadable α activity
...N ACCORDANCE WITH DOE & NRC
...AND INSTRUCTIONS

...ICER (Shipping Organization) — DATE 4-24-79

...ping Organization) — GRADE AND TITLE SR — DATE 4-24-79

WRAMC

REPLACES DA FORM 4291, 1 JUN 84, WHICH IS OBSOLETE.
(Paper size, 8" x 10½"; image size, 7-4/10" x 10")

HEALTH PHYSICS WORK PERMIT

NR.

DESCRIPTION OF WORK

Transfer fuel to fuel cask

LOCATION OF WORK

Diamond Ordnance Radiation Facility
Forest Glen, MD

BEGINNING	COMPLETED

NR. OF PERSONS	CREW CHIEF	PH#
	Walter Giesler	7-5168

HEALTH PHYSICS SURVEY

WORK INVOLVES EXPOSURE TO:

[X] EXTERNAL RADIATION [X] CONTAMINATION

RADIATION LEVEL IS ____________ WHICH IS

[ ] ABOVE [ ] BELOW RECOMMENDED LEVELS.

MAXIMUM WORKING TIME	SURVEY BY	DATE
As directed by Health Physics		

HEALTH PHYSICS INSTRUCTIONS

PRIOR TO STARTING WORK:

[X] NOTIFY SP6 Crafton

[X] THE DAY BEFORE WORK IS TO BE STARTED, ARRANGE WITH HEALTH PHYSICS *(Ext. 5107)* TO MONITOR AREA DURING OPERATION.

[ ] ____________

UPON LEAVING RADIATION AREA:

[X] GET CHECKED FOR CONTAMINATION BY HEALTH PHYSICIST.

[X] GET HEALTH PHYSICIST TO CHECK ALL TOOLS AND EQUIPMENT FOR CONTAMINATION.

[X] RETURN MONITORING AND PROTECTIVE DEVICES *(if issued on temporary basis)*.

[ ] ____________

RADIOACTIVE WASTE DISPOSAL

[X] ALL RADIOACTIVE WASTES MUST BE DEPOSITED IN SPECIAL CONTAINERS AND TURNED OVER TO HEALTH PHYSICS. *(See WRAMC regulations on radioactive waste disposal)*.

[ ] NO WASTE PRODUCTS GENERATED.

[ ] ____________

MONITORING AND PROTECTION REQUIREMENTS

[X] BETA-GAMMA FILM BADGE
[ ] NEUTRON FILM BADGE
[X] GAMMA POCKET DOSIMETERS
[ ] NEUTRON POCKET DOSIMETERS
[ ] ____________

[ ] CAP
[ ] HOOD
[ ] ____________

[X] COVERALLS
[ ] WATERPROOF OUTER LAYER
[ ] LAB COAT
[ ] NO PERSONAL OUTER CLOTHING
[ ] ____________

[ ] CLOTH SHOECOVERS
[X] WATERPROOF SHOECOVERS when working
[ ] around cask or on trailer

[X] SURGEONS GLOVES
[ ] AUTOPSY GLOVES
[ ] CANVAS GLOVES
[ ] PLASTICIZED CANVAS GLOVES
[ ] ____________

[ ] HALF MASK
[ ] FULL MASK
[ ] FRESH AIR
[ ] ____________

[ ] ABSORBENT MATERIAL REQUIRED
[X] CAUTION SIGNS AND LABELS
[X] "NO SMOKING" SIGNS
[ ] NO PIPETTING BY MOUTH
[ ] ADDITIONAL SHIELDING
[X] WASTE CONTAINERS
[ ] ____________
[ ] ____________
[ ] ____________

APPROVALS

HEALTH PHYSICS OFFICE

CHIEF OF ACTIVITY CONCERNED

PERMIT EXPIRATION DATE	DATE WORK COMPLETED

SPECIAL INSTRUCTIONS

1. A Health Physics representative must be present during fuel movement.

2. All operations will be in consideration of ALARA.

HSWP-QHP Water Sample Analysis

Radiation Protection Officer
Diamond Ordinance Radiation Fac
WRAMC

Health Physics Officer
WRAMC

29 Nov 79
SFC Hering/acl/75104

The following are the results of the pool water samples taken from your facility:

DATE	SUBSTANCE	ACTIVITY
26 Nov 79	Tritium	5×10^2 pCi/liter = 5×10^{-7} uCi/ml
29 Nov 79	Gross gamma	$\leq$minimum sensitivity = 2×10^3 pCi/liter = 2×10^{-6} uCi/ml

ROBERT M. QUILLIN
LTC, MSC
Chief,
Health Physics Office

V ms = 8.78 x10 pCi/l

K

Atomics International Division
Rockwell International

SUPPORTING DOCUMENT

NUMBER: N001-FDP-960-001

REV LTR/CHG NO: SEE SUMMARY OF CHG

PROGRAM TITLE: Decontamination and Disposition of DORF

DOCUMENT TYPE: Facilities Dismantling Plan

KEY NOUNS: Dismantling Plan, DORF

DOCUMENT TITLE: Facilities Dismantling Plan for DORF, Diamond Ordnance Radiation Facility

ORIGINAL ISSUE DATE:

GO NO.	S/A NO.	PAGE 1 OF TOTAL PAGES 51
04764	20100	REL. DATE 10-23-79

PREPARED BY/DATE: J. M. Harris — 10/23/79 | DEPT: D/731 | MAIL ADDR: T034

IR&D PROGRAM? YES ☐ NO ☒ IF YES, ENTER TPA NO. ______

SECURITY CLASSIFICATION

(CHECK ONE BOX ONLY)

	AEC	DOD
UNCL	☐	☐
CONF.	☐	☐
SECRET	☐	☐

(CHECK ONE BOX ONLY)

RESTRICTED DATA ☐

DEFENSE INFO. ☐

AUTHORIZED CLASSIFIER | DATE

APPROVALS

	Date
W. D. Kittinger	10-22-79
R. J. Tuttle	22 Oct 79
W. R. McCurnin	22 Oct 79

DISTRIBUTION

*	NAME	MAIL ADDR
*	F. H. Badger	T020
*	R. G. Brengle	T143
*	J. H. Brindley	LA34
*	J. W. Carroll	NB02
*	D. J. Elliott	T020
*	J. M. Harris (5)	T034
*	W. D. Kittinger(10)	NB02
*	W. R. McCurnin	T020
*	K. D. Miller	JB02
*	F. C. Schrag	T020
*	R. J. Tuttle	NB13
*	B. F. Ureda	NB02
*	M. E. Remley (2)	NB08
*	J. H. Walter	T006
	H. C. Wieseneck	LA21
	J. R. Rohlfs	LB39
*	D. Empey	KB45

715-M.1/add/sjh

* COMPLETE DOCUMENT

ABSTRACT

The Department of the Army has contracted with Energy Systems Group for the Decontamination and Dismantling of the Diamond Ordnance Radiation Facility (DORF). The reactor is a Triga type with pool, exposure room and support facilities. This plan is prepared to note the techniques and sequences of events to meet the contract requirements of DAAK-21-79-C-0136. The Diamond Ordnance Radiation Facility will be decontaminated for release for unrestricted use per NRC Regulatory Guide 1.86.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

FORM 734-C REV. 3-74

CONTENTS

FIGURES

TABLES

I. OBJECTIVE

The objective for dismantlement and radioactive decontamination of the Diamond Ordnance Radiation Facility (DORF) is to place it in a condition acceptable for release for unrestricted use. Reactor components will be packaged and shipped to the Department of Energy (DOE) at Hanford, Washington. All radioactive materials and components will be removed and decontaminated for release for unrestricted use, or packaged for disposal as radioactive waste and delivered to a licensed burial site. Areas of the facility and materals released for unrestricted use will be decontaminated to levels which are as low as reasonably achievable (ALARA), but in all cases to levels below those described in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86, Table 1. The methodology in decontamination for release for unrestricted use as stated in this guide will be followed. As part of the ALARA program, Rockwell has established the limits shown in Table 2 as their target for compliance with this contract. The limits are based on experience regarding levels that in most cases are reasonable achievable and can be effectively monitored.

TABLE 1

NRC REGULATORY GUIDE 1.86
ACCEPTABLE SURFACE CONTAMINATION LEVELS

Nuclide[a]	Average[b c]	Maximum[b d]	Removable[b c]
U-nat, U-235, U-238 and associated decay products	5000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	100 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm $\beta\gamma$/100 cm^2	15,000 $\beta\gamma$/100 cm^2	1000 dpm $\beta\gamma$/100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objectives of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

TABLE 2

ROCKWELL INTERNATIONAL/ENERGY SYSTEMS GROUP
CONTAMINATION LIMITS FOR DECONTAMINATION & DISPOSAL OF DORF

	TOTAL	REMOVABLE
Beta-Gamma Emitters	0.1 mrad/hr average[a] and 0.3 mrad/hr maximum[b] at 1 cm with 7 mg/cm^2 absorber	100 dpm/100 cm^2
Alpha Emitters	100 dpm/100 cm^2	20 dpm/100 cm^2

[a]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[b]The maximum contamination level applies to an area of not more than 100 cm^2.

II. SCOPE OF PLAN

The scope of this Dismantling Plan is to delineate the activities necessary to realize the objectives stated in Section I. The activities are categorized as: planning, monitoring, and control; radiological survey; dismantlement and disposal; and documentation.

III. PLANNING, MONITORING, AND CONTROL

The activities which comprise the dismantlement of DORF will be initiated, monitored and controlled by the Rockwell Site Manager at DORF. The site manager will also have the overall technical responsibility for the dismantling activities and will be the onsite interface for all contacts with the Army's Site Contracting Officer or his representative. The DORF D&D organization structure is shown in Figure 1. The Rockwell Radiation and Nuclear Safety representative will be responsible for radiological surveys and survey data analyses. Records of significant radiation surveys and analyses will be made available to the Contracting Officer or his representative.

A schedule listing the specific tasks and the proposed sequence for performance is presented in Table 3. The estimated level of manpower and milestones for these activities are included for information. The milestone schedule will serve as the criteria to measure progress in dismantling DORF.

The Operational Safety Plan on Decontamination and Disposal at DORF is attached as Appendix A. It contains the radiation safety, industrial hygiene, and industrial safety procedures in support of the activities described in this Dismantling Plan.

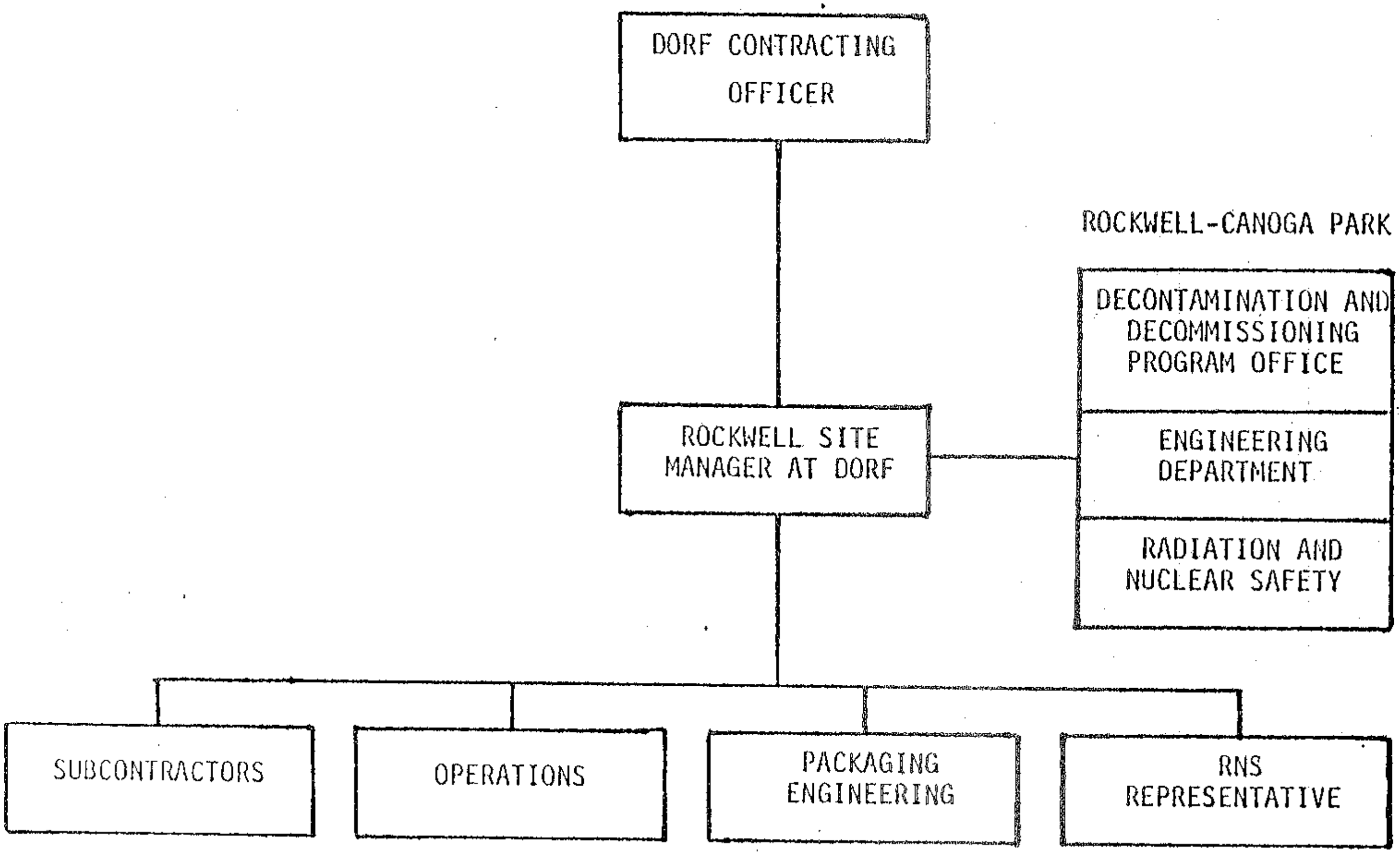
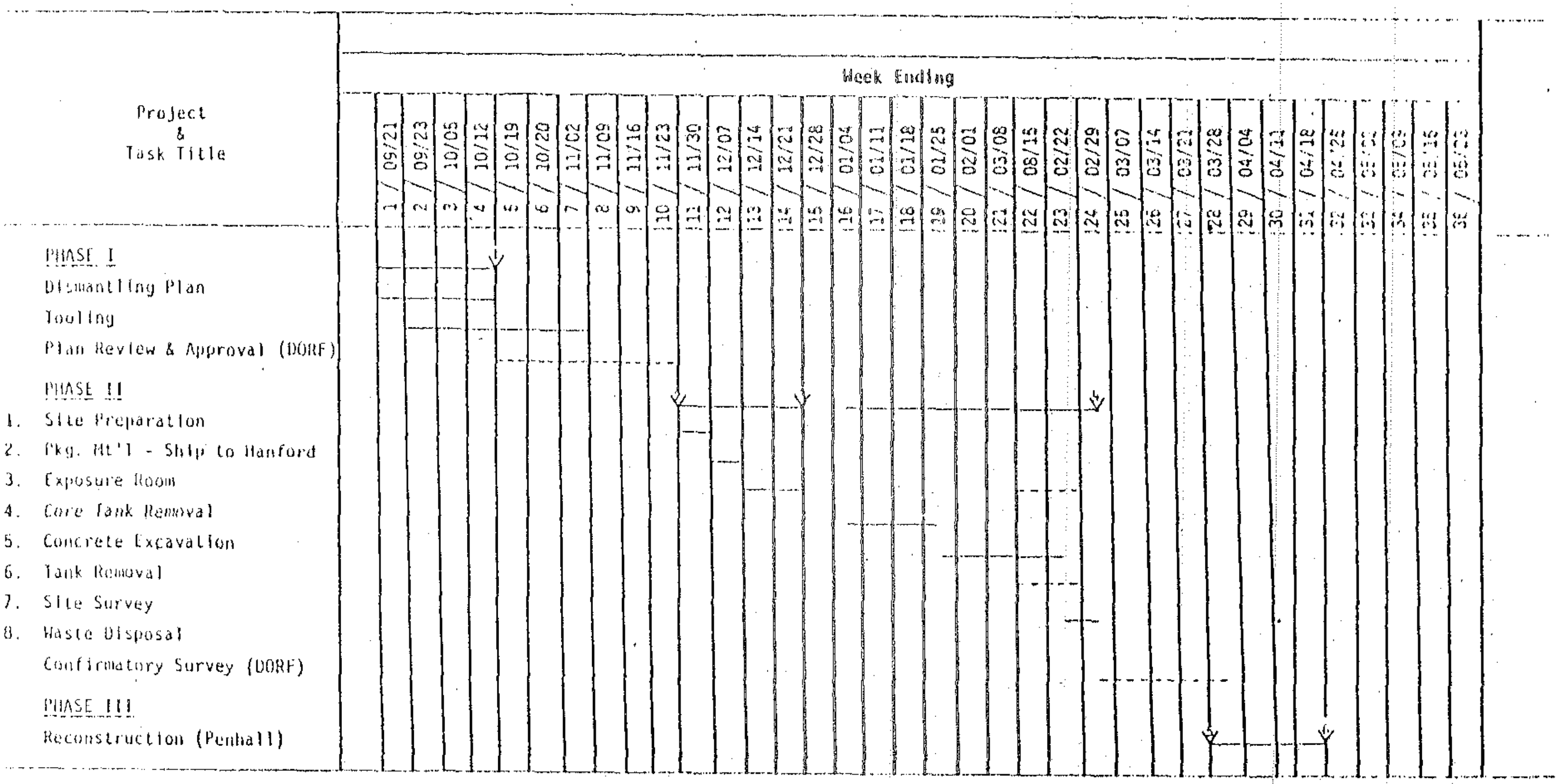


FIGURE 1. DORF D&D ORGANIZATION CHART

TABLE 3

DORF D&D PROJECT SCHEDULE



Milestone 1 - Complete Phase I, 10/14/79
Milestone 2 - Start Phase II, 11/26/79
Milestone 3 - Reactor Components Shipped to HEDL 12/21/79
Milestone 4 - Complete Phase II, 2/26/80
Milestone 5 - Start Phase III Within 30 Days of Survey Acceptance
Milestone 6 - Complete Phase III

IV. RADIOLOGICAL SURVEY

A radiological survey will be made to assess the extent of radioactivity present in the facility. This assessment will include the grounds which surround the facility to establish and record conditions at the site before beginning the dismantling activities.

Radiological surveys will be conducted before and during the Phase II work only to provide information for guidance in determining (1) what areas are radioactive, (2) when sufficient material has been removed to release these areas for unrestricted use, and (3) personnel surveillance.

The comprehensive radio-isotopic analysis appended to the Request for Quotation (RFQ) as Appendix IV was an estimate of the radioactive material remaining in the DORF structure as of the spring of 1978. This will serve as a guide and will be used as a qualitative indication of the presence of radioactivity.

V. DISMANTLEMENT AND DISPOSAL

The scope of work required to dismantle DORF is presented, followed by a description of the principal tasks required to accomplish the Phase II and III work defined in the RFQ. The tasks will be performed in the order shown in Table 3 if practicable. Overlap of the schedule tasks will occur as required to maintain continuity in the overall program.

A. DISMANTLEMENT SCOPE OF WORK

Activities required to accomplish the dismantlement of DORF include: (1) surveying and recording the radiological condition of the facility and surrounding grounds to define the existing condition; (2) the analysis and disposal of the core tank water per 10 CFR 20 limits; (3) the removing, packaging and shipping of the reactor components listed in Table 4 of this plan; (4) removing, packaging, and shipping to a licensed burial site the radioactive materials and components referenced in the RFQ and those generated during the dismantlement of DORF; (5) removing and disposing of the nonradioactive components or materials listed in Paragraph F.4.1 (a through g) of the RFQ (Note that (h) is included in Activity 3 above); (6) removing and delivering the jib-crane to the AURORA facility; and (7) the Health Physics support necessary to assure compliance with NRC Regulatory Guide 1.86 and 10 CFR 20.

B. PHASE II

1. Site Preparation

The site preparation task includes those activities required to move the ESG staff and their equipment to the site and to establish a base of operations. A radiation survey of the nonradioactive portions of the site will be conducted for documentation. An analysis of the pool water to determine compliance with 10 CFR 20 will be performed.

TABLE 4
REACTOR COMPONENTS FOR SHIPMENT TO
DOE-HEDL, RICHLAND, WASHINGTON

Item No.	Description	Unit	Quantity
1	Core Support Structure, Upper Section	Each	1
2	Core Support Structure, Lower Section	Each	1
3	Top and Bottom Grid Plates	Each	1
4	Connecting Rods for Control Rods	Set	1
5	Control Rods	Set	1
6	Carriage Drive Motor	Each	1
7	Water Pump: 1.5 hp	Each	1
8	Incore Experiment Tube	Each	1
9	Ion Chamber Supports and Ion Chambers	Set	3
10	Carriage Support Rails	Set	1
11	Lead Shield Door Drives and Linkage	Set	1
12	Pool Cover Plates	Set	1
13	Fuel Storage Racks, Underwater	Each	8
14	Fuel Measurement Tool with Dial Micrometer	Each	1
15	Aluminum Water System Piping	Each	1
16	Water Pumps	Each	3
17	Demineralizers, 3 ft^3	Each	4
18	Flowmeters, 25 gpm	Each	2
19	Neutron Source, 10 curies, am-be	Each	1
20	Neutron Source Holder	Each	1
21	Pool Lights	Set	1
22	Carriage Positioning Potentiometer	Each	1
23	Carriage Umbilical Arm	Each	1
24	Fuel Element Location Diagram	Each	1
25	Water Box, 1 ft^3 Capacity	Each	1
26	Charcoal Filter, 1 ft^3 Capacity	Each	1

The pool water will be discharged to the sanitary sewer as analysis permits. The appropriate limits are those listed in 10 CFR 20, Appendix B, Table 1, Column 2, as provided by 10 CFR 20.303. A radiation survey of the reactor components that are scheduled for shipment to HEDL will be conducted. Should the water analysis show contamination above limits, the existing purification system will be used for cleanup.

2. Packaging and Shipping Reactor Components to HEDL

The electrical service for the reactor auxiliary systems will be disconnected from the relay and power distribution panels and the wiring will be removed. This will include power disconnects to the lead shield doors, carriage drive, and the diffuser pump.

All of the items listed in Table 4 will be removed, packed into weatherproof containers, and staged for transportation to the DOE, Hanford Engineering Development Laboratory (HEDL), Richland, Washington. The Americium-Beryllium neutron source will be placed in an approved Type A shipping container for shipment with the above items.

The water treatment system in the Filter Room will be removed after the pool water has been discharged to the sanitary sewer and a determination has been made that it will no longer be required. Water will be drained from the piping and filter medium and the water will be dispositioned based on radiation survey analysis. All of the items listed will be shipped to HEDL when they have been packaged and are available for shipment. Components will only be disassembled to the degree necessary to permit packing into reasonably sized containers.

All packaging will conform to the Department of Transportation (DOT) Specification Title 49 Code of Federal Regulations. Each package shall be monitored by the Health Physicist to determine its radioactive content and will be weighed to establish its shipping weight.

When all radioactive components have been removed from the facility, the areas which housed those components will be radiologically surveyed and the survey documented. Those areas which are above the release limits will be identified on a facility plot plan and scheduled for removal during the appropriate demolition task.

The jib crane will be removed from DORF and transported to the AURORA facility when it is no longer required to support dismantling activities.

3. Exposure Room

The exposure room will be stripped of its wood lining, lead shields, lead shield hoist, and other removable components. The material will be separated and dispositioned either to salvage or packaged for radioactive disposal.

Before starting activities in the exposure room, the floor drains will be plugged to reduce the potential for transporting radioactive materials into the sanitary sewer system.

The wood timber lining will be removed from the room using conventional techniques. Each timber will be surveyed to determine radioactivity and will be dispositioned according to Table 2 criteria. Material that is activated to levels that exceed Table 2 limits will be packed in strong, tight shipping containers, while material that is not activated will be set aside for salvage. The lead shields will be removed, surveyed and set aside for disposition. The lead shield hoist will be removed in its entirety and packed in a strong, tight shipping container. All other removable components will be removed from the room and will be dispositioned accordingly. When all removable material has been dispositioned, the exposure room will be vacuumed to remove remaining residue from the surfaces.

A detailed radiation survey of the exposure room will be conducted to establish a mapping of activity in the concrete. Selected areas will be sampled by core drilling to establish the extent of activation. The exposure room door and doorway will be included in the survey analyses. An excavation plan will be developed, for implementation during concrete excavation detailed in Section 5.

4. Core Tank Removal

All extraneous structures will be removed from the core tank, the lead shield doors will be drained of lead, the lead and doors will be removed from the core tank, the core tank will be stripped from the concrete, and the activated tar paper lining will be removed from the concrete surfaces. The materials will be surveyed when they are removed from the area and will be dispositioned accordingly.

The procedure to remove the lead shield doors will consist of drilling holes through the lower wall of each door to drain enough lead to permit them to be lifted with the 3.5-ton overhead crane. A dynamometer will be used to provide assurance that the weight of the load is within the crane limit. Each door will be lifted from the core tank and transferred to an area where the remaining lead can be removed. The doors and lead will be surveyed and dispositioned accordingly.

The procedure to remove the core tank will be to section the tank (by saw cutting) into vertical strips. Each strip will be pulled from the concrete by conventional techniques as determined by experience gained during the first and subsequent removal attempts. Because of the uncertainty associated with the adhesion of the tank to the concrete by virtue of tar paper, trial and error will be required. Leverage tools such as pry-bars, wedges, block and tackle, etc., will be used initially. If these techniques prove unsuccessful, then hydraulic or pneumatic techniques will be applied.

The activated tar paper lining will be removed from the concrete by scraping and/or by chipping away portions of the concrete. Where the concrete is also activated, the tar paper will be left for removal with the concrete.

All activated and contaminated materials will be packaged for disposal as detailed in the Waste Disposal section. Material will be size reduced, where practical, to reduce the volume of radioactive waste.

A detailed radiation survey will be conducted of the exposed concrete structure to establish a map of radioactivity. Selected areas will be sampled by core drilling to establish the extent of the activation. An excavation plan will be developed for the concrete structures. Implementation of the plan will be described in Section 5, "Concrete Excavation."

5. Concrete Excavation

Concrete will be removed from the pool cavity, exposure room, and exposure room door to the extent required to permit release of these structures for unrestricted use. Guidance for the amount of concrete to be removed will be determined by radiation survey and by the excavation plans developed after core tank removal and exposure room cleaning. When the detectable levels of radioactivity in the concrete are below the levels shown in Table 2, they will be considered to be in compliance with NRC Guide 1.86 (Table 1).

The extent of removal will be governed by the extent to which the structures are activated. Where activation is shallow, scabbling or chipping with pneumatic hammers will be used to break the concrete at the surface. High volume vacuum cleaners equipped with HEPA filtration will be used to remove the concrete and to control airborne contamination. Where penetration is several inches deep, jack hammers will be used. This operation will be aided by depth cutting with a concrete saw if necessary. If depth of activation is such that these techniques are not applicable, a hydraulic ram hoe or other devices will be used to break the concrete for removal.

Dust and particulate generation will be monitored by the Radiation and Nuclear Safety representative and control will be accomplished by use of a vacuum cleaner or water mist depending on the operation in progress. High volume air sampling will be conducted within the work area during operations which might produce airborne contamination. Personnel will be required to wear respirators whenever sampling indicates unacceptable levels of airborne contamination. Temporary structures will be built around the work area if necessary to control the spread of contamination.

Radiation survey data generated during the activated concrete excavation will be analyzed to provide a basis for compliance with Regulatory Guide 1.86 and ALARA (Tables 1 and 2). When the data indicates that compliance with these criteria have been met, concurrence will be solicited from the U.S. Army Environmental Health Agency.

6. <u>Site Survey</u>

A final radiation survey will be conducted to verify the site condition. Surface smears and material samples will be selected by the Rockwell Site Manager with assistance from the Radiation and Nuclear Safety representative and the DORF Contracting Officer. These specimens will be sent to an independent laboratory for analysis. The specimens will be taken from representative areas of the buildings and excavations to confirm compliance with ALARA and Regulatory Guide 1.86. These data will provide independent analyses of the site condition and will form a basis for demonstrating that the facility can be released for unrestricted use.

7. Waste Disposal

Radioactive waste will be packaged when it is generated and will be staged in full load lots (~40-45,000 lb) for shipment. Shipments will be made under the exclusive use provision of Title 49 Code of Federal Regulations which permits low specific activity (LSA) waste to be packaged for shipment in strong, tight containers. Shipments will be monitored by the Radiation and Nuclear Safety Representative for conformance to DOT regulations. Radioactive waste will be delivered to a common carrier for delivery to a licensed disposal site as full load lots become available, or at the completion of Phase II.

Noncontaminated components and waste materials listed in the RFQ for disposal (F.4.1) will either be retained and used as backfill in the pool cavity or hauled to a local licensed landfill during Phase III.

C. PHASE III

A concrete wall will be erected between the exposure room and the pool cavity to provide a barrier between the two areas so the pool cavity can be used to hold backfilled concrete debris. The steel ramp structure and concrete parapet surrounding the pool cavity will be dismantled. The steel structures will be removed for salvage and the concrete support wall will be broken up and the debris will be placed into the pool cavity. The debris shall not be filled to a level above the main floor.

The air conditioning system inlet and exhaust ducts to the exposure room will be restored if necessary and made operable. Where practical, all electric outlets, air, water, and sewer lines will be retained in working order during the dismantling activities.

At the completion of Phase III, the accumulated demolition debris not deposited into the pool cavity will be removed from the facility area. The contractor's equipment will be removed and a reasonable effort will be made to clean the interior of the facility building of accumulated dirt and dust resulting from the demolition efforts.

VI. DOCUMENTATION

Documentation of the Dismantlement of DORF will consist of informal weekly progress reports, radiation survey reports, and a final report.

The informal weekly progress reports are primarily written to the Rockwell home office in Canoga Park, California, to keep them informed of the DORF site operations and to alert them to any changes that may impact schedule or cost. Copies of these reports will be sent to the Contracting Officer or his representative.

Radiation survey reports of significant data will be entered on a Rockwell Form 732-A, Health & Safety Analysis Report for distribution within the Rockwell organization. Copies of these reports will be sent to the DORF Contracting Officer or his representative.

At the completion of Phase II, a final report will be written to document the dismantlement of DORF. The report will describe the activities required to accomplish the work, problems encountered, solutions to the problems, and the current status of the facility and structures. The report will also include data, specified in units identical to those shown in Table 1, to show the effort made to reduce residual contamination to levels that are as low as reasonably achievable. It will describe the scope of the radiation survey, the general procedures followed to obtain the data, and any other pertinent information about the radiation survey data. A summary of the radioactive waste disposal information will be included to show the quantities of material removed from DORF.

The final report will be appended at the completion of Phase III to document the final status of the facility at the termination of this contract.

APPENDIX A
OPERATIONAL SAFETY PLAN
FOR DECONTAMINATION AND DISPOSAL OF DORF

I. PURPOSE

To delineate the radiation safety, industrial hygiene, and industrial safety procedures for the decontamination and disposition (D&D) of the Diamond Ordnance Radiation Facilities (DORF).

II. SCOPE

This plan applies to all operations at DORF involving the deactivation, dismantling, decontamination, and disposal of that nuclear facility.

The plan meets or exceeds the requirements set forth in Rockwell/ESG Standard Operating Policies, in applicable regulations and standards, in DOEM 0524, in the Williams-Steiger Occupational Safety and Health Act of 1970 (OSHA), 10 CFR 19 and 20, and in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86.

III. RESPONSIBILITIES

A. HEALTH SAFETY AND RADIATION SERVICES

1. Radiation and Nuclear Safety

Radiation and Nuclear Safety shall establish requirements for design and operational procedures; and approve disposition of source and special nuclear material, and byproduct radioactive material.

Radiation and Nuclear Safety will designate and identify radiologically posted areas, radiological safeguards requirements, and radioactive materials; and will control the use and disposition of radioactive materials, and implement radiological safety standards.

Radiation and Nuclear Safety will perform field measurements of radiation and radioactive contamination levels, evaluate internal and external personnel radiation exposures, and evaluate radioactive material in the workplace environment.

Radiation and Nuclear Safety will maintain records as necessary to demonstrate compliance with ESG standards and applicable regulations. Included in these records will be a chronological log of information dealing with daily operations, conditions, and occurrences relating to radiological safety.

Radiation and Nuclear Safety will advise the Rockwell Site Manager and operations personnel on the safe performance of their assigned tasks.

Radiation and Nuclear Safety will evaluate operational conditions to determine requirements for personnel monitoring and protective devices such as film badges, breathing zone air samplers, bioassay, protective clothing, and respiratory protection devices.

The Radiation and Nuclear Safety (RNS) representative assigned to the project will conduct the radiological surveillance program and will maintain sufficient familiarity with program operations and facility conditions to be aware of those areas which may require increased surveillance or corrective action.

2. <u>Industrial Hygiene and Safety</u>

Radiation and Nuclear Safety will provide the services necessary to control personnel exposures to toxic chemicals and harmful physical agents and to control mechanical and electrical hazards. RNS Representative will maintain surveillance of the occupational environment to identify, evaluate, and control conditions pertinent to health and safety, and to assure compliance with the requirements of DOE Manual, OSHA, 10 CFR 19 and 20, as appropriate.

B. ROCKWELL SITE MANAGEMENT AT DORF

The Rockwell Site Manager is responsible for the safety of all personnel within facilities under the jurisdiction of the DORF D&D Program.

The Site Manager will ensure that all personnel employed at or visiting the facility know and understand the rules and regulations governing work with radioactive materials and will assure compliance with these rules. The Site Manager will carry out the responsibilities charged to "Operating Supervision" and will provide safe conditions at the facility, in conformance with applicable regulations and standards, and under the guidance of Radiation and Nuclear Safety.

The Site Manager will establish the requirements for the packaging of radioactive waste, collecting of packaged waste, and will arrange for disposal by land burial.

Rockwell Program Management and the Site Manager will coordinate radiological and industrial hygiene and safety problems with Radiation and Nuclear Safety as appropriate.

C. OPERATIONS PERSONNEL

Operations personnel are responsible for compliance with all rules governing work with radioactive and hazardous materials as outlined by this procedure and as established by Radiation and Nuclear Safety and D&D Program Management. Operations personnel are responsible for taking every reasonable precaution to minimize radiation exposures to themselves and to fellow workers and to prevent the unnecessary release of radioactive material.

D. CONTRACTOR PERSONNEL

Contractor personnel are responsible for compliance with all safety rules and requirements established by Radiation and Nuclear Safety and for responding to specific instructions from the RNS Representative with regard to radiation safety and industrial hygiene.

IV. ADMINISTRATIVE SAFEGUARDS

A. PROCEDURAL CONTROL

Any changes to the radiation safety or industrial hygiene and safety procedures must be jointly authorized by Radiation and Nuclear Safety and the Site Manager following evaluation of the proposed changes by the RNS Representative. Revised procedures will be distributed to all personnel directly affected by the change.

Operations involving potential radiological hazards or potential industrial safety hazards will be reviewed in advance by Radiation and Nuclear Safety.

B. METHODS OF REPORTING DAY-TO-DAY CONDITIONS

Day-to-day operational safety conditions will be observed by the assigned Radiation and Nuclear Safety representative, who will report all recognized hazardous conditions and each instance of noncompliance with regulatory directives to the Site Manager and the workers involved. Radiological data (film badge and bioassay results, radiation and contamination survey results, air sampling reports, etc.) will be maintained by the RNS Representative. Whenever these data indicate the need for corrective action, the RNS representatives will contact the Site Manager to arrange for such action. Industrial hygiene and safety conditions observed by the RNS representative will also be communicated to the Site Manager. A summary of incidents and data will be reported to the Site Manager, the Radiation and Nuclear Safety Office, and the Rockwell Program Management Office on a weekly basis.

V. GENERAL RADIATION AND INDUSTRIAL HYGIENE AND SAFETY PROCEDURES

Certain radiological and industrial safety controls and procedures are independent of operations in the facilities, and are required to provide facility surveillance and radiological and industrial safety protection commensurate with the ESG contract and regulatory agency standards.

A. AREA DESIGNATION, RADIOLOGICAL SAFETY CONTROL

All areas are designated as either radiologically posted or unposted. A posted radiological area is an area, defined by physical barriers, which is posted with prescribed caution signs or labels for purposes of radiation protection. Signs used to designate posted radiological areas must comply with applicable regulations. There are six posted area classifications as defined below:

1. Radiation Area

A Radiation Area is an area subject to radiation from encapsulated radioactive materials and/or radiation machines within the area, or to radiation from any source outside the area; where there exists radiation at such levels that an individual could receive in any one hour a dose to the whole body in excess of 5 millirem, or in any five consecutive days a dose in excess of 100 millirem.

Each Radiation Area will be posted with a sign meeting all regulatory requirements including the radiation symbol and the words "CAUTION - RADIATION AREA." Where appropriate, indications of the radiation level will be included in the area posting.

2. Radiation Area - Radioactive Contamination

A Radiation Area - Radioactive Contamination is an area in which work with and/or storage of unencapsulated material is permitted with the provision that the radioactive material concentration in air is not likely to exceed 25% of the appropriate occupational exposure limit. Each Radiation Area - Radioactive Contamination will be posted with signs meeting all applicable regulatory requirements including the radiation symbol and the words "CAUTION - RADIATION AREA - RADIOACTIVE CONTAMINATION."

3. Radiation Area - Airborne Radioactivity

A Radiation Area - Airborne Radioactivity is an area in which the radioactive material concentration in air is likely to exceed 25% of the applicable regulatory standard for occupational exposure.

Each Radiation Area - Airborne Radioactivity will be posted with signs meeting all applicable regulatory standards including the radiation symbol and the words "CAUTION - RADIATION AREA - AIRBORNE RADIOACTIVITY."

4. High Radiation Area

A High Radiation Area is an area accessible to individuals in which there exists radiation at such levels that an individual could receive in any one hour a dose to the whole body in excess of 100 millirem. Each High Radiation Area will be posted with signs meeting all applicable regulatory requirements including the radiation symbol and the words "CAUTION - HIGH RADIATION AREA."

5. Radiation Area - Radioactive Materials

A Radiation Area - Radioactive Materials is an area in which work with and/or storage of encapsulated materials is permitted.

Each Radiation Area - Radioactive Materials will be posted with signs meeting all applicable requirements, including the radiation symbol, and the words "CAUTION - RADIATION AREA - RADIOACTIVE MATERIALS." Federal and State regulations also require that storage containers and localized areas in which radioactive materials are present in certain amounts will be posted with signs containing the radiation symbol and the words "CAUTION - RADIOACTIVE MATERIALS." It should be noted that these containers and areas may or may not be located within posted areas. Radiation and Nuclear Safety will advise operating supervision as to the amounts of radioactive materials in containers or localized areas which require such signs.

6. Restricted Access Area

A Restricted Access Area is an area identified by Radiation and Nuclear Safety as requiring special safety precautions for entry and requiring inspection immediately prior to entry by any person. Each

Restricted Access Area will be posted with signs with the following words in yellow over a red background:

"WARNING - RESTRICTED ACCESS AREA - OBTAIN PERMIT
FROM OPERATIONAL SAFETY PRIOR TO ENTRY"

Any area meeting more than one of the above criteria will be posted with all of the applicable signs.

B. AREA DESIGNATION, INDUSTRIAL SAFETY CONTROL

Operations posing potential hazards shall be identified by appropriate caution or warning signs. The signs shall conform to specifications in 29 CFR 17, Section 1910.145. Examples of posting are:

1. <u>Hard Hat Area</u>

A Hard Hat Area will be established wherever personnel are working at different elevations and there is a potential of being hit by falling objects.

2. <u>Eye Protection Area</u>

An Eye Protection Area will be established where a hazard due to flying objects exists.

3. <u>No Smoking Area</u>

No Smoking Areas will be established where explosives, flammable liquids, or gases may be present.

4. Open Excavations

Open Excavations will be protected by appropriate physical barriers.

5. Obstructions

Obstructions will be made clearly visible by the use of yellow and black striping.

C. RADIOLOGICAL SURVEY FREQUENCY

Routine radiation and contamination surveys will be performed in work areas at a frequency to be determined by the assigned Radiation and Nuclear Safety representative in accordance with established procedures. Additional surveys may be required to determine the effectiveness of contamination control procedures. The requirement for these surveys will be established on the basis of initial experience with those tasks which may pose significant personnel radiation or airborne contamination exposure.

D. RADIOACTIVE CONTAMINATION LIMITS

Evaluation of levels of radioactive contamination will be required in order to determine:

1) The adequacy of the level of decontamination performed on the facilities;
2) The extent of required excavation or other demolition of activated structures; and
3) The disposition of equipment, materials, and scrap.

Facilities and equipment will be evaluated for removable and total (fixed plus removable) contamination by means of wipe surveys and instrument surveys. Activated structures will be evaluated for radioactive concentrations by sampling or surveying with detection instruments.

Removable contamination limits for radiologically posted and unposted areas are described in Table A-1. The upper limit of allowable contamination listed in the table is that level which, if reached, requires immediate cessation of operations, immediate decontamination must be effected and measures taken to prevent recurrence. The action limits specified in Table A-1 are the upper limits of the amount of general area contamination tolerable in posted and unposted areas. General contamination in an area in excess of the action limit requires prompt decontamination.

TABLE A-1
REMOVABLE CONTAMINATION LIMITS
(dpm/100 cm^2)

Area	Activity	Upper Limit	Action Limit
Unposted Areas and Radiation Areas	Beta	1,000	100
	Alpha	200	20
Contamination Areas Airborne Radioactivity Areas	Beta	50,000	5,000
	Alpha	20,000	200
Restricted Access Areas	Beta	Not Defined	Not Defined
	Alpha	Not Defined	Not Defined

The levels of contamination which will be considered acceptable for unconditional release of equipment or facilities are as follows:

Removable Contamination

20 dpm/100 cm^2 alpha

100 dpm/100 cm^2 beta-gamma

Total Contamination

100 dpm/100 cm^2 alpha

0.1 mrad/hr beta-gamma measured through 7 mg/cm^2 absorber at 1 cm

Water

3×10^{-7} μCi/ml beta-gamma

3×10^{-8} μCi/ml alpha

Soil (If Subject to Contamination)

100 ρCi/g gross detectable beta-gamma

10 ρCi/g alpha

Where practicable, items may be decontaminated to levels lower than the acceptable limits.

During demolition activities, all scrap generated will be evaluated for radioactive contamination prior to release to normal waste channels or packaging for disposal by land burial.

E, SURVEY REPORTS

The original copy of radiation, contamination survey, and special radioanalysis reports will be forwarded promptly to the Site Manager and Radiation and Nuclear Safety supervision. These reports will indicate contamination and radiation levels at specific locations throughout the facility. Copies of these survey reports will be retained indefinitely by Radiation and Nuclear Safety.

Radiation and Nuclear Safety will post or have posted such signs as are necessary for the clear identification of potential radiological hazards. To assure that the posting of radiological hazards is current, periodic surveys will be conducted by the Site Manager and the RNS Representative. Signs which have been approved by Radiation and Nuclear Safety will be used to indicate radiological hazards in the facility. No such signs will be removed without the approval of Radiation and Nuclear Safety. In addition, warning signs relative to hazardous conditions and/or special safety requirements may also be posted.

G. FACILITY VENTILATION

The DORF facility ventilation systems will be used to control airborne contamination. If greater control is necessary in localized areas, a system will be constructed.

Direction of air flow from areas of lower contamination to areas of higher contamination will be maintained at all times.

Exhaust from areas in which airborne contamination potential is present will be directed through prefilters and high efficiency particulate air (HEPA) filters.

Filter replacement will be performed when pressure differential across HEPA filters exceed 6 in. of water, or when indicated by reduced air flows. Prefilters will be replaced when pressure differentials across the filters exceed 1 in. of water.

Where practical, a minimum of six air changes per hour will be provided in areas posted as airborne radioactivity areas.

Ventilation systems will provide once-through air with no provision for recirculation.

H. EVALUATION OF AIRBORNE CONTAMINATION

Airborne contamination will be evaluated to assure that no individual is exposed to airborne radioactive or toxic material in excess of regulatory limits.

1. Air Monitoring

Air monitoring for airborne radioactive material will be performed by means of continuous air monitors in such areas as deemed necessary by the RNS Representative.

2. Air Sampling

Air sampling for airborne radioactive or toxic material could be performed by the following methods:

a) Continuously or intermittently by Gast Vacuum Pump air sampling units located at various points throughout a facility. Data from these samples will be evaluated and recorded weekly or daily as indicated by the potential for airborne activity.

b) Special, "hi-volume grab samples" at the discretion of the Radiation and Nuclear Safety representative.

d) Toxic gas detectors, such as the "length of stain" type will be used as indicated by the potential for such exposure.

I. LIMITS FOR AIRBORNE RADIOACTIVITY LEVELS

Every reasonable effort will be made by the use of engineering safeguards to maintain airborne contamination levels at less than 10% of the applicable limits described in DOEM 0524 and 10 CFR 20. In the event airborne contamination levels approach or exceed the applicable limits, the appropriate respiratory protective devices will be utilized to control the exposure.

The applicable limits for airborne contamination levels in the radiologically posted areas are the limits described in Column 1, Table I, Appendix B, 10 CFR 20. These limits will apply to occupational exposure for 40 hours in any seven consecutive days which translates to a time-integrated exposure for seven consecutive days. In the event any employee receives a time-integrated exposure to airborne radioactive materials in excess of 25% of the allowable exposure in seven consecutive days, as indicated by lapel air sampling, appropriate respiratory protection will be required to prevent exposures in excess of the limit. Specific protection factors will be applied to specific types of respirators. Protection factors are applied to airborne concentrations to determine the concentration inhaled by the wearer, according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Airborne Concentration}}{\text{Protection Factor}}$$

Applicable protection factors for air purifying respirators will be 10 (0.1% toxic gas or vapor concentration) for half-face masks and 50 (0.5% toxic gas or vapor concentration) for full-face masks. Only limited use of atmosphere supplying respirators is anticipated. If required, they will be used only by persons specifically qualified and trained in the use of such devices.

At the discretion of the RNS Representative, certain specific operations may require the use of respiratory protective devices strictly on the basis of the potential for exposure to airborne contaminants. Such operations will be identified as work progresses.

In the event that air sampling indicates airborne radioactive material in concentrations greater than the occupational limits, all persons entering the facility will be required to wear lapel air samplers and, if necessary, appropriate respiratory protection devices if the use of such devices is authorized by the RNS Representative.

J. PERSONNEL MONITORING DEVICES

1. Film Badges

Film badges will be worn by all persons entering radiologically posted areas. Film badges will normally be exchanged at the end of each calendar quarter, or in the case of persons with greater exposure potential, at the end of each month. Special film badges and direct reading dosimeters may be required in addition to the regular personal badge for radiation exposure control during work in High Radiation Areas. The special badges will be processed as required to evaluate cumulative radiation exposure. An exposure report sheet will be provided to supervision listing the reported radiation exposure for each person assigned to the program. Radiation exposure to personnel will be maintained to as-low-as-practicable levels. During any calendar quarter the occupational dose to the whole body of radiation workers shall not exceed 3 rems, as modified by the lifetime occupational exposure limit of 5 (N-18) rems, where "N" equals the individual's age in years at his last birthday. Whenever practicable, dismantling tasks will be planned to utilize remote tooling or shadow shielding to reduce the personnel exposure associated with the performance of the task. Personal film badges will be distributed to job-site personnel by the RNS representative. Visitors film badges will also be located at the job site for issuance by the RNS representative. A signout sheet will be provided for use in the issuance of the visitor badges. All visitors entering a radiologically posted area will complete the signout sheet and obtain a visitor's badge prior to entry. The badge will be returned following the visit, with the exception that visitors anticipating multiple entries may keep the badge for the balance of the calendar quarter.

All film badges used for the DORF program will contain beta-gamma sensitive film packets with the appropriate shields for radiation quality assessment.

2. Dosimeters

Dosimeters may be issued in conjunction with film badges during certain operations at the discretion of the RNS representative to provide an additional control on planned radiation exposures.

3. Extremity Monitoring

Whenever operations are performed which pose a potential for significant extremity exposure, extremity monitoring will be performed. Finger ring film badges or thermoluminescent dosimeters will be utilized for extremity monitoring.

K. AREA RADIATION MONITORING SYSTEMS

1. Area Film Badges

Area film badges will be mounted at selected locations throughout those facilities under the jurisdiction of the D&D program. These film badges will provide a record of integrated radiation levels for the exposure period at these locations. Area badges will be exchanged once each quarter and records of the badge exposures will be maintained by Radiation and Nuclear Safety.

L. BIOASSAY

1. Requirement

Bioassay, principally by means of urinalysis, will be utilized as a means of assessing internal radiation exposure of personnel. A baseline specimen will be obtained from each worker assigned to work in the radiologically posted areas. During the initial period of actual facility decommissioning, specimens may be collected at frequencies of 1 week to 1 month (depending on the nature of the work). Following the initial period, the collection frequency may be reduced, assuming engineering safeguards against airborne radioactivity are demonstrated to be effective. Specimens will then be submitted at least once each calendar quarter, with the exception that specimens will be submitted once each 6 months by persons not routinely assigned the radiologically posted areas.

Special bioassay specimens, including urine and fecal specimens, will be submitted at the discretion of the RNS representative or Radiation and Nuclear Safety Management whenever there is reason to believe that personnel may have been subjected to internal exposure.

Whenever the analysis of a routine or special bioassay specimen indicates radioactivity present in excess of the minimum detection limit of the analysis, resampling will be performed at a frequency no greater than biweekly.

Invivo lung counting or whole body counting may be used to provide direct evaluation of internal deposition of radioactivity for purposes of confirming urinalysis data, or of providing further evaluation of suspected exposures.

Radiation and Nuclear Safety will notify the Site Manager of the names of employees for whom bioassay specimens are due. The Site Manager will assure that those employees pick up a specimen bottle on the date indicated and collect and return the specimen as directed on the bottle.

2. Analysis

Bioassay specimens will be accumulated by Radiation and Nuclear Safety and shipped to a vendor laboratory for appropriate analysis. Radiation and Nuclear Safety will notify the Site Manager and Rockwell Program Management of any significantly positive results of bioassay analysis. In the event urinalysis indicates excretion rates which are indicative of the presence in an employee of greater than 50% of a maximum permissible body burden, that employee will be restricted from further work in radiologically posted areas until such time as two consecutive urinalyses submitted at least 5 days apart each indicate less than 25% of a maximum permissible body burden.

3. Incidents and Injuries

Any injury, no matter how small, received while working in a radiologically posted area must be reported immediately to the Site Manager or the RNS representative. Medical services will be obtained as required. The RNS representative will conduct wound monitoring, as necessary.

Employees with open cuts, abrasions, etc., will be restricted from work in radiologically posted areas unless specific approval is given by Radiation and Nuclear Safety. All incidents suspected, or known to have caused internal deposition of radioactivity must be reported immediately to the RNS representative.

M. PROTECTIVE CLOTHING AND EQUIPMENT

All persons entering a radiologically posted area in which unencapsulated radioactive material is processed will be required to don protective clothing at the change line located outside the entrance to these areas. The items of protective clothing required for entrance into these areas include, as appropriate:

1) Red-trimmed laboratory coat or coverall
2) Plastic or canvas show covers
3) Respirators.

Protective clothing and equipment for protection against potential hazards other than ionizing radiation will be prescribed on a case by case basis.

Respirators will only be fitted and issued by the RNS representative. No employee will be allowed to work in areas in which respirators are required unless he has been fitted and has completed the Rockwell/ESG Respiratory Protection training course within the past 12 months, including appropriate medical evaluation.

The RNS representative will establish respirator exchange frequencies as indicated by individual requirements. In addition to the protective clothing required for entry, certain additional items of clothing, such as skull caps or red-trimmed coveralls, may be required for certain operations posing high potential for contamination. Surgeons gloves will be required for operations involving direct handling of contaminated equipment. Persons exiting radiologically posted areas will remove their protective clothing at the change line and place the items of clothing in the drums, racks, or hangers provided as appropriate.

Respirators will be returned to the plastic bag in which they were issued pending re-use or return to the respirator maintenance laboratory. Immediately upon exiting these areas, all persons will monitor their hands and feet with the count rate meter provided there. They will then proceed to the nearest washroom and wash their hands.

N. HANDLING OF CONTAMINATED PROTECTIVE CLOTHING

All reusable items of protective clothing will be removed from the facility for decontamination and reissue. Disposable items will be collected and disposed of as radioactive waste. Laundry drums, lined with 50-gallon plastic bags, will be provided at the change line for the accumulation of contaminated laboratory coats, canvas shoe covers, and coveralls. The contaminated laundry will be collected as the bags are filled and will be processed through a licensed vendor.

Waste drums, lined with plastic bags, will be provided at the change line for the accumulation of disposable items such as caps, plastic shoe covers, and surgeons gloves. This waste will be pacakged as the bags are filled and will be processed for ultimate disposal.

O. INSTRUCTION OF PERSONNEL

Prior to beginning work in the radiologically posted areas, all employees will be indoctrinated with regard to radiation and industrial safety rules.

Employees whose regular assignments include for the first time work in radiologically posted areas, must complete a training course covering the general aspects of working with radioactive materials. This course will include (a) a description of the properties and potential hazards of radiation and radioactive material; (b) the basic principles of radiation protection; (c) the requirements of applicable Standard Operating

Policies and applicable regulations; (d) safe handling practices; and (e) emergency procedures.

P. EMPLOYEE QUALIFICATIONS

The Site Manager will furnish to Radiation and Nuclear Safety the names of all persons who will be assigned to work in the radiologically posted areas. Subsequently, whenever additional employees are to be assigned to work in these areas, Radiation and Nuclear Safety will be notified prior to each assignment. Radiation and Nuclear Safety will review the qualifications of persons assigned to work in the radiologically posted area and establish that these persons are fully qualified "radiation workers" and that they have sufficient familiarity with the operations in the posted areas to allow them to work safely in these areas. Included in the required qualifications or preparations for assignment to work in these areas are:

1) Personal film badge assignment
2) Bioassay baseline sample
3) Inclusion on periodic bioassay roster
4) Medical baseline examination
5) Inclusion on periodic medical examination roster
6) Completion of radiation worker training course
7) Completion of respirator training course
8) Successfully fitted with an approved respiratory protective device
9) Completion of facility indoctrination
10) Completion of required special training
11) No precluding physical limitations or radiological restrictions
12) NRC Form 4 or equivalent on file with Radiation and Nuclear Safety.

Rockwell Program Management or the Site Manager will also notify Radiation and Nuclear Safety of those persons whose assignments in posted areas are being terminated.

Q. INSTRUMENTATION

Radiation and Nuclear Safety will establish the requirements for radiological instrumentation, provide the instruments from general inventory if available, request calibration and repairs as required and instruct operations personnel in the use of these instruments as required.

Personnel monitors will be provided at change lines and in change rooms. Each of these monitors will consist of an alpha or beta-sensitive (as appropriate) detector, a count rate meter, and an audible "poppy-type" signal. These monitors will be inspected by Instrument Repair at least once each 3 months.

Continuous air monitors will be provided as required. These monitors will sample air through a filter media at a rate of about 1 cfm and will continuously monitor the particulate radioactive material collected on the filter media. The monitors will provide a ratemeter display of activity levels and an audible alarm which will actuate automatically in the event the radioactive material collected on the filter exceeds a preset level. These monitors will be serviced and calibrated at least once each 3 months.

Beta-gamma and alpha sensitive counting systems will be provided for use by the RNS representative in evaluating air samples and surface contamination samples for radioactivity. These systems will be serviced and calibrated at least once each 6 months.

Various types of beta-gamma and alpha sensitive portable radiation survey instruments will be provided for use by the Radiation and Nuclear

Safety representative in the day-to-day surveillance of operations in radiologically posted areas. All portable radiation survey instruments will be serviced and calibrated at least once each 3 months, or at shorter time intervals if recommended by the manufacturer.

R. DECONTAMINATION REQUIREMENTS

The requirements for decontamination in day-to-day operations will be determined by the RNS representative and communicated to the Site Manager who will assure that the required decontamination is performed.

1. Personnel Decontamination

In the event radioactive contamination is detected or suspected to be present on the skin or hair of an employee, the RNS representative will evaluate the degree of contamination and direct the decontamination efforts. In the event the contaminated employee is injured, the Site Manager will arrange for medical services. The RNS representative will direct or perform decontamination of the employee to acceptable limits using prescribed methods, unless it becomes apparent that further decontamination efforts will cause significant skin damage. In this case, the RNS representative will ask that further decontamination be accomplished under the direct supervision of a licensed practicing physician.

2. Equipment Decontamination

In the event that equipment, components, materials, etc., are found to be contaminated in excess of the appropriate limits, the RNS representative will promptly notify the Site Manager who will effect the required decontamination by operations personnel.

3. Area Decontamination

In the event the floors or walls of an area are found to be contaminated in excess of the appropriate limits, the RNS representative will notify the Site Manager and request decontamination. The RNS representative will coordinate decontamination efforts with operations personnel as necessary.

S. REMOVAL OF EQUIPMENT FROM RADIOLOGICALLY POSTED AREAS

All equipment or materials moving into unposted areas from any radiologically posted area must be surveyed for radiation and radioactive contamination levels. No item may be moved into any unposted areas if it is contaminated in excess of the limits established for such areas as shown in Table A-1. The radiation and contamination levels will be assessed by the RNS representative immediately prior to the transfer of the item. Required decontamination will be performed by the operations personnel.

In case of packaged items, the outer surfaces of the package will be surveyed for radiation and contamination levels, and these surfaces must be free of contamination in excess of the limits for unposted areas as shown in Table A-1. Packaged contaminated items will comply with the provisions of Title 49 Code of Federal Regulations and will be tagged with a completed radioactive materials tag prior to transfer into radiologically unposted areas.

T. RESTRICTED ACCESS AREA ENTRY PERMIT (FORM 719-L)

Varying degrees of control, consistent with the hazard involved, are exercised over posted areas by Radiation and Nuclear Safety. The Form 719-L is a means of restricting access to posted areas on the basis of personnel and potential hazards. The highest degree of hazard is associated with an area in which the active contamination and radiation levels are of such significance that special rigid entry controls and safety precautions are necessary.

1. Subcontractors

Subcontractor personnel who have cause to work within any radiologically posted area must have a completed Form 719-L which will be submitted by the Site Manager and approved by the RNS representative prior to the start of work. Groups representing the same contractor and who work in the same general area need only one tagged area entry permit.

All contractor personnel entering any radiologically posted area must obtain a film badge prior to entry. Radiation and Nuclear Safety shall determine if previous entries into posted areas had been made during the current calendar year and if so, shall ascertain from personnel monitoring records the dose received, and plan radiation exposures accordingly.

Contractor personnel performing work in a radiological posted area will be surveyed prior to breaks, lunch and quitting time by means of portable battery operated or ac survey instruments. If the instrument survey indicates contamination, decontamination will be effected immediately.

Radiation and Nuclear Safety will attend any operation involving contractor personnel in a posted area to the extent necessary to ensure that such personnel perform their duties in such a manner as not to cause the release of radioactive material or become unduly exposed to radiation. Should such an event occur, work will be stopped until appropriate surveys have been performed and necessary corrections effected.

All tools and equipment used by contractor personnel in radiologically posted areas must be surveyed and found to be free of contamination before they may be removed. The removable contamination limit for tools and equipment shall be 20 dpm/100 cm^2 alpha activity and 100 dpm/100 cm^2 beta. Fixed contamination shall be undetectable with appropriate portable survey instruments.

Contractor personnel shall not be exposed to concentrations of radioactive material in air and water greater than 10% of the maximum permissible concentrations as listed in 10 CFR 20, "Standards for Protection Against Radiation," under Table I of Appendix B unless they are qualified as radiation workers as described below.

Whole body dose to contractor personnel will be limited to 500 mrem/year. An exception to the standard 500 mrem/year will be made if an affidavit, signed by a representative of the contractor, authorizes their employees to be considered radiation workers, in which case the employee will be required to execute an NRC Form 4 or equivalent authorizing Rockwell/ESG to obtain occupational radiation exposure histories.

Cumulative records or radiation exposures will be maintained by Radiation and Nuclear Safety to ensure that personnel are not exposed in excess of applicable standards.

If a contractor employee receives a radiation exposure in excess of 25 mrem, Rockwell/ESG will notify the contractor of the dose within 30 days following the determination of such exposure.

Contractor personnel under 18 years of age will be limited to 125 mrem/calendar quarter.

Approved visitors will be considered in the same category as outside contractors and must have completed a Form 719-L prior to performing work in radiologically posted areas, or visiting such facilities for extended periods during which they are not under continuous escort.

2. Rockwell Employees - Other Than Personnel Assigned to the DORF D&D Program

Personnel who have cause to enter the radiologically posted areas for maintenance or repair purposes will complete a Form 719-L prior to entry.

3. Restricted Access Areas

All personnel who have reason to enter certain rigidly controlled areas, "Restricted Access Areas," such as radioactive exhaust system filter plena or liquid waste holdup tanks, will complete a Form 719-L prior to entry into these areas. A minimum of two persons will be assigned to perform operations in these areas, or as otherwise specified by HSRS.

4. Preparation of Form 719-L

When a Form 719-L is required, the individual or manager of the group requesting entry will fill out his portion of the form and give it to the RNS representative who will outline the pertinent radiological safety instructions, sign the form, and return it to the originator.

The originator will obtain the signature of the Site Manager and distribute copies are required.

The entire working crew will initial the Form 719-L, signifying receipt and comprehension of instructions on the form.

Copies of completed Forms 719-L will be kept on file indefinitely by Radiation and Nuclear Safety.

U. EMERGENCY CONDITIONS

1. Ventilation Loss or Airborne Radioactivity Alarms

In the event of a radioactive exhaust system failure, or of other evidence of loss of airflow in ventilated areas, personnel will leave these areas and await evaluation of the facility conditions by Radiation and Nuclear Safety.

In the event of actuation of the "variable warble" and/or bell alarm of continuous air monitors, personnel present will evacuate the facility and await evaluation by Radiation and Nuclear Safety.

V. RADIOACTIVE WASTE MANAGEMENT

All radioactive waste will be collected, evaluated, processed, and shipped for disposal to a licensed radioactive waste disposal site.

1. Solid Waste

Low level solid radioactive waste will be packaged into standard containers such as steel DOE Specification 17-H 55-gallon drums or wood type box DOT Specification 19-A or low specific activity, strong, tight containers. In the case of low level waste such as concrete rubble which is generated in large volumes, special containers may be designed. Low-level solid radioactive waste generated in support of decontamination operations (i.e., plastic shoe covers, surgeons gloves, kim-wipes, miscellaneous plastic, etc.) will be collected in drums lined with plastic bags. When the bags are filled, they will be final packaged for disposal.

High-level radioactive waste will be packaged and shipped in containers such as reusable lead-shielded casks, one-way concrete shielded containers, or DOT approved overpacks.

2. Liquid Waste

Liquid radioactive waste will be solidified. In the event there is other liquid radioactive waste such as acids or corrosives, they will be neutralized in drums and then solidified for disposal by land burial. Solidification will consist of cementation.

W. INDUSTRIAL SAFETY REQUIREMENTS

Industrial safety requirements are described by Rockwell/ESG Health and Safety Procedures, SOP's ANSI, AEC Manaul, and OSHA.

1. Hoisting and Rigging

Hoisting and rigging operations shall be conducted in compliance with the requirements of 29 CFR 17, Part 1910, Subpart N, and the ANSI B30 Series.

Equipment used in material handling shall be proof-loaded and maintained per PL Series 8.

All personnel engaged in hoisting and rigging shall be qualified by appropriate experience and training.

2. Explosives

Explosives use and handling shall be in accordance with Federal, State, and local regulations including 29 CFR 17, Part 1910.109, Health and Safety Procedure G-16, and Section XXV, EM 385-1-1, "General Safety Requirements Manual," of the Corp of Engineers with the exception of Paragraph 25.8.04.

3. Insulation Removal

The removal of insulation material containing asbestos will be done in compliance with 29 CFR 17, Part 1910.93a.

Warning signs, OSHA approved, will be displayed at each location where the airborne concentration of asbestos fibers may exceed the allowable exposure limits. The airborne concentration will be verified by environmental sampling.

Insulation will be removed in a manner that will minimize the generation of airborne dust. Wet methods will be used if practical. Approved respirators will be worn during all asbestos insulation removal operations. If the presence of asbestos in a material is questionable, it will be assumed to be present.

4. Burning, Cutting, and Welding

Burning, cutting, and welding will be performed only as authorized by the Site Manager. Additional requirements may be imposed by the RNS representative as necessary to protect against toxic and/or radioactive vapors and fumes.

5. Noise

Personnel exposed to noise in excess of the limits specified in 29 CFR 17, Part 1910.95 will be required to wear approved ear protection. The noise level for an 8-hour exposure is 90 dbA. Personnel exposures to noise over 90 dbA shall be evaluated by the RNS representative. Personnel exposures to noise in excess of 115 dbA are prohibited without approved ear protection.

6. Confined Space Entry

Confined space entires will be made in compliance with Health and Safety Procedure G-19. All Class 2 entires will require the preparation of a Form 719-L, Rev. 5-70, "Restricted Access Area Entry Permit," which will designate the required control measures.

7. Contractor Safety

Contractors will conduct operations in compliance with Federal, State, and local codes, standards and regulations as applicable. Contractors are subject to compliance with Rockwell Internationla and ESG regulations as indicated in Form 511-C which describes Contractor Safety Requirements at Rockwell International facilities.

MAZZONE/av

STANDARD FORM 26, JULY 1966
GENERAL SERVICES ADMINISTRATION
FED. PROC. REG. (41CFR) 1-16.101

AWARD/CONTRACT

1. CONTRACT (*Proc. Inst. Ident.*) NO. DAAK21-79-C-0136

2. EFFECTIVE DATE 79 SEPT 14

3. REQUISITION/PURCHASE REQUEST/PROJECT NO. DAAK21-79-B-9012

4. CERTIFIED FOR NATIONAL DEFENSE UNDER BSDA REG. 2 AND/OR DMS REG. 1. RATING: S1

5. ISSUED BY CODE W71BFJ

Department of the Army
Harry Diamond Laboratories
2800 Powder Mill Road
Adelphi, MD 20783

6. ADMINISTERED BY (*If other than block 5*) CODE

7. DELIVERY
[X] FOB DESTINATION
[ ] OTHER (*See below*)

8. CONTRACTOR NAME AND ADDRESS CODE FACILITY CODE

(*Street, city, county, State, and ZIP code*)

Rockwell International Corp.
Energy Systems Group
8900 DeSoto Avenue
Canoga Park, CA 91304

9. DISCOUNT FOR PROMPT PAYMENT

None

10. SUBMIT INVOICES (*4 copies unless otherwise specified*) TO ADDRESS SHOWN IN BLOCK 12

11. SHIP TO/MARK FOR CODE

See Section H, Paragraph H.4

12. PAYMENT WILL BE MADE BY CODE W71BFJ

Finance and Accounting Officer
Harry Diamond Laboratories
2800 Powder Mill Road
Adelphi, MD 20783 *(See Block 16)

13. THIS PROCUREMENT WAS [X] ADVERTISED, [ ] NEGOTIATED, PURSUANT TO: [ ] 10 U.S.C. 2304 (a)( ) [ ] 41 U.S.C. 252 (c)( )

14. ACCOUNTING AND APPROPRIATION DATA 2182040 6A-7063 P612120-2572 S18129 Proj: T0280C
612120H250050 A18R005201A1A9
6.2 R&D FY78 1L162120AH29 Amount Obligated: $335,800.00

15. ITEM NO.	16. SUPPLIES/SERVICES	17. QUANTITY	18. UNIT	19. UNIT PRICE	20. AMOUNT
	* ADDRESS TO WHERE PAYMENT SHOULD BE MAILED: Rockwell International Corporation Atomics International Division Post Office Box 360876M Pittsburgh, PA 15251			Firm Fixed Price	

21. TOTAL AMOUNT OF CONTRACT $ 335,800.00

CONTRACTING OFFICER WILL COMPLETE BLOCK 22 OR 26 AS APPLICABLE

22. [ ] CONTRACTOR'S NEGOTIATED AGREEMENT (*Contractor is required to sign this document and return ____ copies to issuing office.*) Contractor agrees to furnish and deliver all items or perform all the services set forth or otherwise identified above and on any continuation sheets for the consideration stated herein. The rights and obligations of the parties to this contract shall be subject to and governed by the following documents: (a) this award/contract, (b) the solicitation, if any, and (c) such provisions, representations, certifications, and specifications, as are attached or incorporated by reference herein. (*Attachments are listed herein.*)

26. [X] AWARD (*Contractor is not required to sign this document.*) Your offer on Solicitation Number DAAK21-79-B-9012, including the additions or changes made by you which additions or changes are set forth in full above, is hereby accepted as to the items listed above and on any continuation sheets. This award consummates the contract which consists of the following documents: (a) the Government's solicitation and your offer, and (b) this award/contract. No further contractual document is necessary.

23. NAME OF CONTRACTOR

BY (Signature of person authorized to sign)

27. UNITED STATES OF AMERICA

BY (Signature of Contracting Officer)

24. NAME AND TITLE OF SIGNER (*Type or print*)

25. DATE SIGNED

28. NAME OF CONTRACTING OFFICER (*Type or print*) JOHN B. PETRO

29. DATE SIGNED 79 SEPT 14

PART II	SECTION E	REF. NO. OF DOC. BEING CONT'D. DAAK21-79-C-0136	PAGE E.1	OF I

NAME OF OFFEROR OR CONTRACTOR

Rockwell International Corp., Energy Systems Group

ITEM NO.	SUPPLIES/SERVICES	QUANTITY	UNIT	UNIT PRICE	AMOUNT
0001	Contractor shall dismantle and radioactively decontaminate the Diamond Ordnance Radiation Facility; remove and ship activated material to disposal site; in accordance with Section F.	1	LO	$335,800.00	$335,800
0001AA	Data, in accordance with DD Form 1423, Attachment No. 5 to Section M.	1	LO	* NSP	* NSP
	* NSP = Not Separately Priced.				

☆ U.S. GOVERNMENT PRINTING OFFICE: 1969 - 329-953

"F.1. The contractor shall be responsible for the dismantlement and radioactive decontamination of the reactor facility. (This does not include the removal and shipment of the reactor fuel elements from the facility.)

F.2. The contract will be performed in three (3) phases:

Phase I Preparation of a Dismantlement Plan.
Phase II Dismantlement and radioactive decontamination of the facility.
Phase III Completion of all remaining tasks after Radioactive Survey by U.S. Army Environmental Health Agency.

F.3. Phase I
F.3.1 The contractor shall prepare a dismantlement plan which shall describe the methods and implementing procedures necessary to accomplish the dismantlement and radioactive decontamination of the facility. The plan shall describe the general methods of approach in accomplishing the tasks listed in Phase II of these specifications. (A specific plan format is not required.)

"The general techniques of concrete removal shall be listed. In the event that the use of controlled explosives for the reactor dismantlement is contemplated by the contractor, the use of the explosives shall comply with Federal, State and local regulations. Also, blasting operations shall comply with Section XXV, EM 385-1-1, "General Safety Requirements Manual" of the Corps of Engineers with the exception of paragraph 25.B.04. All explosives and explosive waste shall be removed from the work site by the contractor".

F.4.2 The contractor shall be responsible for the spectral analyses and dose rate measurements necessary to assure that acceptable surface contamination and residual activity levels are below the limits stated in NRC Guide 1.86. The methodology in decontamination for release for unrestricted use as stated in this Guide shall be followed.

F.4.3 The contractor shall provide its own health physics support including radiation survey instruments and necessary gamma ray spectral analyses. Documentation of current calibration data on these instruments shall be available to the Contracting Officer. All health physics procedures shall be reviewed by the Contracting Officer's Representative for compliance, where applicable, with current copy of Title 10 Chapter 1 Code of Federal Regulations, Part 20 (10CFR20).

F.4.4 The contractor shall be responsible for contamination control, including:

a. Dust collection and absolute filtering procedures.
b. Respirator and protective clothing requirements.
c. Air sampling procedures.
d. Records of significant radiation surveys and analyses.

Contractor shall make these records available to the Contracting Officer's Representative.

F.4.5 The contractor shall be responsible for any subcontracted work such as shipment and burial of radioactive materials, construction/demolition operations, and the design, test and certification of shipping container, if required.

F.4.6 The contractor shall remove and dispose of three (3) 5000 gallon water hold-up tanks. Disconnect water dilution, air mixing systems, and remove the inlet and exit valving system.

F.4.7 The Contracting Officer has the authority to stop any operation that indicates a radiological hazard to WRAMC personnel, general public, and the environment.

F.4.8 The contractor shall furnish evidence that each individual who will be involved in the dismantling operations has had a medical examination by a practicing licensed physician and a complete blood count.

F.4.9 The Contracting Officer's Representative shall attend or be briefed, in detail, on any informal meetings or discussions with regulatory agency personnel by the contractor.

F.6 Hazardous Items List (Safety)

1. Neutron activated materials.
2. Radioactive materials and radioactive contamination.
3. Possible radioactive airborne materials resulting from dismantling work.
4. Possible radioactive liquids (water) resulting from decontamination procedures.

F.7 Drawing List - DORF Decommission

1.	Lead Curtain	T35618J300
2.	Rolling Door Exposure Room	M5
3.	Plans - Air Conditioning	M1
4.	Power	E4
5.	Equipment Room Details	M2
6.	Plumbing	M3
7.	Sections	S2
8.	Ramp Plans	S4
9.	Vertical Section of DORF Reactor	
10.	Sectional Elevation of DORF Reactor	
11.	Warm-liquid Waste Storage System	

F.8 Backup Data

1. Radio-Isotopic Analysis of Radioactive Material in the DORF Structure Before Decommissioning, Attachment No. 1.
2. AEC Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, dated June 1974, Attachment No. 2.

F.9 Salvable Equipment

Contractor shall have the right to salvage and remove for his own use the following items, having a combined fair market value of $500.00:

Plug Door Drive Mechanism	1 Each
Detention Tank Valves	6 Each

"G.1 The contractor shall pack all recoverable equipment, not to be discarded, in weather proof containers. The Contracting Officer or his duly authorized representative shall determine equipment to be salvaged and its disposition. Selective reactor components (some items radioactive) shall be packaged for shipment and shipped to DOE, Hanford Engineering Development Laboratory, Richländ, Washington. The items to be shipped to HEDL are listed as follows:

ITEM NO.	DESCRIPTION	UNIT	QUANTITY
1	Core Support Structure, Upper Section	EA	1
2	Core Support Structure, Lower Section	EA	1
3	Top and Bottom Grid Plates	EA	1
4	Connecting Rods for Control Rods	SET	1
5	Control Rods	SET	1
6	Carriage Drive Motor	EA	1
7	Water Pump: 1.5 HP	EA	1
8	Incor Experiment Tube	EA	1
9	Ion Chamber Supports and Ion Chambers	SET	3
10	Carriage Support Rails	SET	1
11	Lead Shield Door Drives and Linkage	SET	1
12	Pool Cover Plates	SET	1
13	Fuel Storage Racks, Underwater	EA	8
14	Fuel Measurement Tool with Dial Micrometer	EA	1
15	Aluminum Water System Piping	EA	1
16	Water Pumps	EA	3
17	Demineralizers, 3 Cu. Ft.	EA	4
18	Flowmeters, 25 GPM	EA	2
19	Neutron Source, 10 Curies, AM-BE	EA	1
20	Neutron Source Holder	EA	1
21	Pool Lights	SET	1
22	Carriage Positioning Potentiometer	EA	1
23	Carriage Umbilical Arm	EA	1
24	Fuel Element Location Diagram	EA	1
25	Water Box, 1 Cu. Ft. Capacity	EA	1
26	Charcoal Filter, 1 Cu. Ft. Capacity	EA	1"

G.2 The contractor will transport the jib crane from DORF to the Aurora facility, HDL, in accordance with Para F.4.13.

CONTINUATION SHEET

PIIN:	PART II -- SECTION I INSPECTION AND ACCEPTANCE	PAGE:
DAAK21 - 79 - C - 0136		I.1

I.1 Inspection and acceptance will be performed at the Harry Diamond Laboratories DORF upon the completion of each phase by a representative of the Government.

HSWP-QHP 4 OCT 1979

SUBJECT: Request for Modification of Harry Diamond Laboratories (HDL) Contract No. DAAK21-79-C-0136 for the Dismantlement and Decontamination of the Diamond Ordnance Radiation Facility (DORF), Forest Glen Section, WRAMC

HEALTH PHYSICS COMEBACK COPY:

Commander
Harry Diamond Laboratories
ATTN: DELHD-N-RB (W. L. Gieseler)
2800 Powder Mill Road
Adelphi, MD 20783

1. References:

a. Letter, DELHD-N-RB, 21 Sep 79, subject: Contract Award for the Dismantlement and Decontamination of the Diamond Ordnance Radiation Facility (DORF), Forest Glen Section, WRAMC.

b. Letter, DELHD-N-RBI, 27 Sep 79, subject: Correction to Letter from DELHD-N-RB to HSWP-QHP, dated 21 September 1979, Subject: Contract Award for the Dismantlement and Decontamination of the Diamond Ordnance Radiation Facility (DORF), Forest Glen, MD.

c. HDL Contract No. DAAK21-79-C-0136, 14 Sep 79.

2. Upon completion of the dismantlement and decontamination of the Diamond Ordnance Radiation Facility and its transfer to Walter Reed Army Medical Center (WRAMC), it is the plan of WRAMC to utilize the facility as a radioactive waste holding and processing facility.

3. Consequently, the following changes to the above referenced contract are requested:

a. Delete Part II, Section F, "Description/Specifications," Paragraph F.4.6.

b. Change Part II, Section F, "Description/Specifications," Paragraph F.5.2.b. to read as follows: "Remove and dispose of absolute filters."

FRANK SCHMIDT
Deputy Director
Facilities Engineering
WRAMC

DMA

4 Oct 79

1961 - DORF

HSWP-QHP

SUBJECT: Request for Modification of Harry Diamond Laboratories (HDL) Contract No. DAAK21-79-C-0136 for the Dismantlement and Decontamination of the Diamond Ordnance Radiation Facility (DORF), Forest Glen Section, WRAMC

c. Change Part II, Section F, Description/Specifications, Paragraph F.5.2.c. to read as follows:

> ~~"Inspect, test, and perform all work necessary to assure the water-tight-integrity of all sewer, and water liners leading to or exiting from the three (3) 5000 gallon water hold-up tanks. Provide written certification that said water and sewer liners meet or exceed the Plumbing and Gas Fitting Regulations of the Washington Suburban Sanitation District Code, Industrial and Special Waste Section, Paragraph 701 to Paragraph 702.15."~~

WSSC
441-4342
PLUMBING + HEATING ASSN
312 MARSHALL ARBUTHAN BLDG
MS DUDLEY
LAUREL MD
(OFF US 1)

d. Modify Part II, Section F, Description/Specifications, Paragraph F.9 to remove "Detention Tank Valves" from the list of equipment the contractor may salvage.

4. In addition to the above changes, it is requested that WRAMC be furnished with a copy of the approved vendor's dismantlement plan described in Part II, Section F, Paragraph F.3.1. and F.3.2. and upon completion of the contract a reproducible copy of the "as-built" facility plans as modified by this contract.

5. In order to assure that the Health Physics Office, Walter Reed Army Medical Center (WRAMC) may effectively fulfill the responsibilities delineated in Interservice Support Agreement No. W74MYG-78223-101 between WRAMC and HDL, it is requested that Mr. James E. Stafford, Chief, Radioactive Materials Control Branch, WRAMC Health Physics Office be appointed as one of the "Contracting Officer's Representatives (Technical)" specified in Part II, Section J, Paragraph J.2. of subject contract.

FOR THE COMMANDER:

FRANKLIN J. BORIUP
MAJ, MSC
Adjutant

DEPARTMENT OF THE ARMY
HARRY DIAMOND LABORATORIES
2800 POWDER MILL ROAD
ADELPHI, MD. 20783

DELHD-N-RB 21 September 1979

SUBJECT: Contract Award for the Dismantlement and Decontamination of the Diamond Ordnance Radiation Facility (DORF), Forest Glen Section, WRAMC

Commanding Officer
Walter Reed Army Medical Center
ATTN: HFWP-QHP
Washington, D.C. 20012

1. A fixed-price contract for the DORF dismantlement and decontamination was awarded on 14 Sep 1979 to Rockwell International, Canoga Park, CA.

2. The contract delivery or performance schedule is as follows:

2.1 Phase I (Preparation of a Dismantlement Plan) shall commence on the effective date of the contract with the delivery of the Plan to the Harry Diamond Laboratories (HDL) within thirty (30) days thereafter. 14 NOV 79

2.2 Phase II (Dismantlement and Radioactive Decontamination of the Facility) shall commence six (6) weeks following the completion of Phase I, and shall be completed three (3) months thereafter.

2.3 Phase III (Restoring the building for alternate use) shall commence thirty (30) days following the completion of Phase II, and shall be completed ~~two (2)~~ months thereafter.
THREE (3) as specified in section H.3 of contract

2.4 All deliveries to and from DORF and to and from the disposed area(s) shall be f.o.b. destination with all shipping and transportation costs to be borne by the contractor.

3. Contracting personnel and equipment will require access to and from the DORF area for the facility performance of the contract. The facility access gate will be closed and locked when contracting personnel leave the facility area after the end of each work-day.

4. Requested changes, if any, to the contract specifications by WRAMC shall be submitted, in writing, to the HDL contracting officers representative (W. L. Gieseler) by 5 October 1979.

DELHD-N-RBI

21 September 1979

SUBKECT: Contract Award for the Dismantlement and Decontamination of the Diamond Ordnance Radiation Facility (DORF) Forest Glen Section, WRAMC

FOR THE COMMANDER:

Walter L. Gieseler

WALTER L. GIESELER
Physicist
Diamond Ordnance Radiation Facility

CF:
DELHD-PR-CA, A. Mazzone
DELHD-FA, G. Chapman
DELHD-N-RB, J. Rosado
DELHD-N-RBI, P. Caldwell
DELHD-N-RBC, J. McGarrity
DELHD-SA, D. Williams
DELHD-SE, Y. Yeick

Harry Diamond Laboratories

Diamond Ordnance Radiation Facility

Fuel Handling Procedures for Transfer of Irradiated Fuel Elements to Shipping Casks

1. Introduction

The decommissioning of the Diamond Ordnance Radiation Facility (DORF) reactor will require the shipment of 92 TRIGA type fuel elements from the facility. These procedures are for the 89 irradiated (partically spent) fuel elements which are to be shipped in a licensed shipping cask to the University of Pennsylvannia, the University of Utah and to the Hanford Engineering Development Laboratory. An intra-agency agreement has been prepared by DOE, Savannah River Operations Office to provide services and responsibility for supplying approved irradiated nuclear material shipping casks and providing for their transport. A modified MH-1A Fuel Shipping Cask will be used with basket inserts to accommodate 48 TRIGA elements per shipment. Two shipments will be required. The basket design provides spaces for the long length requirements of the fuel follower control rods and the thermocouple instrumented elements.

a. Equipment

The fuel elements will be shipped in an approved shipping cask (S cask) and will arrive at DORF on a trailer. A transfer cask (T cask) will be used to transport the fuel from the reactor pool to the S cask.

The DORF 7 element transfer cask will be used. (The DORF transfer cask was designed for transferring fuel from the reactor pool to the fuel storage pits located on the main reactor floor.) The DORF fuel handling tool will be used to move the fuel between the pool storage racks to the T cask and then from the T cask to the S cask The building 3 1/2 ton capacity monorail hoist will be used to move the T cask from the pool to the S cask which will be on the trailer in the basement level directly below the main floor hatch. If there is insufficient vertical door clearance for the cask and trailer, the cask will be transferred to a low-boy trailer using a mobile crane. The procedure for handling the cask is contained in the Operations Manual for Spent Fuel Shipping Casks, Serial No. MH-1A, Bureau of Explosives Permit No. 2087. Minor changes to these procedures have been made for DORF loading application.

b. Radiation Monitoring

Radiation levels will be monitored and recorded at key stages of the operation. The DORF staff will be responsible for securing the facility area during fuel transfer. The entrance gate will be closed and monitored during fuel transfer operations and unauthorized persons will be excluded. All personnel involved with the fuel transfer will be issued film badges and pocket dosimeters. Conventional survey equipment will be used and a sufficient number of monitors will be used to provide a redundancy of measurements. The WRAMC Health Physicist will supervise personnel radiation safeguards.

c. Rehearsal

All phases of the transfer operation will be rehearsed before the actual transfer to provide personnel complete familiarity with equipment and procedures. A dummy fuel element is available for practice runs.

II. Shipment of Fuel Elements

a. General

The area involved in the fuel transfer operations is shown in Figures 1,2 & 3. Restricted access will be the area inside the facility fence which includes all buildings within the compound. The Physicist-in-Charge (PIC), or his designee, will have total authority and responsibility during the transfer of the fuel elements from the reactor pool to the shipping cask. Specific tasks and assignments will be given by the PIC. These assignments may be changed for subsequent transfers if necessary.

The vehicle transporting the empty S cask will be positioned inside the truck access door so that the cask is directly below the overhead hatch and mono-rail or outside the building on the truck ramp (Fig. 4).

b. Initial Preparation

The top of the empty S cask will be removed by the mono-rail hoist or a mobile crane. The inside of the cask will be checked to insure that the fuel storage liner is in place. The spacer, if present, which prevents fuel movement during shipment will be removed. The cask may be filled with water if radiation levels warrant (determined by the Health Physicist, Note: the dose-rate at 10 ft. from a single fuel element in air after a 3 month cool-down was 20 mr/hr). A trial placement of the empty T cask adjacent to the S cask and the transfer of a dummy fuel element from the T cask to the S cask fuel storage ring may be done at this time. After trial practice operations are satisfactory the T cask will be transferred to the reactor pool floor by the mono-rail hoist for the first loading. Using the standard TRIGA fuel element handling tool (Figure 5), the fuel elements will be transferred from the pool storage racks to the cask. The T cask will then be moved by the mono-rail hoist to a position above the open floor hatch and then lowered adjacent to the S cask on the trailer located in the basement area. The elements will be transferred from the T cask to the S cask using the fuel element handling tool from the main floor hatch area. The procedure will be repeated until assigned number of elements are transferred to the S cask for the scheduled shipment.

c. Closing the Cask

The crane will replace the top shield plug of the S cask and plug will be bolted in place as described in cask operational manual.

Radiation Monitoring

All personnel involved in the fuel transfer will wear film badges and pocket dosimeters as prescribed by the WRAMC Health Physicist. Portable survey meters will be used to measure the dose rates. The gamma area monitoring system (NMC scintillation detectors) will also be used. The NMC system has a chart recorder which prints out the dose rate readings. There are six of these detectors located in the reactor building.

The radiation levels at each significant state will be:

(1) T cask with fuel in air (1 ft and 10 ft)

(2) Level outside the S cask (3 ft)

The anticipated radiation dose rates to personnel involved in the fuel transfer have been determined from experiments performed with one of the "B" ring elements. The dose rates in air after a 16 month decay period were as follows:

(1) "B" ring element 10 ft from side = 6 mr/hr

(2) "B" ring element 1 ft from side = 135 mr/hr

(3) "B" ring element 10 ft above (along axis) = 3 mr/hr

The dose rate with 7 elements in the T cask were measured to be:

(1) T cask, 10 ft from side = 0.7 mr/hr

(2) T cask, 1 ft from side = 7.0 mr/hr

(3) T cask, 1 ft above = 28.0 mr/hr
= 3.0 mr/hr with cover plate

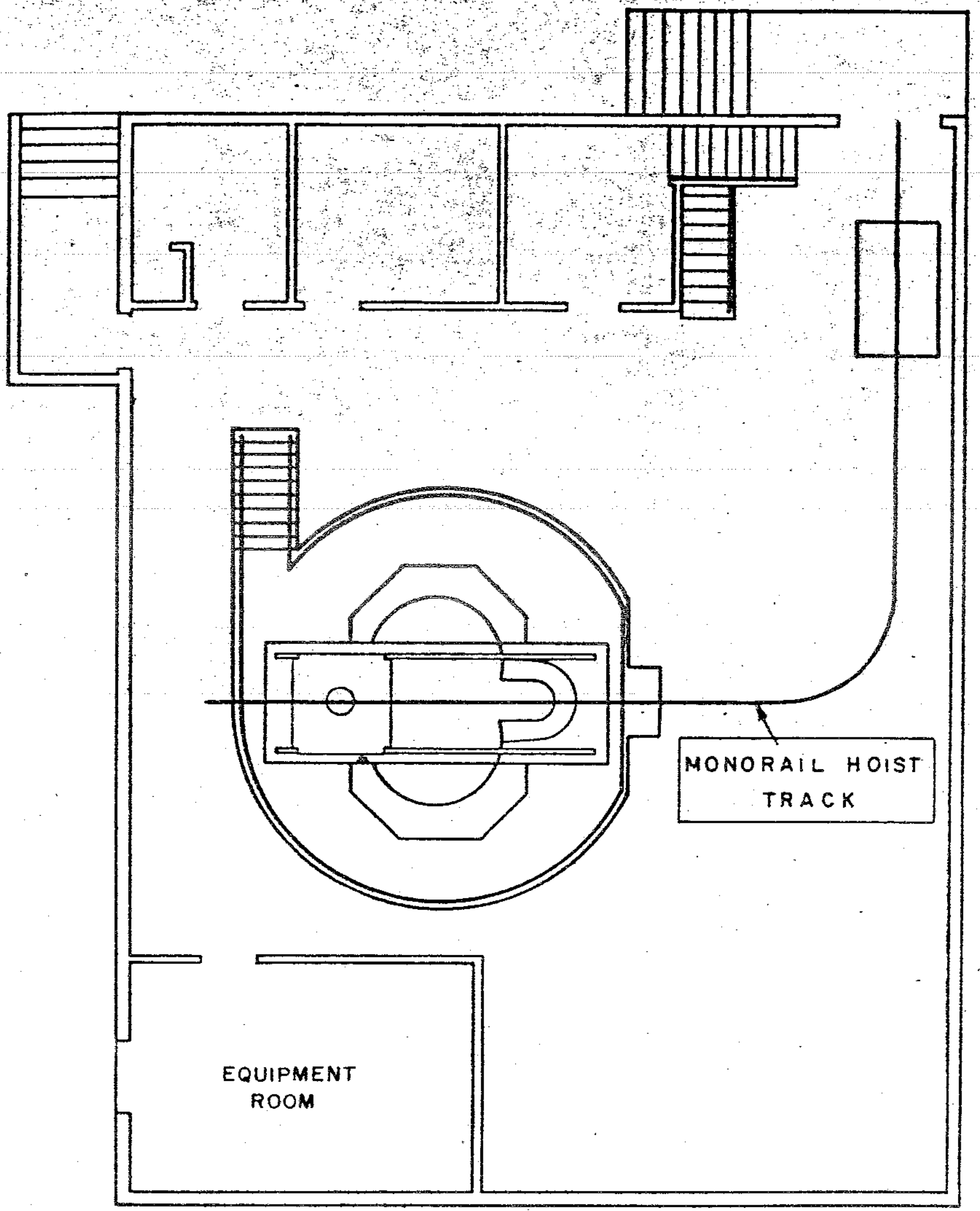
(4) T cask, 10 ft above cask: = 3.0 mr/hr

These distances are the minimum that personnel will be required to be at anytime. The elements are to be transferred from a position about 10 ft above the casks.

As the S cask is filled, the significant dose rate measurement is at the fuel handler position. The distance the fuel handler will be is about 10 feet above the cask, therefore, as the S cask is filled with fuel elements the dose rate at this position will increase. When the S cask is loaded with 46 fuel elements, the dose rate after the last fuel element transfer will be 6.6 times the dose rate above the T cask (7 elements) dose rate measurement. If we assume that 7 elements are transferred from the T cask to the S cask in each step, the dose rate 10 ft above the S cask would be as follows:

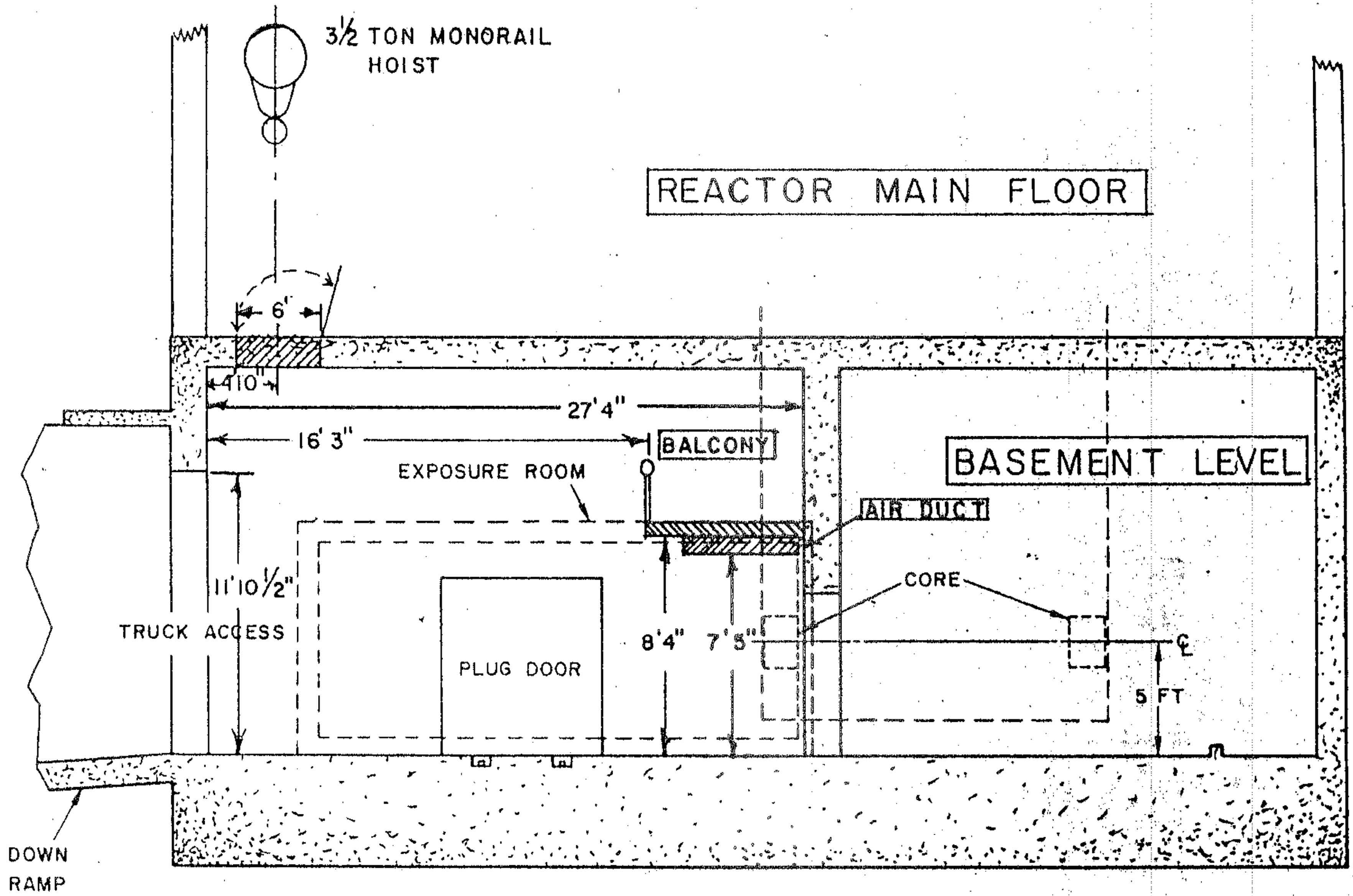
Step No.	No. of Elements	Dose Rate (mr/hr) 10 ft above S cask	Exposure (mr)
1	7	3	0.5
2	14	6	1.0
3	21	9	1.5
4	28	12	2.0
5	35	15	2.5
6	42	18	3.0
7	46	20	3.3
		total	13.8

From previous experience moving fuel with the DORF fuel element handling tool, it is estimated that it will take less than 10 minutes to transfer 7 elements in each step. Therefore, if one person performed all of the transfers from the T cask to the S cask the accumulated exposure would be only about 14 mr. We plan, however, to divide the fuel handling tasks among several persons to reduce the radiation exposures.



PLAN VIEW MAIN FLOOR

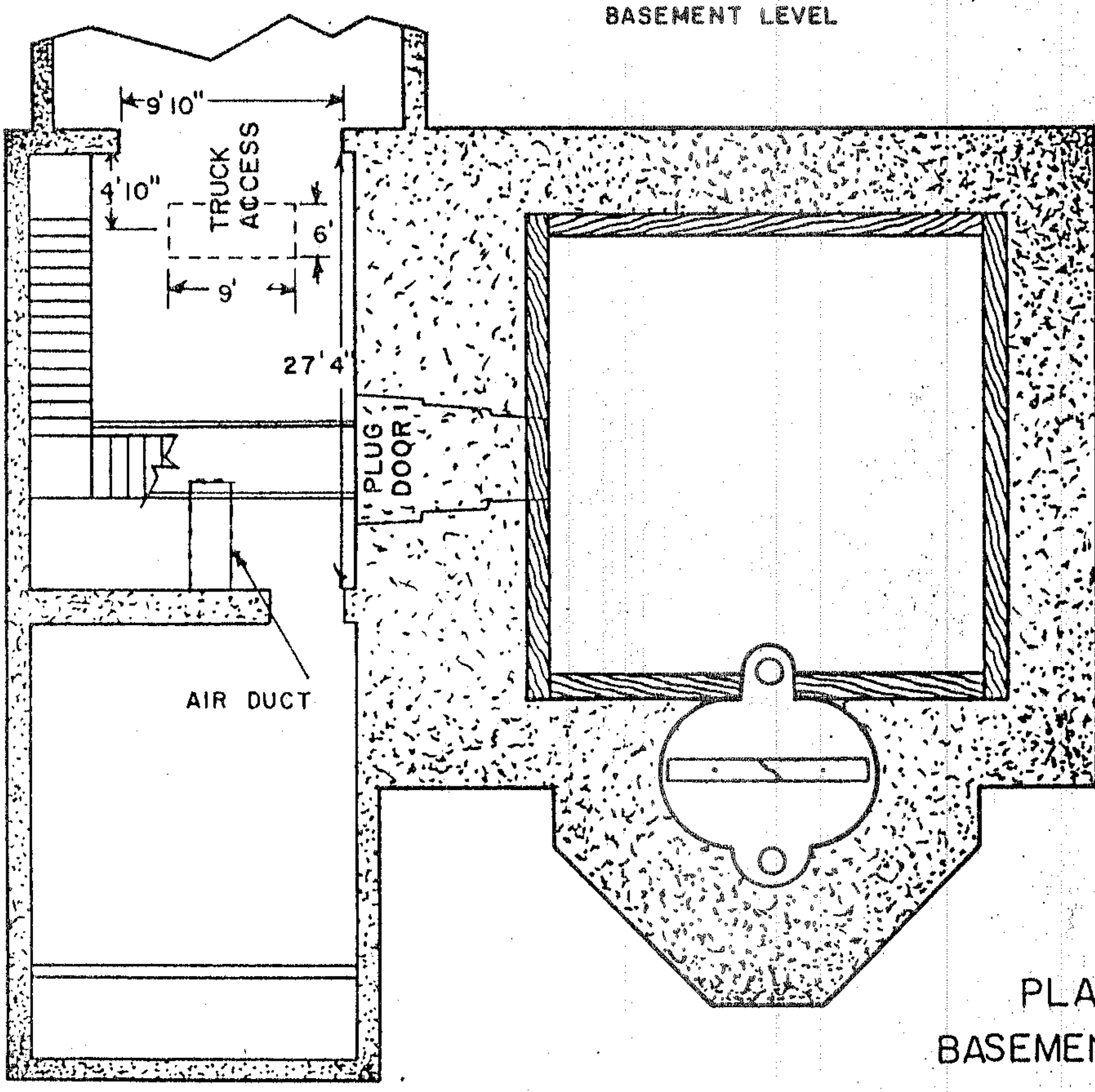
Fig. 1



VERTICAL SECTION

Fig. 2

BASEMENT LEVEL



PLAN VIEW
BASEMENT LEVEL

Fig. 3 .

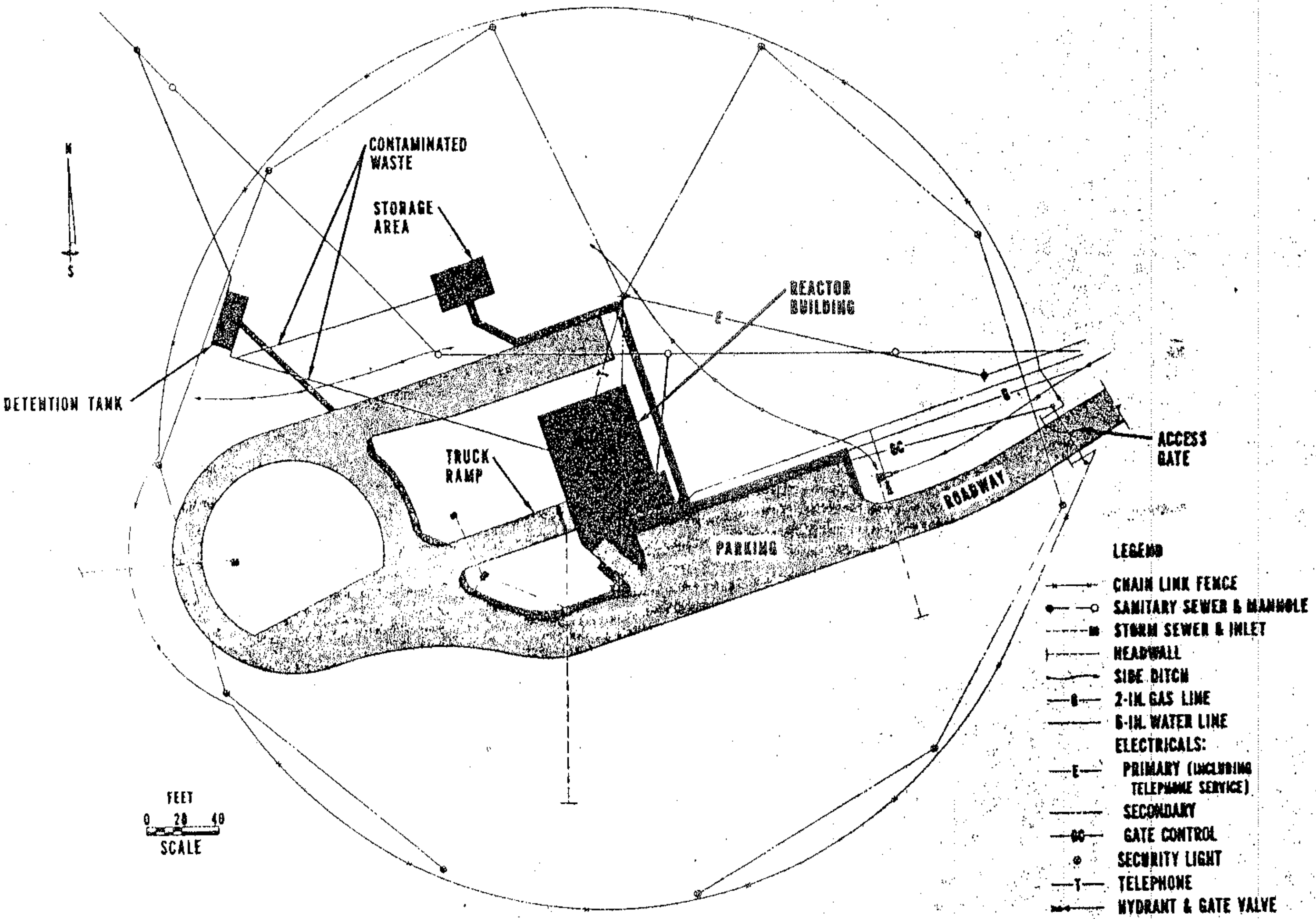
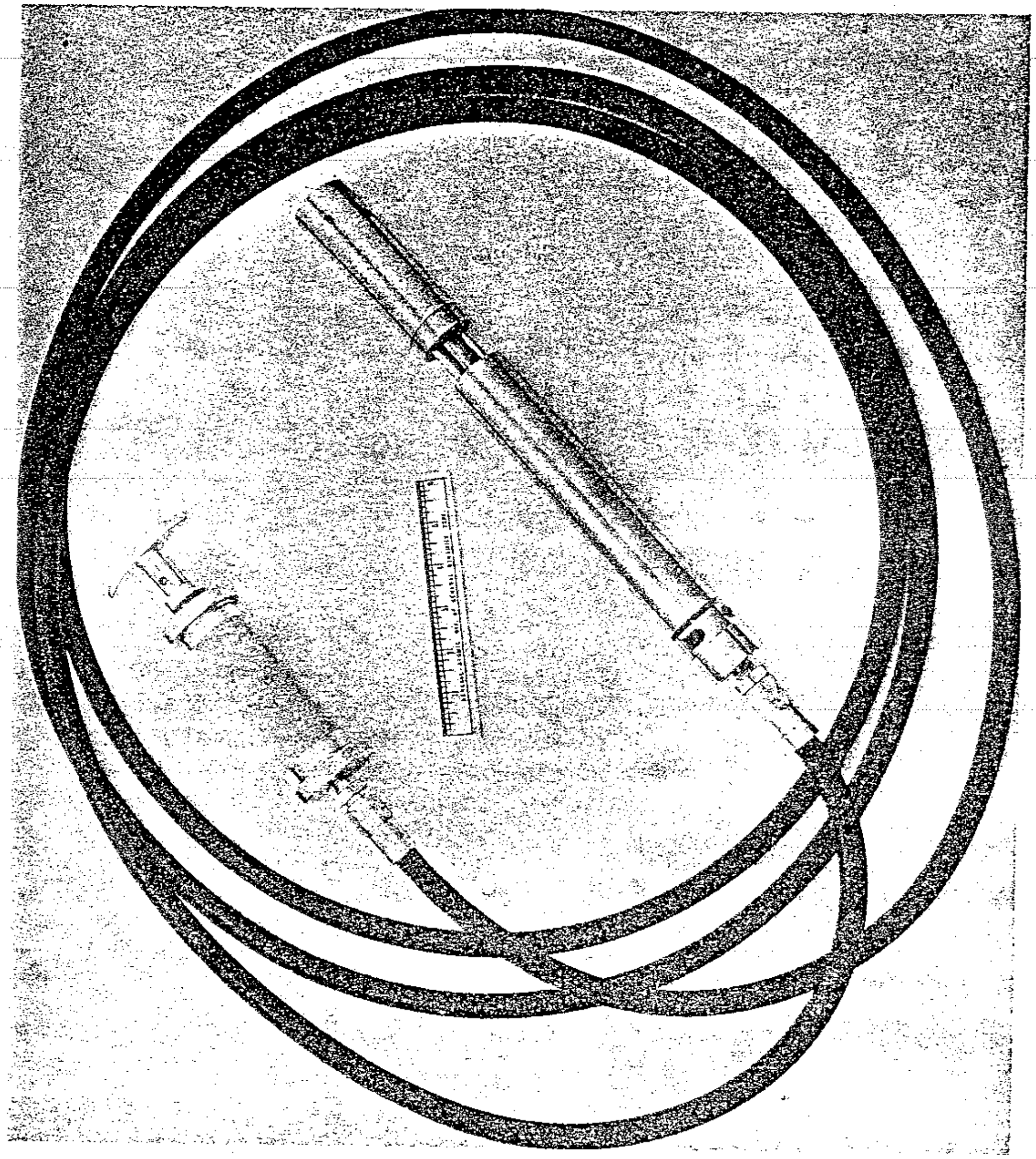


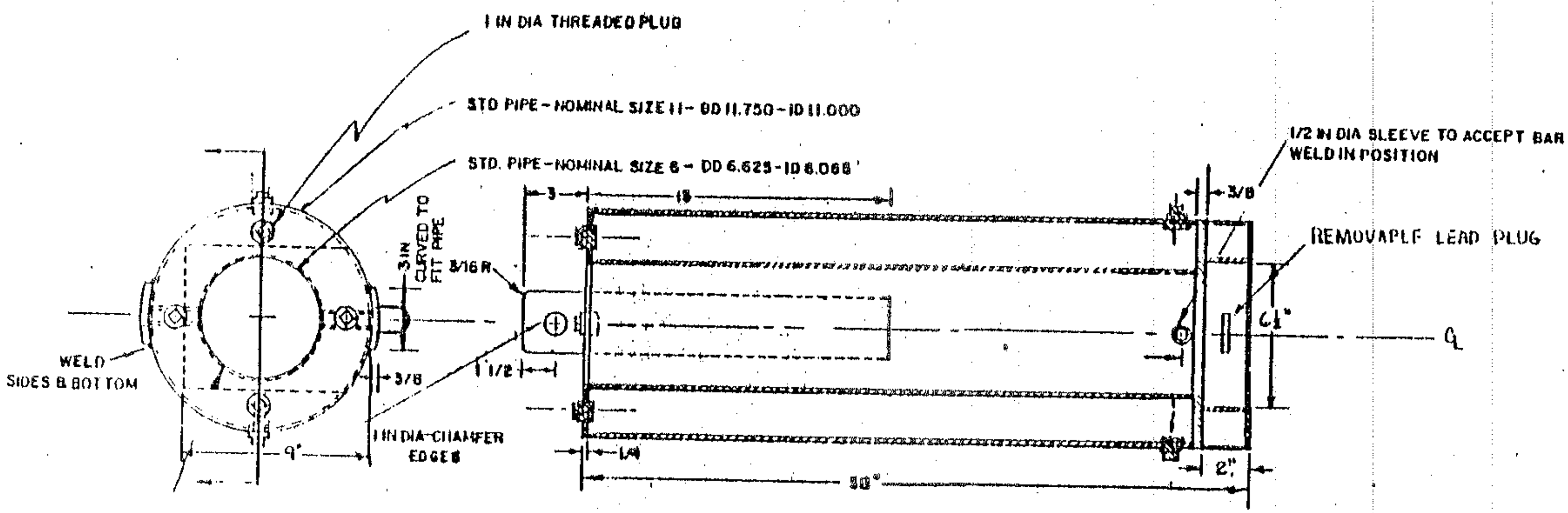
Fig. 4--The DORF site

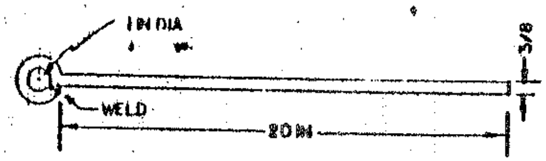


MIS 17

Fig. 5 Fuel Handling Tool

AMC 1031-3





STEEL BAR

NOTE - WELD TOP PLATE OF 1/4 IN STEEL TO BOTH PIPES, HOLDING DIA. OF CENTER HOLE AS CLOSE TO ID OF INNER PIPE AS POSSIBLE. REPEAT SAME FOR BOTTOM PLATE

CAREFULLY CLEAN, PRIME AND PAINT WITH A HIGH GRADE TITANIUM DIOXIDE PIGMENT WHITE PAINT AFTER ASSEMBLY

MATL - MILD STEEL FOR TOP & BOTTOM PLATES & LIFTING SUPPORTS
PIPE - STEEL

APPROX WT ~~480~~ 880 LBS
LOADED WITH LEAD SHOT

PART NO.

CODE IDENT. NO.

SEE ENGINEERING RECORDS

TRANSFER FUEL CASK DORF

U.S. ARMY MATERIEL COMMAND
HARRY DI... LABORAT...
WASHINGTON, D.C. 20438

OPERATING PROCEDURES FOR THE USE OF

FUEL SHIPPING CASK MH-1A

MODIFIED FOR TRIGA FUEL ELEMENTS

1. Loosen the tension on the four (4) turnbuckles and detach them from the cask.
2. Remove the eight bolts which attach the cask to the skid.
3. Lift cask from transport vehicle with crane using lifting yoke attached to cask lifting trunnions. The yoke is supplied with the cask.
4. Place cask on low-boy trailer, disconnect crane from cask yoke or yoke from the cask, and move trailer to basement truck entrance area so that the cask is directly below main floor hatch.
5. Remove covers from lifting holes on cask cover and attach cable sling to remove cover. Use 3 1/2 ton capacity mono-rail hoist.
6. Check "O" ring placement and "O" ring sealing surface on cover for scratches, nicks, gouges, etc. If any defects are discovered, they should be repaired and cask checked for leakage.
7. Check that the four 12 element TRIGA fuel baskets are in place. (NOTE: the baskets may be installed at DORF)
8. Place the DORF seven element transfer cask (T cask) on the reactor pool floor using the 3 1/2 ton mono-rail hoist.
9. Move standard fuel elements from the pool storage racks into transfer cask using TRIGA fuel handling tool until seven elements are loaded. Check each fuel element identification number, transfer cask position and record in log book.
10. Move the transfer cask from the pool, using the mono-rail hoist, to a position adjacent to the shipping cask located at the basement truck entrance area.

11. Move the elements, one at a time, from the transfer cask to a standard fuel element position in the MH-1A cask basket. (NOTE: the standard positions are not color coded). Check the element identification number, basket position and record in log book.

12. Repeat 8 through 12 until all standard fuel elements are loaded for the specific shipment. (One shipment of 44 standard elements to Utah, and one shipment of 38 standard elements to Penn State and Handford, not including FFCR's or TCE's).

13. The following, 14 through 20, are procedures for the TCE loading.

14. Move the transfer cask into the reactor pool for the TCE loading.

15. Move TCE from the pool storage rack into the transfer cask.

16. Raise the transfer cask to the surface of the water so that the swage-lok fitting located 18 inches above the top of the element is a few inches above the water surface.

17. Disconnect the conduit by loosening the top swage-lok nut. Remove the conduit sliding the thermocouple wire through the removed conduit.

18. Fasten TCE handling fixture provided to the swage-lok fitting guiding the thermocouple wire through the slot in the fixture.

19. Tighten swage-lok nut and coil TC wire into about a 6 inch diameter circle.

20. Repeat, 8 through 12, except that the TCE are placed in a TCE basket position identified by the color RED.

21. The following, 22 through 27, are for FFCR loading.

22. Move transfer cask into the reactor pool and load one FFCR.

23. Raise cask to the pool surface so that the top fitting of the FFCR is a few inches above the surface of the water.

24. Remove the shear-pin from the threaded section of the FFCR top fixture and connecting rod using a pin-punch and ballpeen hammer.

25. Unscrew connecting rod from FFCR threaded stud.

26. Thread fuel handling fixture provided on the FFCR threaded stud.

27. Repeat 10 and 11 except place the FFCR in one of the FFCR basket positions colored YELLOW.

28. Lift cask cover with mono-rail hoist using cable sling. Before moving cover over cask, rotate the crane hook until the cover is correctly oriented with respect to the two cover guide pins attached to the top of the cask. Place cover in position on the loaded cask and disengage sling.

29. Replace nuts on cask cover and tighten nuts to 40 ft-lbs torque.

30. Perform leak test described in SAR.

31. Attach lifting yoke to cask lifting trunnions.

32. Move low-boy trailer and cask to transport trailer. Tighten 8 base nuts to 50 ft-lbs torque. Tighten turnbuckles to 100 ft-lbs and turnbuckle lock-nuts to 75 ft-lbs torque.

34. Cask is ready for shipment. Provision for anchoring the cask to the vehicle floor has been provided by holes in the skid beams and by cable anchor pins between the fins at the top corners of the cask.

ENRADMON PLAN

for

DIAMOND ORDNANCE RADIATION FACILITY
(DORF)

1. PURPOSE. The environmental radiological surveillance program is designed to assure that all ionizing radiation and radioactivity levels existing in unrestricted areas in the vicinity of DORF are within permissible limits and as low as reasonably achievable.

2. SCOPE. The plan consists primarily of the measurement of environmental levels of direct ionizing radiation and radionuclide concentrations. The data is intended to demonstrate compliance with AR 385-80. In addition, the plan includes the estimating of DORF stack activity releases, special studies of environmental conditions and other activities required in support of the plan's objective.

3. RESPONSIBILITY. The Health Physics Office, WRAMC, will initiate all environmental monitoring activities, review and evaluate all environmental monitoring data.

4. GENERAL DATA.

a. The Diamond Ordnance Radiation Facility (DORF) is located in the Forest Glen Section of the Walter Reed Army Medical Center (WRAMC), Washington, D.C. The DORF-TRIGA Mark F Reactor is used as a research tool in the study of neutron and gamma irradiation of electrical and electronic components and systems. Through Host-Tenant Agreement, Health Physics, WRAMC, provides Health Physics support to the DORF.

b. Description of facility. The DORF is a TRIGA Mark F Reactor, designed and built by Gulf General Atomics, San Diego, California. It is an inherently safe reactor designed for both steady-state and pulsed operation. The DORF-TRIGA has the capability of steady-state or square-wave operation up to 250 kW for a maximum power generation of 1 MW-hr per day and also pulsed operation resulting in a peak power of 2000 MW with a pulse width of 9.5 milliseconds at half maximum. Additional data concerning this site is contained in the original SAR.

(1) The DORF is located within the metropolitan area of Washington, D.C., at the Forest Glen Section of WRAMC which is eight miles from the center of Washington, D.C. and approximately two miles south of Kensington, MD. The Forest Glen site is an area of approximately 190 acres of rolling, partially wooded and cleared areas, on which are located numerous WRAMC related facilities. The DORF is located near the southern border of the

Terminated operation April 1979
Final close out survey of decommission in Feb 1980.

I. Introduction

A. Background. The Harry Diamond Laboratories (HDL) operates the Diamond Ordnance Radiation Facility (DORF) which utilizes a research reactor with associated experimental equipment. The facility occupies a single remote building on 4.2 acres of the Forest Glen Annex of the Walter Reed Army Medical Center (WRAMC) near Silver Spring, Maryland. An intraservice agreement between the Commanding Officer, WRAMC and Commanding Officer, HDL establishes the WRAMC support services for DORF.

The reactor is the familiar General Atomic Company TRIGA Mark F, moderated by light water and mounted on a track support carriage assembly which can be moved through a 15,000 gallon capacity pool. The reactor core consists of 85 (maximum 87) fuel elements, four control rods, neutron source, and miscellaneous neutron detectors. The fuel elements are composed of zirconium hydride moderator homogeneously combined with 20% enriched uranium fuel. The control system consists of borated graphite safety, shim, regulating and pulse control rods, having either solid aluminum or fuel followers. Experiments are conducted in a 20 x 20 x 8 foot high fast neutron exposure room adjacent to pool, the pool itself, and within the core.

The facility was orginally developed in late 1959 and began operations in September 1961. Modifications applied since then include (1) replacement of the aluminum clad fuel elements with stainless steel clad elements (1964), (2) automatic SCRAM timing (1969), (3) replacement of the poison-followed transient rod with an aluminum follower (1964), (4) replacement of aluminum follower control rods with fuel-followed control rods (1971), and (5) replacement of reactor instrumentation with up-to-date instrumentation (1973).

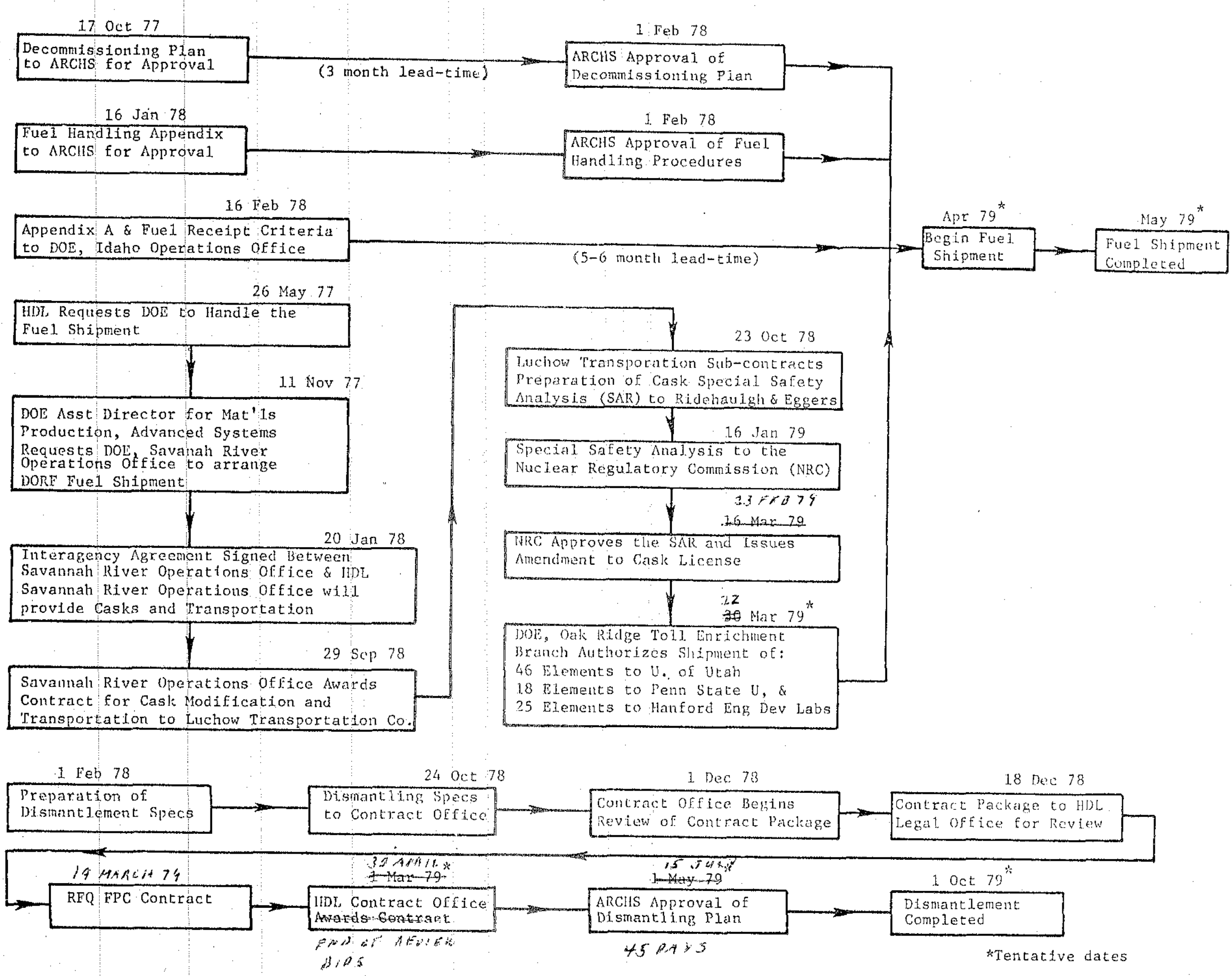
The reactor has the capability of the following modes of operation:

1. Steady-state operation up to 250 kW.

2. Square-wave operation up to 250 kW.

3. Pulse operation resulting in up to a maximum peak power of 2000 MW with a pulse width of 9.5 ms at half maximum.

B. The decision to decommission the DORF reactor is the culmination of an Army reactor utilization study begun in mid-1975 to examine the requirement for the three Army research reactors. This study was done by HDL Nuclear Weapons Effects Program Office (NWEPO) which investigated the following alternatives:



DECOMMISSIONING MILESTONES

[redacted]

4.1.1.6 NEUTRON SOURCES/REACTORS

UNITED STATES

1. Name and Location: Diamond Ordnance Radiation Facility (DORF). The reactor facility was first pulsed in September 1961 and became fully operational in January 1962.

   The facility is located in the State of Maryland at the Forest Glenn Annex of the Walter Reed Army Medical Centre (WRAMC) approximately 8 miles north of the centre of Washington, D.C.

2. Purpose

   The reactor is designed to provide a large source of neutron and gamma radiation in steady state or pulse (msec) conditions for experimental programmes to investigate and determine susceptibility of electronic material to simulated nuclear weapons environment, mechanisms of those effects, and ways and means of developing less susceptible material.

3. Philosophy and Approach for Use

   The design of the DORF TRIGA Mark-F reactor is based on the technology of the proven TRIGA Mark I and Mark II reactors that maintained the inherent safety characteristics of these reactors and whose performance would more specifically match the programme requirements of the Harry Diamond Laboratory experimental users. A variety of high dose irradiation facilities are provided, including (1) a 6.1 m square and 2.44 m high fast neutron exposure room, (2) a re-entrant thimble projecting into the reactor core accessible from the exposure room, and (3) the water filled tank and (4) positions within the TRIGA core in the region of highest flux. The reactor is supported from a track-mounted movable carriage so that it may be operated in any position within the water-filled tank. The extreme end positions within the pool are shielded from one another by swinging lead doors which allows experimental programmes to be conducted at one end of the pool while other experiments are set-up or dismantled in the other

[redacted]

position. The reactor is capable of long steady state operation at power levels up to 250 kW and pulsed operations resulting in a prompt energy release up to 20 MW-sec per pulse (4×10^{14} n/cm^2, E>10 keV).

4. Physical and Mechanical Characteristics

The reactor, exposure room and auxiliary facilities are contained in a three level building constructed of reinforced concrete, structural steel and masonry as shown in Figures 1 and 2. The basement level contains an exposure room which is about 6.1 m square and 2.44 m high. This exposure room is lined with one foot of wood to minimize secondary radiations. Access to this room for equipment is through a tapered and stepped rolling plug door which provides a minimum opening 2.44 m wide and 1.73 m high. Access to the room for special monitoring equipment and instrumentation is provided by several helical conduits which penetrate the concrete shield from the reactor floor above. The basement level also contains a warm storage and decontamination room and a sample preparation area.

The first floor of the facility contains a counting room, and office, and an equipment room, as well as a large area around the reactor shielding structure in which instrumentation for experiments can be set up.

The reactor shield rises 2.13 m above floor level in the centre of the room and encloses the upper portion of the reactor tank. The reactor tank extends down to the basement level and projects into the exposure room, as shown in Figure 1. The mezzanine area over the first floor offices contains the reactor operating area and reactor console, and conference and storage areas. The reactor operating area is arranged so that the reactor operator has an unobstructed view of the reactor from the console. A second mezzanine, adjacent to the equipment room, provides additional space for instrumentation and data handling.

An area for full-sized instrumentation trailers is available adjacent to the building. Electrical power outlets are available at the trailer site. Cables can be run from this area through penetrations into reactor building wall to experimental irradiation facilities.

The reactor is operated by a staff of 8 which includes 3 Reactor and Dosimetry Engineers, 3 Nuclear Reactor Technicians, 1 Electrical and Mechanical Technician and Secretary. Health Physics support is provided by the Walter Reed Army Medical Centre (WRAMC). The DORF staff is available for pre-test planning and consultations to aid in setting up experiments, perform dosimetry and to assist in data acquisition. The facility is available for radiation effects studies approximately 20 days per month.

Personnel dosimetry, health physics monitoring, survey, decontamination and related services are supplied by a health physics staff.

5. Measurements Made and Instrumentation Used

Reactor Dosimetry: The equipment available for neutron dosimetry is summarized below. Neutron fluence and flux data, together with information on the neutron fluence (n/cm^2)-to-gamma exposure dose (Roentgens) ratio is summarized in Table 1. Routine dosimetry is performed by facility personnel and, in general, data can be made available within several days after the irradiation.

DORF Dosimetry Equipment:

(a) Fission foil and threshold detectors together with Boron-10 therman neutron shields.

(b) Gold, cobalt, vanadium, and tantalum thermal and resonance detector foils with cadmium covers.

(c) Lithium-6 and lithium-7 fluoride thermoluminescent gamma dosimeters (for doses of $>$ 500 J/kg).

(d) Cobalt-impregnated glass gamma dosimeters (for doses of 500 J/kg).

Co-60 Source: A pair of NBS-calibrated cobalt 60 sources shielded by a water medium are available at HDL. The larger of the two sources presently provides an exposure dose rate of $258 \cdot 10^{-3}$ A/kg. The smaller source is exactly 1/35th of the strength dose rate of the large source.

Data Processing: Digital data processing is available on an IBM system through the Nuclear Radiation Effects Laboratory of HDL. On site (i.e., at DORF) processing on a "Basic" type system is available through a teletype terminal.

6. Performance Data

The DORF reactor has both pulse and steady state operational capability. The maximum authorized pulse has a peak power of 2,000 megawatts, and the maximum steady state power is 250 kW.

Typical of TRIGA reactors, characteristics of the pulse including prompt energy release, pulse width (FWHM), and the initial reactor period (as well as peak power) depend primarily on the reactivity insertion and the fuel loading of the reactor core. As a result, pulse characteristics can be varied over a wide range within the authorized operating limits of the facility. Table 2 tabulates the pulse characteristics averaged over the experimental data obtained with the DORF reactor. The pulse repetition rate is 5 per hour, providing the experimeter does not have to enter the exposure room for changes or adjustments. Based on operating experience, the uncertainty in reproducing a given pulse characteristic is $\pm$ 10 per cent. A typical pulse is shown in Figure 3.

In steady state operation, the reactor can be operated from milliwatts to a maximum of 1 MW-h per day.

7. Comparison with the Nuclear Environment

The major portion of the fast neutron population of the DORF spectrum lies in the energy range of 200 keV to 2 MeV. Approximately 30 per cent of the neutron population is thermal. The neutron spectra does not contain 14 MeV neutrons. Therefore, the DORF radiation environment closely approximates a moderated fission weapon spectrum.

8. Types of Tests

The irradiation exposure areas available at the DORF provide a maximum flexibility for experimental arrangements. The highest fluence and dose rates are available in the exposure room dimple (25.4 cmx10.16 cmx7.62 cm) and in the in-core positions (3.81 cm diameter). The exposure room is 6.1 m square and 2.44 m high, with a 1.73 m square access plug door opening into the room. A portion of the reactor tank projects into the one wall of the exposure room. The size of the experiment to be irradiated is limited by the dimensions of the plug door opening. Access to the room for special monitoring equipment and instrumentation is provided by several 12.7 cm and 7.6 cm diameter helical conduits which penetrate the concrete shield from the reactor floor above. Additional irradiation positions are available using water-tight cans within the pool adjacent to the reactor core.

Experimental apparatus can perturb the radiation environment in any of the irraddiation exposure facilities and may also effect the reactor reactivity. The type and mass of the irradiated material will determine the magnitude of these effects. Measurements of such effects are provided by the DORF staff on a case-by-case basis.

9. Test Procedures

A proposal for a reactor radiation experiment is required for all irradiation tests. This proposal is reviewed by the Reactor Test Planning Committee, the Reactor Safeguards Committee, and the Reactor Supervisor. The reviews consist of evaluating the experiment proposal for the desirability, validity, the use of proper radiation test procedures, reactor and personnel safety, instrumentation requirements, and scheduling. It is recommended that the user visit the facility beforehand to obtain information regarding the applicability of his experiment to the DORF reactor and that all objectives and safety requirements will be met. Approvals for all experiments are determined locally within the Harry Diamond Laboratories.

10. Availability to other Nations

This facility can be made available to other nations on a non-interference basis upon approval of the Technical Director, HDL and subject to and within the restrictions set forth by security regulations. All requests for facility time or visits must be processed through the normal diplomatic channels. The costs for use of the facility will depend upon the exposure and time requirements. The facility is unclassified and is available to other nations under Quadripartite, TTCP, and NATO terms of reference. Requests for such use must be submitted through appropriate channels.

11. Literature and Reports

(a) Kilminster, D.T., Kappes, J., McNeilly, J.H., "Neutron Flux Measurements at the Diamond Ordnance Facility", *Nuclear Defence Labs* - TR-56 (February 1965).

(b) Gieseler, W.L., McGarry, E.D., McGarrity, J.M., "Technical Specifications for the Diamond Ordnance Radiation Facility Reactor", Harry Diamond Labs Special Report (24th November 1971)

(c) Berman, Philip G., "The Radiation Environment in the Experimental Facilities of the Diamond Ordnance Radiation Facility", Harry Diamond Labs - TR-1307 (1st December 1965).

(d) Leonard, B.E., McManamon, V.L., Lusk, J.A., and Verrelli, D.N. (AFRRI), "Tissue-Equivalent Kerma Variation During and After a Reactor Excursion", *Trans.Am.Nucl.Soc.*, 12, 2, 934 (1969).

(e) McGarrity, James M., "Hazards Summary Report, Addendum", *Harry Diamond Labs. Special Report* (3rd January 1967).

Future Developments

No closures, modifications, replacements or changes in procedures are being considered at this time.

TABLE 1. DORF FLUENCE AND FLUX DATA

Greater than 10 keV fluence and fluxes as measured with boron-10 (1 cm thick) and cadmium (762 μm thick) covered plutonium-239 foils. The core contains 87 fuel elements, including two fuel-followed control rods, and is positioned next to an air-filled exposure room 6,1 m x 6,1 m x 2.44 m high

EXPERIMENTAL LOCATION	PULSE DATA-[1] FLUENCE/PULSE (n/cm^2)	FLUX[2] ($n/cm^2/sec$)	STEADY STATE-[3] FLUENCE/MIN @ 100 kW	RATIO of NEUTRON/GAMMA (n/cm^2 / C/kg)
1. In-core experiment dry tube	3.9×10^{14}	3.5×10^{16}	1.2×10^{14}	0.89×10^{12}
2. In the center of the high-exposure dimple (accessed from the exposure room)	2.0×10^{14}	1.8×10^{16}	6.2×10^{13}	1.86×10^{12}
3. One inch from tank wall in the exposure room in front of high exposure dimple	9.5×10^{13}	8.6×10^{15}	3.0×10^{13}	1.86×10^{12}
4. One inch from tank wall in exposure room 30° to the right of the high-exposure dimple	4.7×10^{13}	4.3×10^{15}	1.5×10^{13}	1.86×10^{12}
5. In pneumatic rabbit facility	2.1×10^{13}	2.0×10^{15}	6.5×10^{13}	1.67×10^{12}
6. In pool at core shroud	4.6×10^{13}	4.2×10^{15}	1.4×10^{13}	0.54×10^{12}

NOTES: 1) For a nominal reactivity insertion of \$ 2.65; values may be increased by 25 % for the present maximum insertion of \$ 3.00. 2) Flux is derived from fluence information by dividing by a pulse half-width of 11 milliseconds. 3) Steady-state power up to 250 kW for 4 hours per day is available. Essentially, 3.2 minutes of 100 KW steady-state irradiation is equivalent to the fluence delivered in a \$ 2.65 pulse.

fluence delivered in a $ 2.65 pulse.

[redacted]

TABLE 2. DORF REACTOR PULSE PARAMETER CHARACTERISTICS

REACTIVITY INSERTIONS $	PEAK POWER MW	PULSE-WIDTH (FWHM) ms	INITIAL REACTOR PERIOD, ms	PROMPT ENERGY RELEASE, MW-s	PROMPT FISSIONS x 10^{17}
1.50	103	40.9	12.5	4.9	1.5
1.60	139	35.4	10.8	5.7	1.8
1.75	215	27.7	7.9	7.0	2.2
2.00	394	20.0	5.6	9.0	2.8
2.25	704	15.0	4.4	11.0	3.5
2.65	1215	11.2	3.4	15.0	4.6
2.75	1485	10.6	3.3	16.7	5.2
2.85	1647	10.3	3.1	16.9	5.3
2.90	1697	10.3	2.8	18.2	5.6
3.00	1863	9.6	2.8	19.4	6.0

[redacted]

for: weame

DRCSG-E (3 Apr 80) 1st Ind
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Headquarters, US Army Materiel Development and Readiness Command
5001 Eisenhower Avenue, Alexandria, VA 22333 9 Apr 80

TO: Commander, US Army Electronics Research and Development Command
ATTN: DELHD-N-RBI, Adelphi, MD 20783

Subject report has been reviewed by this office and is forwarded for information and appropriate action.

FOR THE COMMANDER:

CARL W. JOHNSON
MAJOR, MSC
MEDICAL ENTOMOLOGIST
OFFICE OF THE SURGEON

1 Incl
nc

ROBERT T. CUTTING, M.D.
Colonel (P), MC
Command Surgeon

CF:
DRCSF-P
DRCSA-NS
DRXOS-ES
DRCIS-A

DEPARTMENT OF THE ARMY
U. S. ARMY ENVIRONMENTAL HYGIENE AGENCY
ABERDEEN PROVING GROUND, MARYLAND 21010

Mr. Lodde/cw/AUTOVON
584-3526

HSE-RH/WP

3 APR 1980

SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Commander
US Army Materiel Development and
Readiness Command
ATTN: DRCSG
5001 Eisenhower Avenue
Alexandria, VA 22333

1. AUTHORITY. Letter, DELHD-N-RBI, Harry Diamond Laboratories, 2 November 1979, subject: Request for a Radiological Health Special Study, and indorsement thereto.

2. PURPOSE. This special study was performed to determine the presence and extent of radioactive contamination and whether the facility met the radioactive contamination levels stated in Nuclear Regulatory Commission, Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974, following decontamination.

3. GENERAL.

a. This radiation protection special study was conducted by Mr. Gordon M. Lodde, Health Physicist, and 2LT Roger M. Davis, Jr., Health Physics Division, this Agency, during the period 25-28 February 1980.

b. An entrance interview and an exit briefing were provided to Mr. Charles Ware, Contracting Officer's Representative, Harry Diamond Laboratories.

4. FINDING.

a. The results of smear surveys are provided in Inclosure 1.

b. The results of concrete analysis are provided in Inclosure 2.

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c. Surveys by direct radiation measurements indicated that the highest radiation values were obtained on the north, south, and west walls of the exposure room. The values ranged from 20-400 microroentgen per hour (μR/h) on contact as measured with an Eberline, Model PRM-7, Micro-R-Meter and up to 350 μR/Hr when measured with a Victoreen, Model 440, Ionization Chamber. These two methods of radiation measurements are in close agreement.

5. DISCUSSION.

a. Samples were taken from the wastewater holding tanks and the sewage system down stream from the holding tanks.

b. Core samples were taken off site and soil and vegetation samples were taken both on and off site.

c. The final report will be forwarded in about 60 days following analysis of the samples.

6. CONCLUSION. A review of the findings indicated that after decontamination the facility conformed to the requirements of Regulatory Guide 1.86.

7. RECOMMENDATION. None

FOR THE COMMANDER:

2 Incl
as

FRANK E. McDERMOTT
COL, MSC
Director, Radiation and
Environmental Sciences

CF:
Cdr, ERADCOM
Cdr, HSC (HSPA-P)

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

RESULTS OF ANALYZING WIPE TEST SAMPLES

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm^2 Gross Alpha Activity	Gross Beta Activity
1	L244	< 1.4	4.4 ± 2.5
2	L245	< 1.4	< 2.5
3	L246	< 1.4	< 2.5
4	L247	< 1.4	< 2.5
5	L248	< 1.4	< 2.5
6	L249	< 1.4	< 2.5
7	L250	< 1.4	< 2.5
8	L251	< 1.4	2.8 ± 2.0
9	L252	< 1.4	< 2.5
10	L253	< 1.4	6.0 ± 2.7
11	L254	< 1.4	2.6 ± 2.0
12	L255	< 1.4	< 2.5
13	L256	< 1.4	< 2.5
14	L257	< 1.4	< 2.5
15	L258	< 1.4	< 2.5
16	L259	< 1.4	3.6 ± 1.9
17	L260	< 1.4	< 2.5
18	L261	< 1.4	< 2.5
19	L262	< 1.4	14.6 ± 3.7
20	L263	4.7 ± 2.4	14.0 ± 3.6
21	L264	< 1.4	< 2.5
22	L265	< 1.4	6.2 ± 2.3
23	L266	< 1.4	7.0 ± 2.6
24	L267	3.2 ± 1.9	< 2.5
25	L268	< 1.4	5.2 ± 2.4
26	L269	< 1.4	< 2.5
27	L270	< 1.4	3.0 ± 2.0
28	L271	< 1.4	< 2.5
29	L272	< 1.4	< 2.5

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SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm^2 Gross Alpha Activity	Gross Beta Activity
30	L273	< 1.4	3.2 ± 2.2
31	L274	< 1.4	9.8 ± 3.2
32	L275	< 1.4	3.2 ± 2.3
33	L276	< 1.4	< 2.5
34	L277	< 1.4	< 2.5
35	L278	< 1.4	3.2 ± 2.4
36	L279	< 1.4	3.2 ± 2.1
37	L280	< 1.4	5.0 ± 2.4
38	L281	< 1.4	4.8 ± 2.3
39	L282	< 1.4	< 2.5
40	L283	< 1.4	< 2.5
41	L284	< 1.4	3.4 ± 2.1
42	L285	< 1.4	< 2.5
43	L286	< 1.4	< 2.5
44	L287	< 1.4	< 2.5

Alphus L. Jones
ALPHUS L. JONES, Chief
Radl & Biol Chem Div, USAEHA

HSE-RH/WP
SUBJECT: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

INTERIM RESULTS OF ANALYZING CONCRETE SAMPLES

Sample Identification	RCB Lab No.	Microcurie per Gram ±2 Standard Deviations: Europium-152 Activity	Europium-154 Activity	Cobalt-60 Activity
EX-N	RC1	$3.5 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.8 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$1.0 \times 10^{-5} \pm 0.4 \times 10^{-6}$
EX-S	RC2	$5.9 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$4.5 \times 10^{-6} \pm 0.8 \times 10^{-6}$	$3.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
ES In Pool	RC3	$1.6 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$1.4 \times 10^{-6} \pm 0.4 \times 10^{-6}$	$5.4 \times 10^{-6} \pm 0.3 \times 10^{-6}$
ES-W	RC4	$2.8 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.2 \times 10^{-6} \pm 0.5 \times 10^{-6}$	$1.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
EX LIFT-S	RC5	$1.1 \times 10^{-4} \pm 0.2 \times 10^{-5}$	$7.9 \times 10^{-6} \pm 0.9 \times 10^{-6}$	$3.0 \times 10^{-5} \pm 0.1 \times 10^{-5}$

Alphus L. Jones

ALPHUS L. JONES, Chief
Radl & Biol Chem Div, USAEHA

DEPARTMENT OF THE ARMY
U. S. ARMY ENVIRONMENTAL HYGIENE AGENCY
ABERDEEN PROVING GROUND, MARYLAND 21010

2LT Davis/ldr/AUTOVON
584-3526

02 SEP 1980

HSE-RH/WP

6 AUG 1980

SUBJECT: Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Commander
US Army Materiel Development and
Readiness Command
ATTN: DRCSG
5001 Eisenhower Avenue
Alexandria, VA 22333

1. AUTHORITY.

a. AR 40-5, Health and Environment, 25 September 1974.

b. Letter, DELHD-N-RBI, Harry Diamond Laboratories, 2 November 1979, subject: Request for a Radiological Health Special Study, and indorsements thereto.

2. REFERENCES.

a. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974.

b. Letter, HSE-RH/WP, this Agency, 3 April 1980, subject: Preliminary Report, Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980.

3. PURPOSE. This special study was performed to determine the presence and extent of radioactive contamination and whether the facility met the radioactive contamination levels stated in NRC Regulatory Guide 1.86, following decontamination.

4. GENERAL.

a. This radiation protection special study was conducted by Mr. Gordon M. Lodde, DAC, Health Physicist, and 2LT Roger M. Davis, Jr., MSC, Health Physics Division, this Agency, during the period 25-28 February 1980.

b. An entrance interview and an exit briefing were provided to Mr. Charles Ware, Contracting Officer's Representative, Harry Diamond Laboratories.

c. This report includes all results presented in the Preliminary Report.

5. FINDINGS.

a. The results of smear surveys are provided in Inclosure 1.

b. The results of concrete analysis are provided in Inclosure 2.

c. The results of water analysis are provided in Inclosure 3.

d. The results of soil analysis are provided in Inclosure 4.

e. The results of vegetation analysis are provided in Inclosure 5.

f. Photographs of the decontaminated DORF are provided in Inclosure 6.

g. Surveys by direct radiation measurements indicated that the highest radiation values were obtained on the north, south, and west walls of the exposure room. The values ranged from 20-400 microroentgens per hour (μR/h) on contact as measured with an Eberline, Model PRM-7, Micro-R-Meter and up to 350 μR/Hr when measured with a Victoreen, Model 440, Ionization Chamber. These two methods of radiation measurements are in close agreement.

6. CONCLUSION. A review of the findings indicated that after decontamination the facility conformed to the requirements of Regulatory Guide 1.86.

7. RECOMMENDATIONS. None.

FOR THE COMMANDER:

6 Incl
as

Frank E. McDermott
FRANK E. McDERMOTT
COL, MSC
Director, Radiation and
Environmental Sciences

CF:
HQDA (DASG-PSP)
~~Cdr,~~ ERADCOM
Cdr, HSC (HSPA-P)
Supt, AHS (HSA-IPM)
Cdr, WRAMC (PVNTMED Actv) (2 cy)
C, USAEHA-Rgn Div North
Cdr, Harry Diamond Labs (2 cy)

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RESULTS OF ANALYZING WIPE TEST SAMPLES

Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm^2 Gross Alpha Activity	Gross Beta Activity
1	L244	< 1.4	4.4 ± 2.5
2	L245	< 1.4	< 2.5
3	L246	< 1.4	< 2.5
4	L247	< 1.4	< 2.5
5	L248	< 1.4	< 2.5
6	L249	< 1.4	< 2.5
7	L250	< 1.4	< 2.5
8	L251	< 1.4	2.8 ± 2.0
9	L252	< 1.4	< 2.5
10	L253	< 1.4	6.0 ± 2.7
11	L254	< 1.4	2.6 ± 2.0
12	L255	< 1.4	< 2.5
13	L256	< 1.4	< 2.5
14	L257	< 1.4	< 2.5
15	L258	< 1.4	< 2.5
16	L259	< 1.4	3.6 ± 1.9
17	L260	< 1.4	< 2.5
18	L261	< 1.4	< 2.5
19	L262	< 1.4	14.6 ± 3.7
20	L263	4.7 ± 2.4	14.0 ± 3.6
21	L264	< 1.4	< 2.5
22	L265	< 1.4	6.2 ± 2.3
23	L266	< 1.4	7.0 ± 2.6
24	L267	3.2 ± 1.9	< 2.5
25	L268	< 1.4	5.2 ± 2.4
26	L269	< 1.4	< 2.5
27	L270	< 1.4	3.0 ± 2.0
28	L271	< 1.4	< 2.5
29	L272	< 1.4	< 2.5

Incl 1

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Sample Identification	RCB Lab No.	Disintegrations per Minute ±2 Standard Deviations/100 cm^2 Gross Alpha Activity	Gross Beta Activity
30	L273	< 1.4	3.2 ± 2.2
31	L274	< 1.4	9.8 ± 3.2
32	L275	< 1.4	3.2 ± 2.3
33	L276	< 1.4	< 2.5
34	L277	< 1.4	< 2.5
35	L278	< 1.4	3.2 ± 2.4
36	L279	< 1.4	3.2 ± 2.1
37	L280	< 1.4	5.0 ± 2.4
38	L281	< 1.4	4.8 ± 2.3
39	L282	< 1.4	< 2.5
40	L283	< 1.4	< 2.5
41	L284	< 1.4	3.4 ± 2.1
42	L285	< 1.4	< 2.5
43	L286	< 1.4	< 2.5
44	L287	< 1.4	< 2.5

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Radl & Biol Chem Div, USAEHA

(See Figures 1-4 for smear sampling locations.)

Incl 1
2

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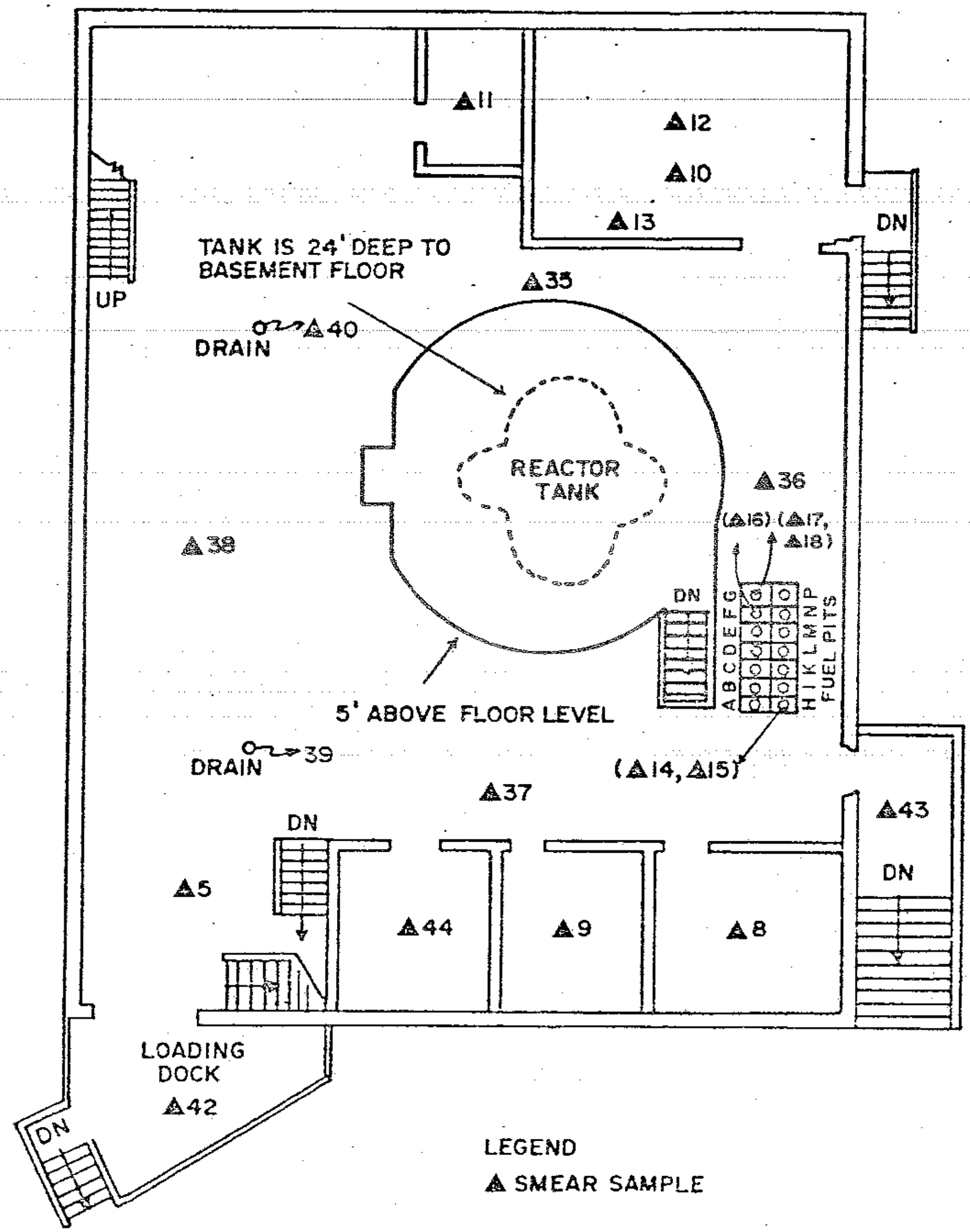
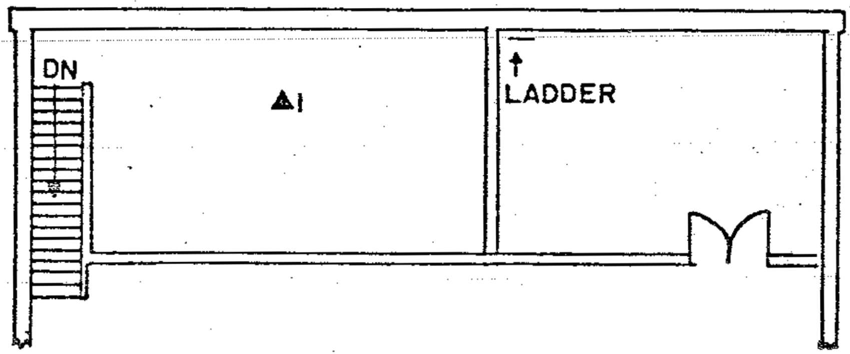
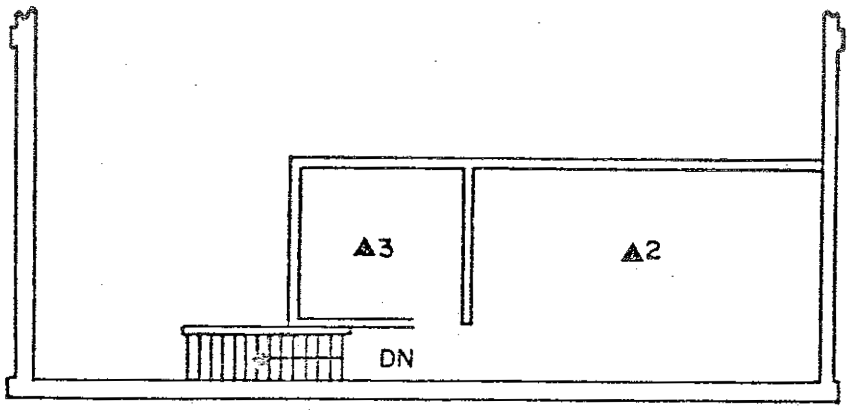


FIRST FLOOR PLAN

Figure 1

Incl 1
3





LEGEND

▲ SMEAR SAMPLE

MEZZANINE PLAN

Figure 2

Incl 1
4

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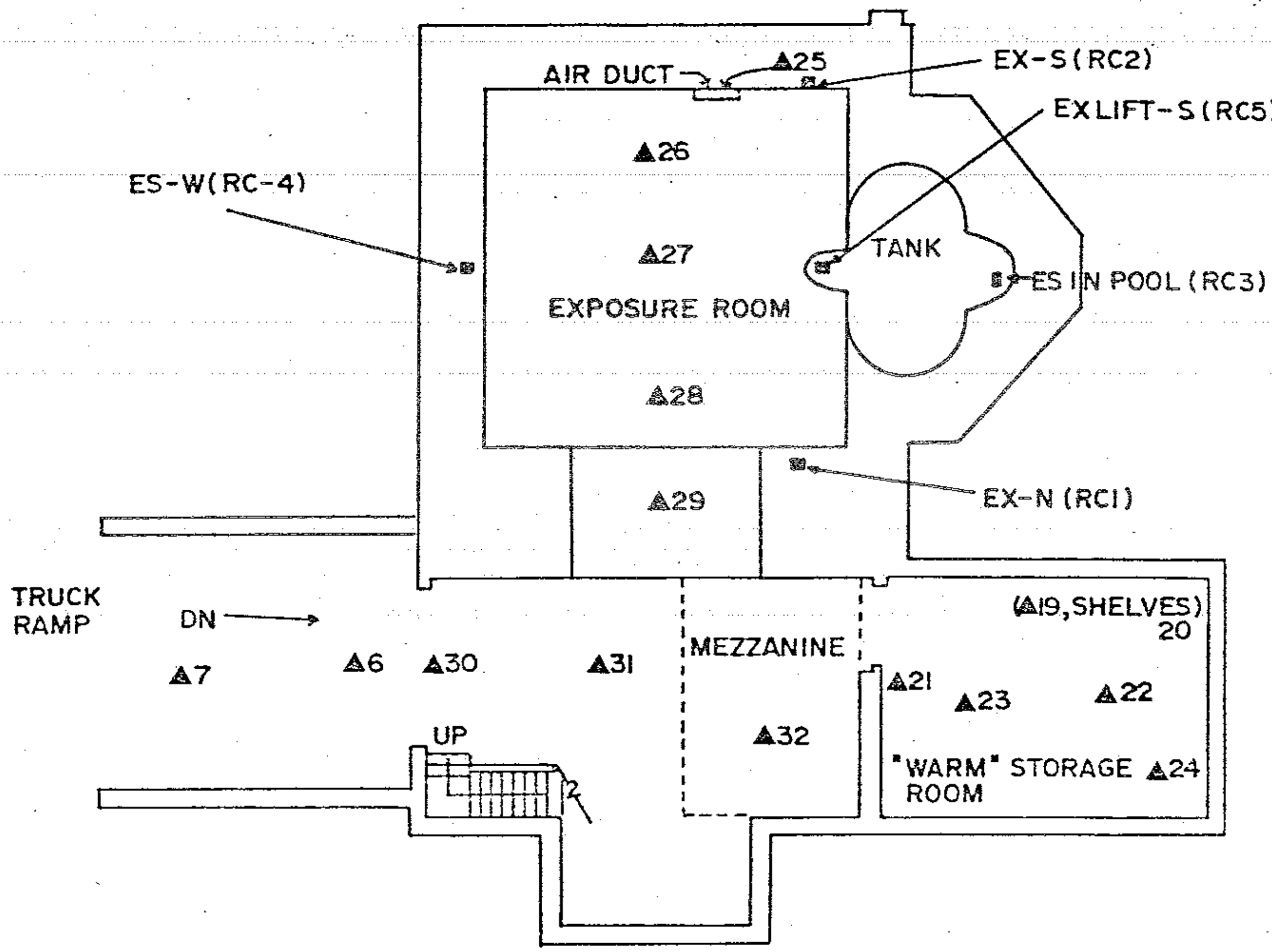

BASEMENT PLAN

LEGEND
▲ SMEAR SAMPLE
■ CONCRETE SAMPLE

Figure 3

Incl 1

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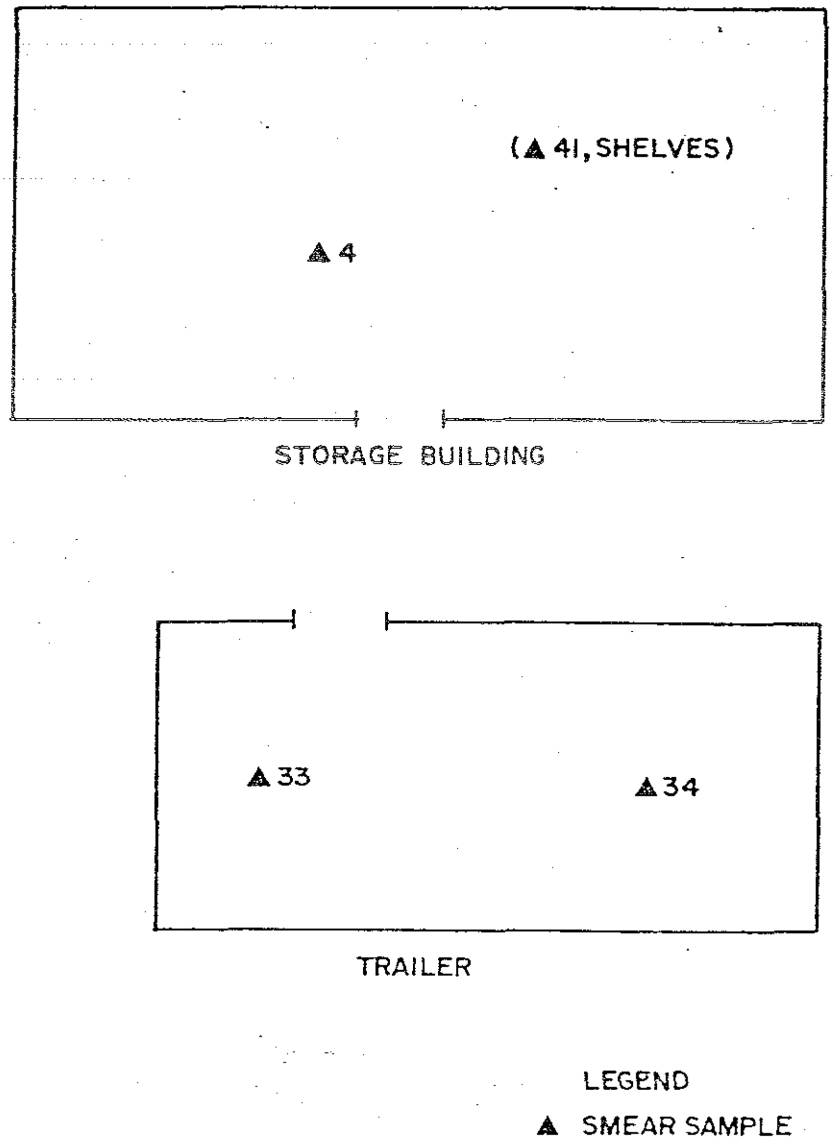

SMEAR SAMPLING AT TRAILER AND STORAGE BUILDING

Figure 4

Incl 1
6

Incl 2

INTERIM RESULTS OF ANALYZING CONCRETE SAMPLES

Sample Identification	RCB Lab No.	Microcurie per Gram ±2 Standard Deviations: Europium-152 Activity	Europium-154 Activity	Cobalt-60 Activity
EX-N	RC1	$3.5 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.8 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$1.0 \times 10^{-5} \pm 0.4 \times 10^{-6}$
EX-S	RC2	$5.9 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$4.5 \times 10^{-6} \pm 0.8 \times 10^{-6}$	$3.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
ES In Pool	RC3	$1.6 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$1.4 \times 10^{-6} \pm 0.4 \times 10^{-6}$	$5.4 \times 10^{-6} \pm 0.3 \times 10^{-6}$
ES-W	RC4	$2.8 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$2.2 \times 10^{-6} \pm 0.5 \times 10^{-6}$	$1.4 \times 10^{-5} \pm 0.1 \times 10^{-5}$
EX LIFT-S	RC5	$1.1 \times 10^{-4} \pm 0.2 \times 10^{-5}$	$7.9 \times 10^{-6} \pm 0.9 \times 10^{-6}$	$3.0 \times 10^{-5} \pm 0.1 \times 10^{-5}$

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RESULTS OF ANALYZING CONCRETE SAMPLES*

Sample Identification	RCB Lab No.	Microcurie per Gram ±2 Standard Deviations Cesium-134 Activity
EX-N	RC1	$2.2 \times 10^{-7} \pm 1.6 \times 10^{-7}$
EX-S	RC2	$5.6 \times 10^{-7} \pm 2.4 \times 10^{-7}$
ES in Pool	RC3	$< 1.9 \times 10^{-7}$
ES-W	RC4	$< 2.7 \times 10^{-7}$
EX Lift-S	RC5	$5.8 \times 10^{-7} \pm 2.8 \times 10^{-7}$

*Other gamma activities reported earlier in the interim report.

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(See Figures 2-4 Inclosure 1; and Photograph 2-7, Inclosure 6, for concrete sampling locations.)

Incl 2

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RESULTS OF ANALYZING WATER SAMPLES

Sample Identification	RCB Lab No.	Microcurie per Milliliter ±2 Standard Deviations Cobalt-60 Activity	Tritium Activity
1	W229	$< 2.8 \times 10^{-8}$	$< 4.2 \times 10^{-7}$
2 A-3-B	W230	$< 2.1 \times 10^{-8}$	$< 4.2 \times 10^{-7}$
3 A-2-B	W231	$< 2.1 \times 10^{-8}$	$3.9 \times 10^{-6} \pm 0.3 \times 10^{-6}$
4 A-1-B	W232	$< 2.4 \times 10^{-8}$	$1.4 \times 10^{-6} \pm 0.3 \times 10^{-6}$
WA-1-AA	W233	$< 2.1 \times 10^{-8}$	$1.3 \times 10^{-6} \pm 0.3 \times 10^{-6}$
WA-2-AA	W234	$< 1.6 \times 10^{-8}$	$4.5 \times 10^{-7} \pm 2.6 \times 10^{-7}$
WA-3-AA	W235	$< 1.8 \times 10^{-8}$	$< 4.2 \times 10^{-7}$
W-505	W236	$< 1.8 \times 10^{-8}$	$< 4.2 \times 10^{-7}$

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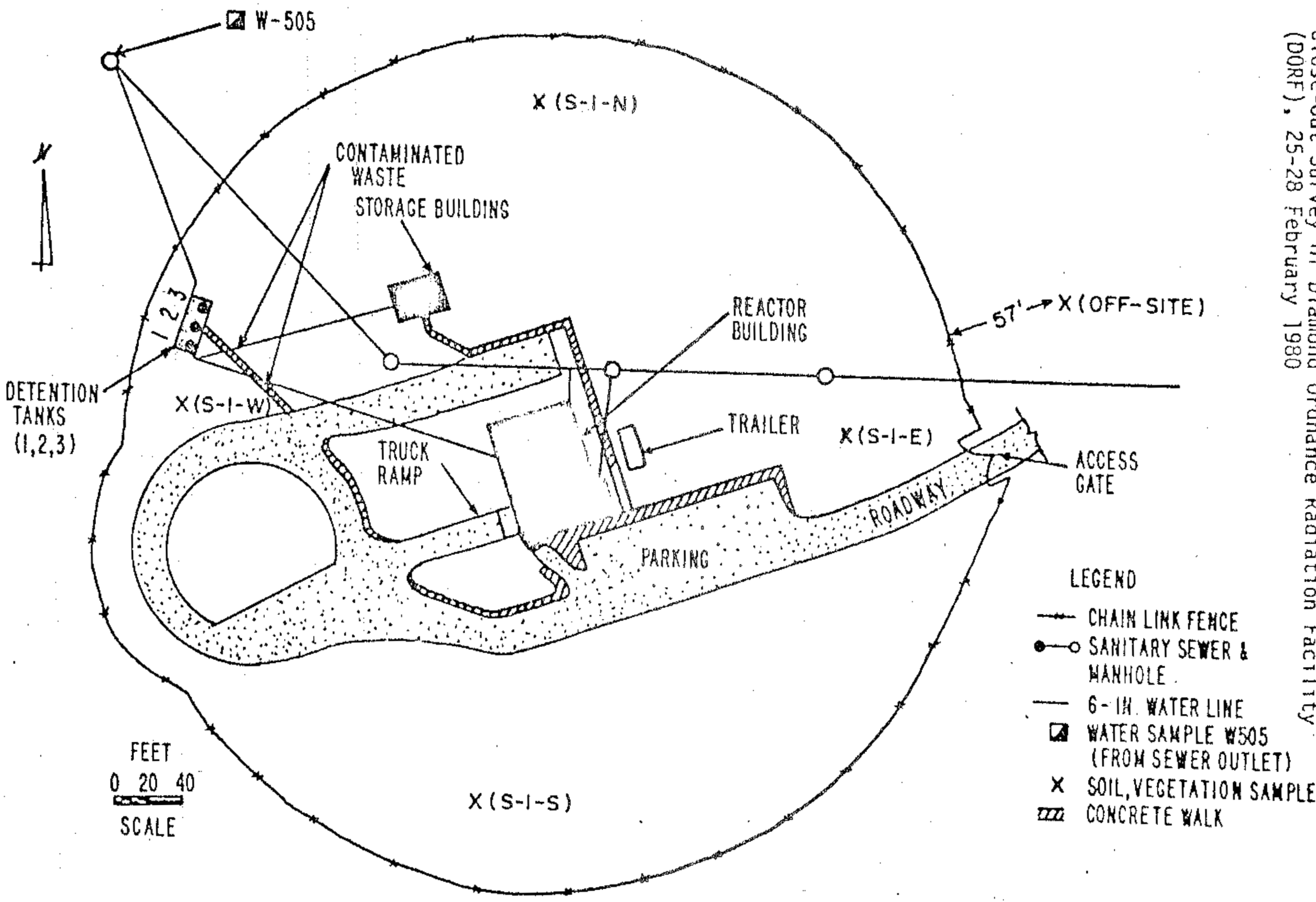
(See Figures 1 and 2 for water sampling locations.)

Incl 3

Incl 3



THE DORF SITE
SOIL SAMPLES AND WATER SAMPLE (W-505)

Figure 1

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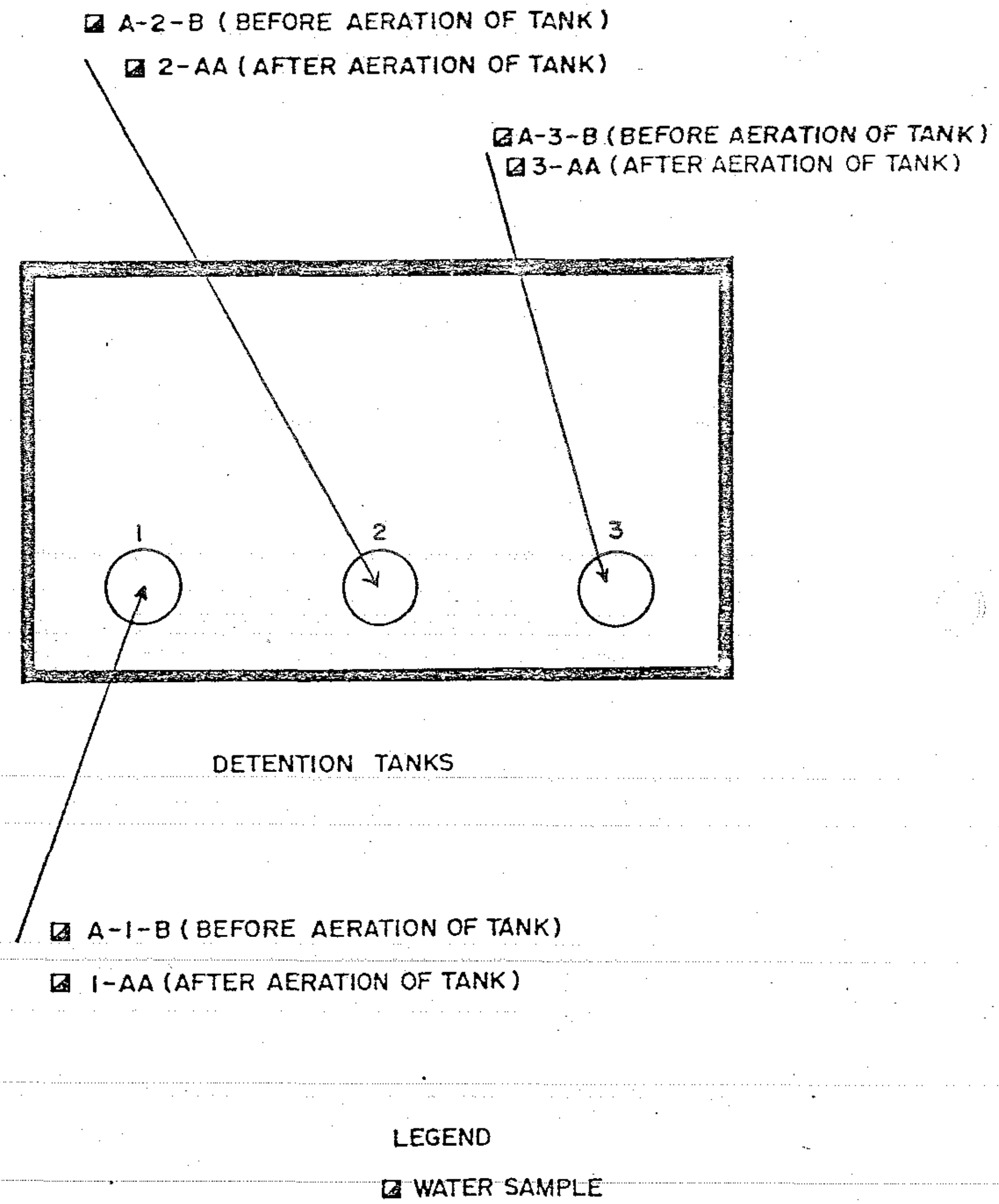

WATER SAMPLING AT DETENTION TANKS

Figure 2

Incl 3

Inc 4

RESULTS OF ANALYZING SOIL SAMPLES (N.N.E. of Stack Approx. 18 Ft. Outside of Fence)

Sample Identification	RCB Lab No.	Microcurie per Gram ± 2 Standard Deviations: Lead-212	Lead-214	Potassium-40	Bismuth-214
#1 Bottom, 8" Depth	S4	$1.6 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$9.9 \times 10^{-7} \pm 2.2 \times 10^{-7}$	$1.9 \times 10^{-5} \pm 0.1 \times 10^{-5}$	$1.3 \times 10^{-6} \pm 0.3 \times 10^{-6}$
#2, 7" Depth	S4	$1.6 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.0 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.9 \times 10^{-5} \pm 0.3 \times 10^{-5}$	$1.1 \times 10^{-6} \pm 0.3 \times 10^{-6}$
#3, 6" Depth	S4	$1.9 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.2 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$2.1 \times 10^{-5} \pm 0.3 \times 10^{-5}$	$1.2 \times 10^{-6} \pm 0.2 \times 10^{-6}$
#4, 5" Depth	S4	$2.0 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.5 \times 10^{-6} \pm 0.3 \times 10^{-6}$	$2.3 \times 10^{-5} \pm 0.3 \times 10^{-5}$	$1.3 \times 10^{-6} \pm 0.3 \times 10^{-6}$
#5, 4" Depth	S4	$1.8 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.3 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.7 \times 10^{-5} \pm 0.2 \times 10^{-5}$	$1.1 \times 10^{-6} \pm 0.2 \times 10^{-6}$
#6, 3" Depth	S4	$1.5 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.3 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.6 \times 10^{-5} \pm 0.2 \times 10^{-5}$	$1.2 \times 10^{-6} \pm 0.2 \times 10^{-6}$
#7, 2" Depth	S4	$1.7 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.2 \times 10^{-6} \pm 0.3 \times 10^{-6}$	$1.8 \times 10^{-5} \pm 0.3 \times 10^{-5}$	$1.1 \times 10^{-6} \pm 0.3 \times 10^{-6}$
#8 Top, 1" Depth	S4	$1.6 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.2 \times 10^{-6} \pm 0.2 \times 10^{-6}$	$1.7 \times 10^{-5} \pm 0.3 \times 10^{-5}$	$9.4 \times 10^{-7} \pm 2.5 \times 10^{-7}$

Sample Identification	RCB Lab No.	Actinium-228	Radium-226	Cesium-137
#1 Bottom, 8" Depth	S4	$1.8 \times 10^{-6} \pm 0.5 \times 10^{-6}$	$4.4 \times 10^{-6} \pm 1.6 \times 10^{-6}$	$< 1.0 \times 10^{-7}$
#2, 7" Depth	S4	$1.1 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$2.6 \times 10^{-6} \pm 1.5 \times 10^{-6}$	$< 1.5 \times 10^{-7}$
#3, 6" Depth	S4	$1.8 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$6.5 \times 10^{-6} \pm 1.7 \times 10^{-6}$	$< 1.3 \times 10^{-7}$
#4, 5" Depth	S4	$2.3 \times 10^{-6} \pm 0.6 \times 10^{-6}$	$4.2 \times 10^{-6} \pm 2.2 \times 10^{-6}$	$< 1.3 \times 10^{-7}$
#5, 4" Depth	S4	$1.7 \times 10^{-6} \pm 0.4 \times 10^{-6}$	$3.1 \times 10^{-6} \pm 1.6 \times 10^{-6}$	$< 1.1 \times 10^{-7}$
#6, 3" Depth	S4	$1.5 \times 10^{-6} \pm 0.4 \times 10^{-6}$	$4.3 \times 10^{-6} \pm 1.9 \times 10^{-6}$	$< 1.1 \times 10^{-7}$
#7, 2" Depth	S4	$1.8 \times 10^{-6} \pm 0.8 \times 10^{-6}$	$2.9 \times 10^{-6} \pm 2.2 \times 10^{-6}$	$< 1.3 \times 10^{-7}$
#8, Top, 1" Depth	S4	$1.6 \times 10^{-6} \pm 0.5 \times 10^{-6}$	$2.9 \times 10^{-6} \pm 1.3 \times 10^{-6}$	$< 1.4 \times 10^{-7}$

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RESULTS OF ANALYZING VEGETATION SAMPLES

Sample Identification	RCB Lab No.	Microcurie per Gram Dry Vegetation ±2 Standard Deviation Cesium-137 Activity
S-1-N	V1	$< 2.5 \times 10^{-6}$
S-1-S	V2	$2.3 \times 10^{-6} \pm 1.1 \times 10^{-6}$
S-1-E	V3	$< 1.6 \times 10^{-6}$
S-1-W	V4	$< 1.9 \times 10^{-6}$
Off-Site	V5	$< 1.8 \times 10^{-6}$

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(See Figures 1 and 3 Inclosure 3, for vegetation sampling locations.)

Incl 5

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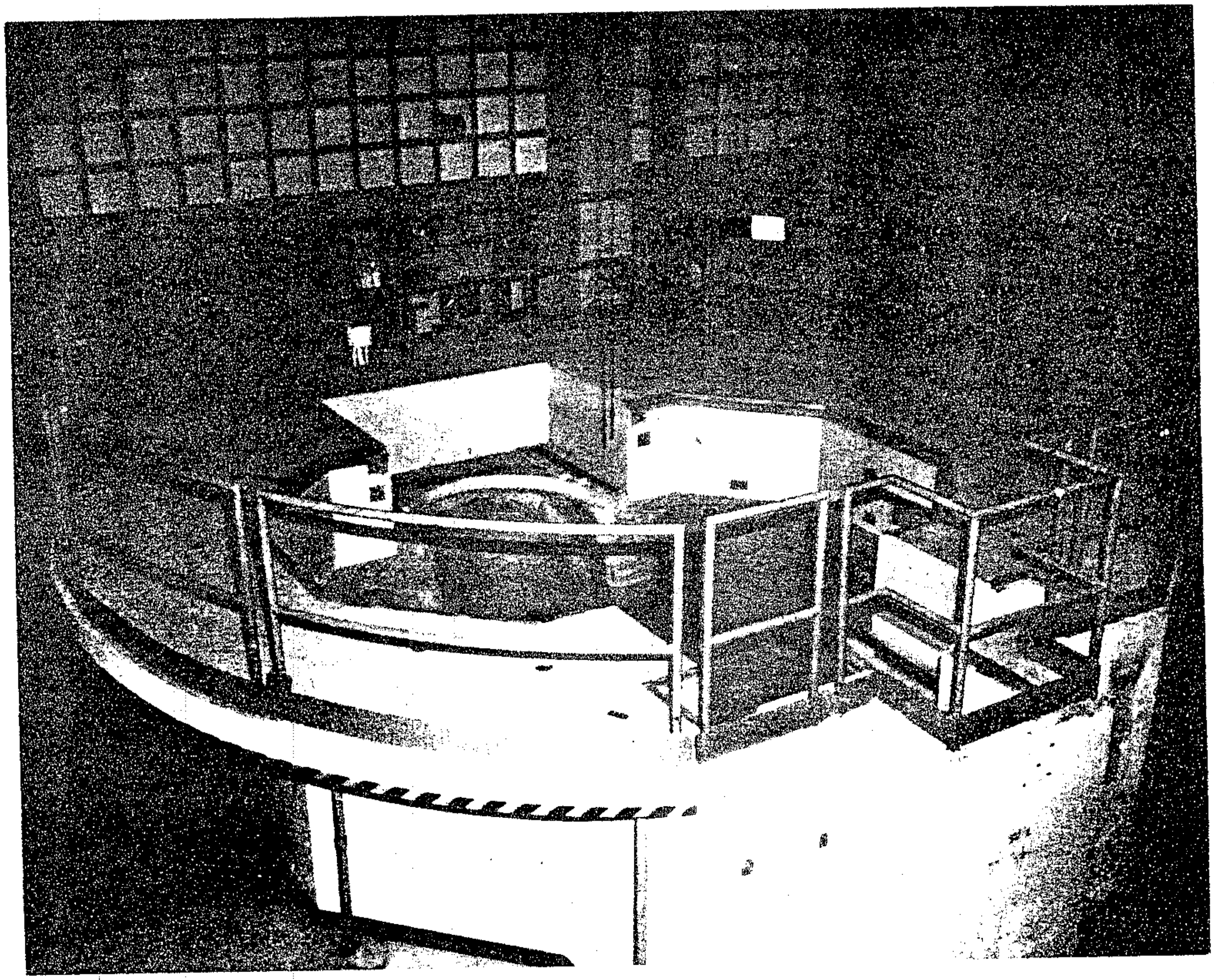
Incl 6



REACTOR TANK--FIRST FLOOR

Photograph 1

HSE-RH/WP
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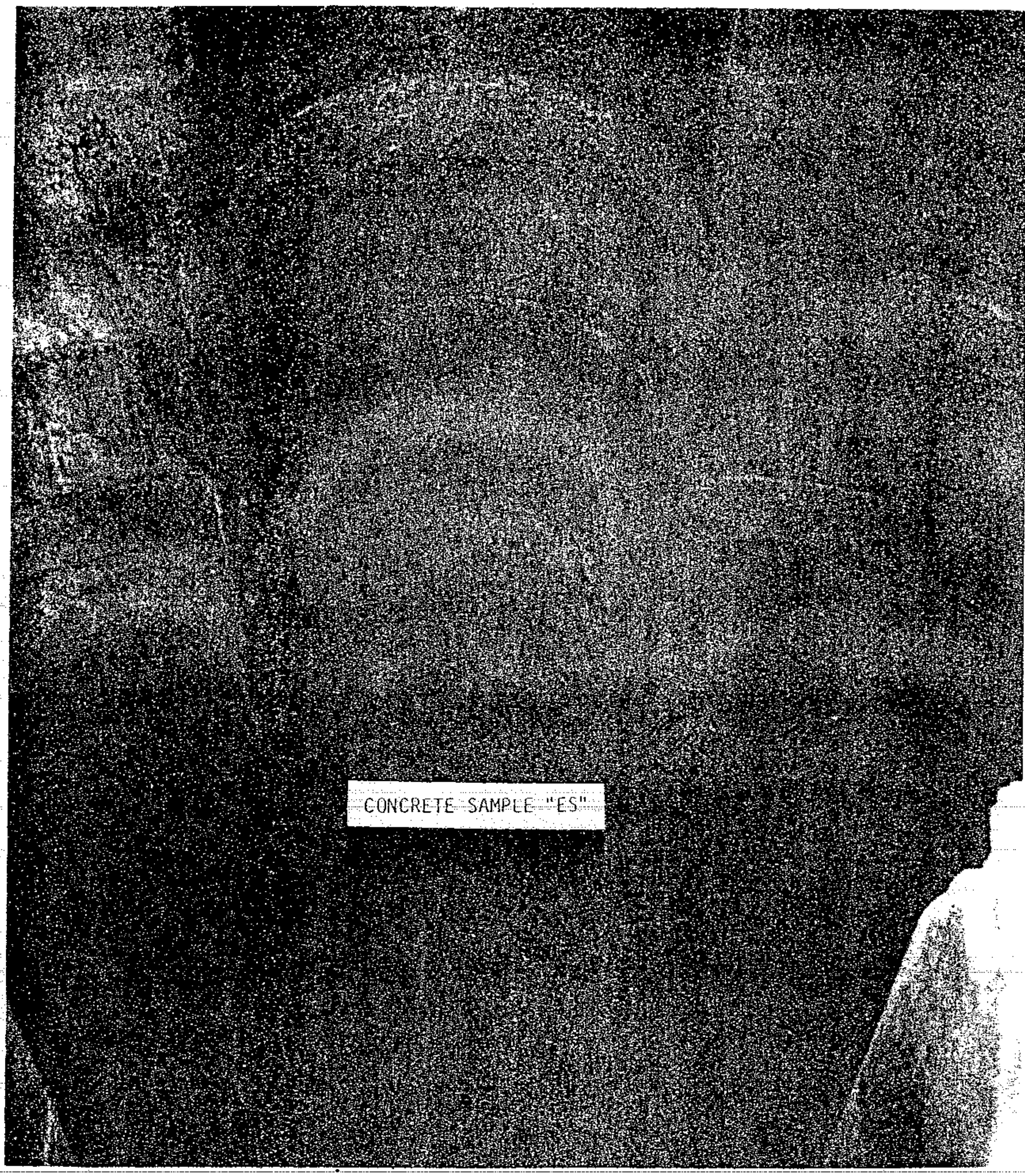

Photograph 2 REACTOR TANK VIEWED FROM FIRST FLOOR

Incl 6

HSE-RH/WP
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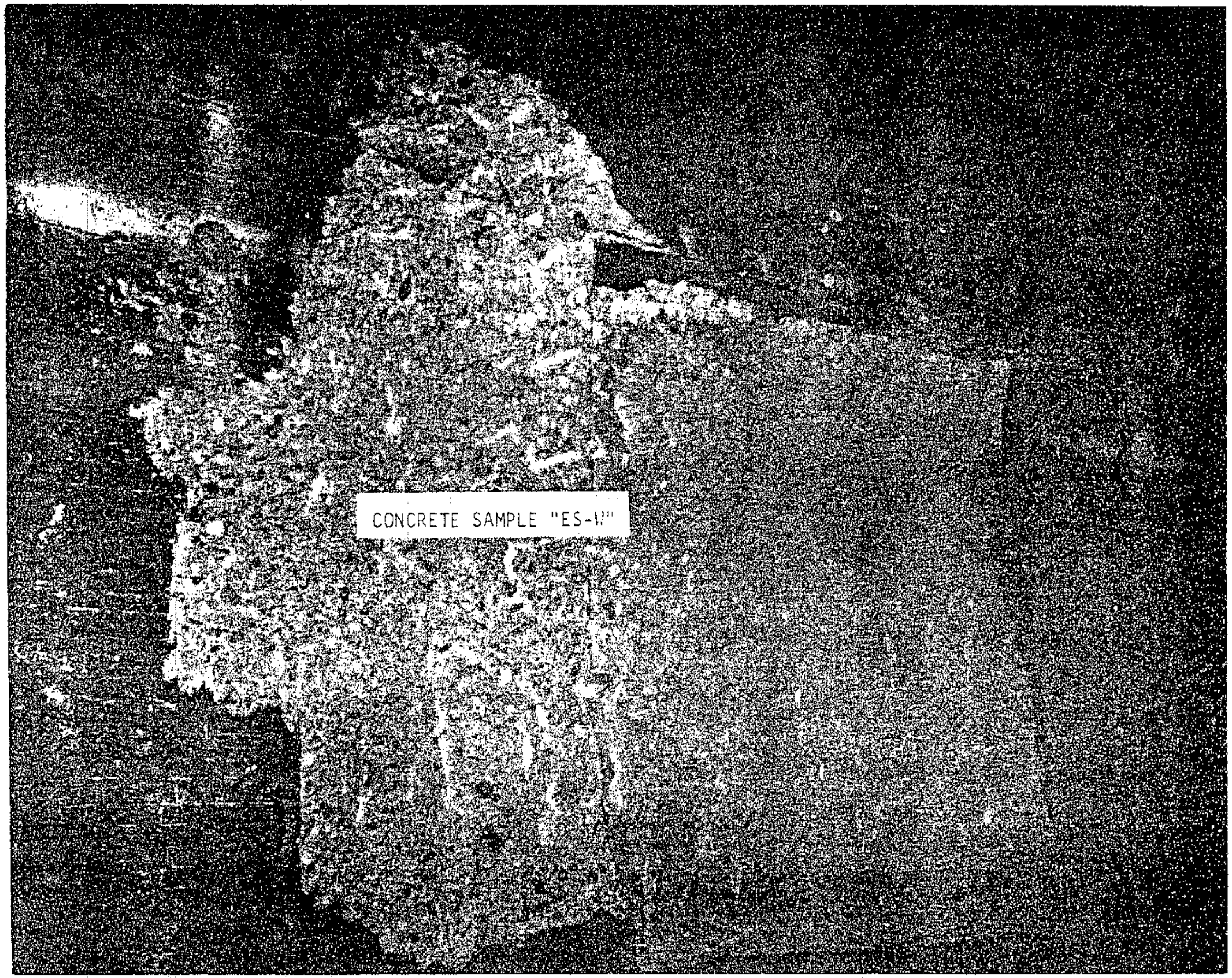
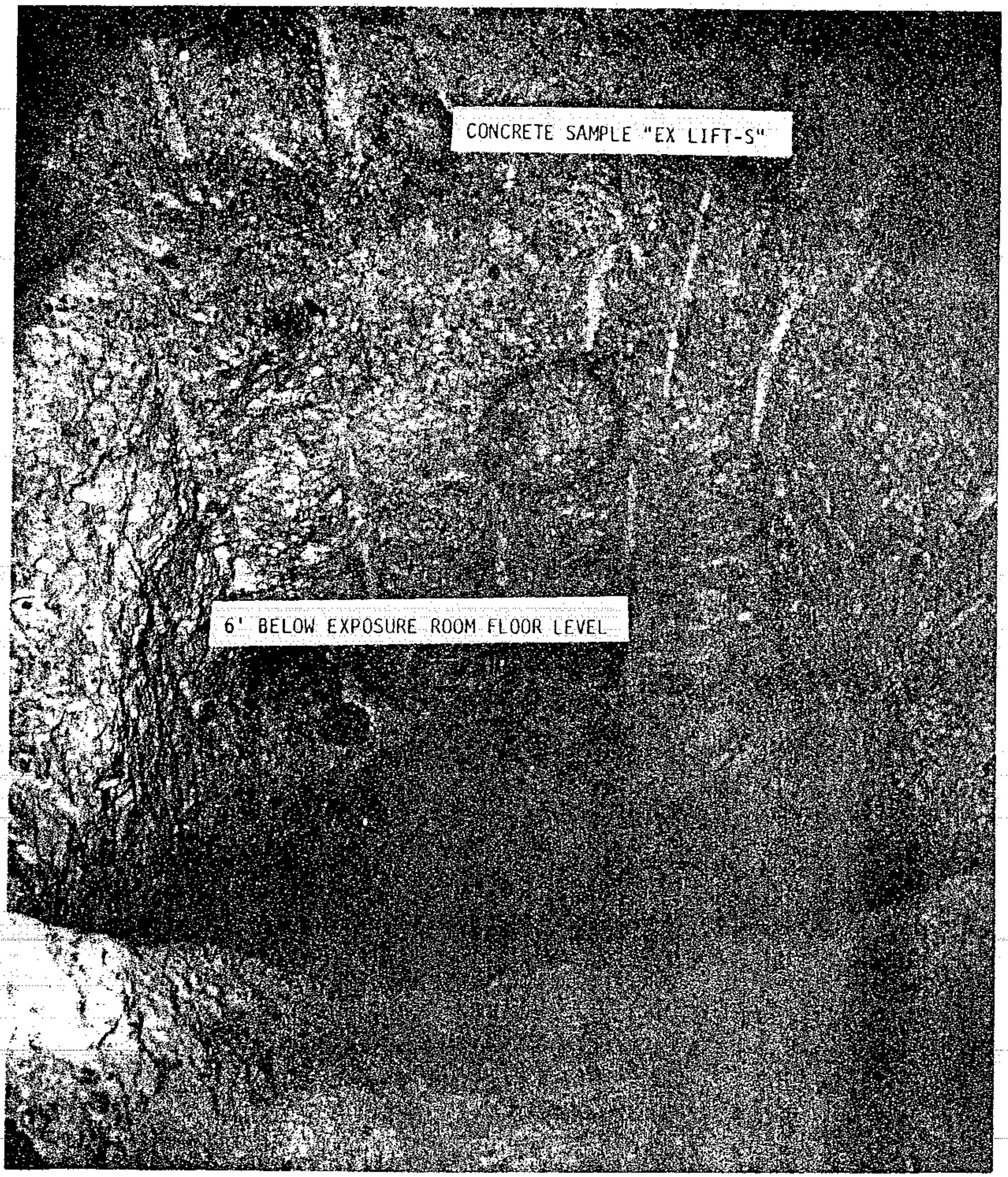
Incl 6



WEST WALL OF EXPOSURE ROOM

Photograph 3

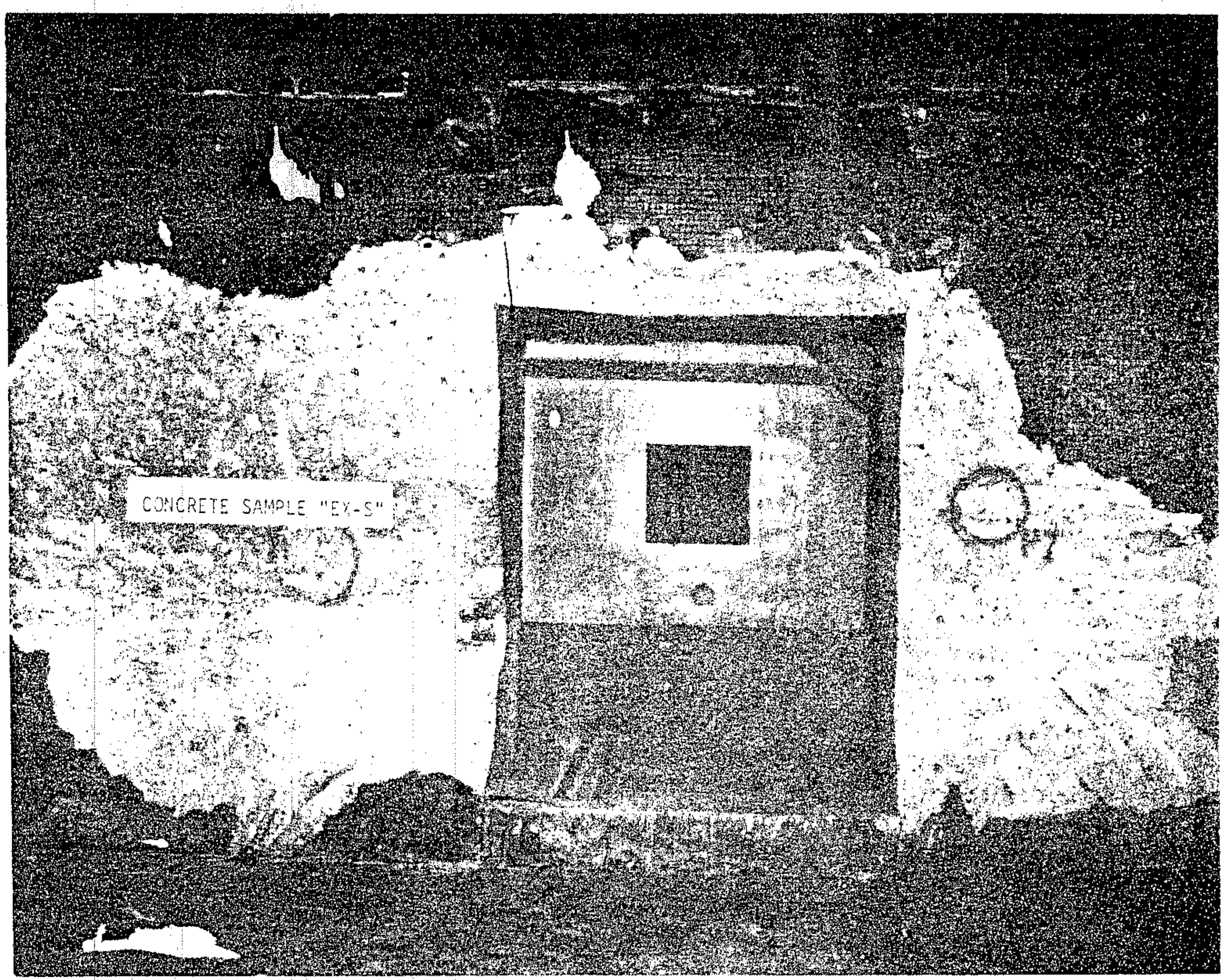
HSE-RH/WP
SUBJECT: Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980



Photograph 4 LIFT WELL---EAST END OF EXPOSURE ROOM

Incl 6

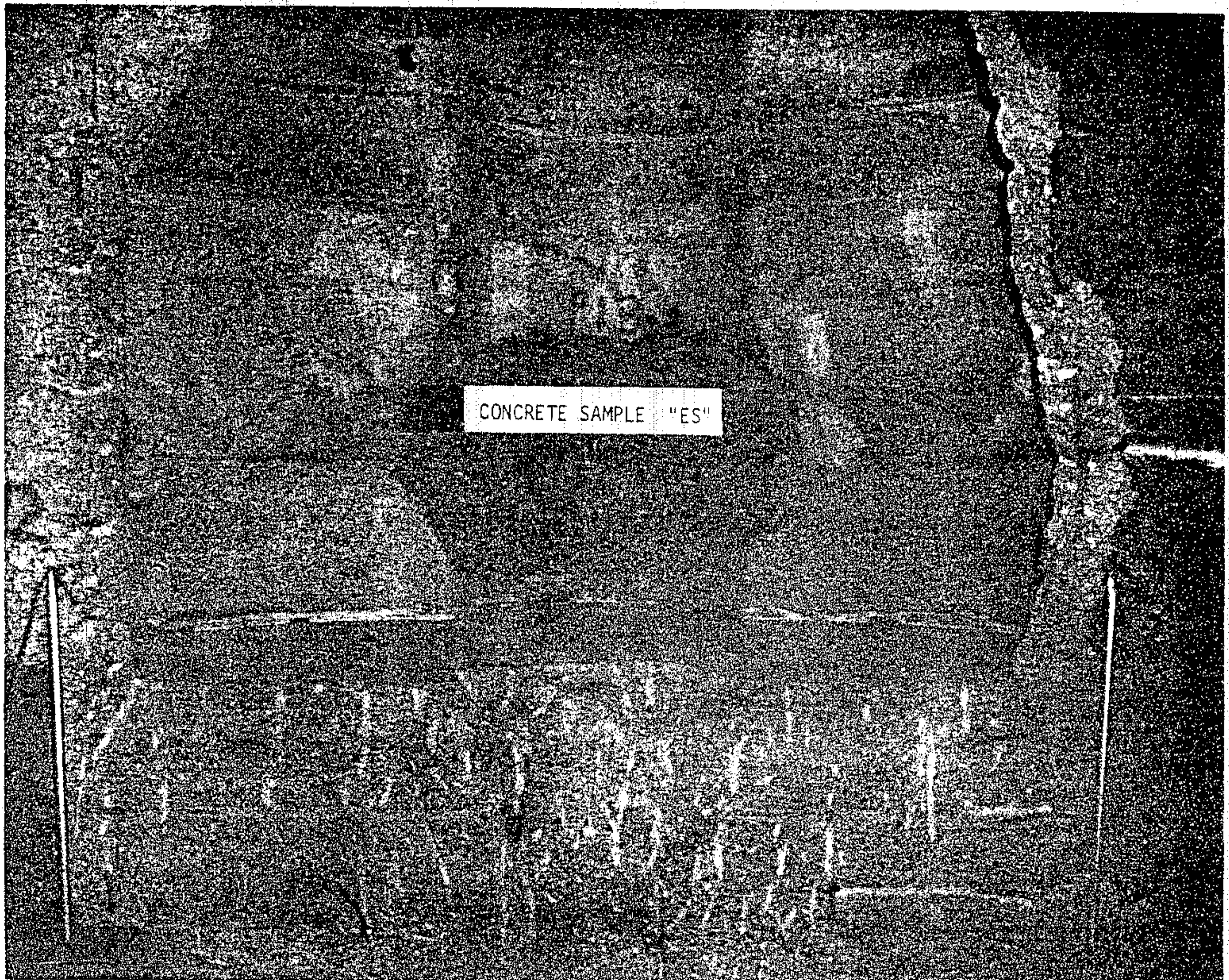
Encl 6



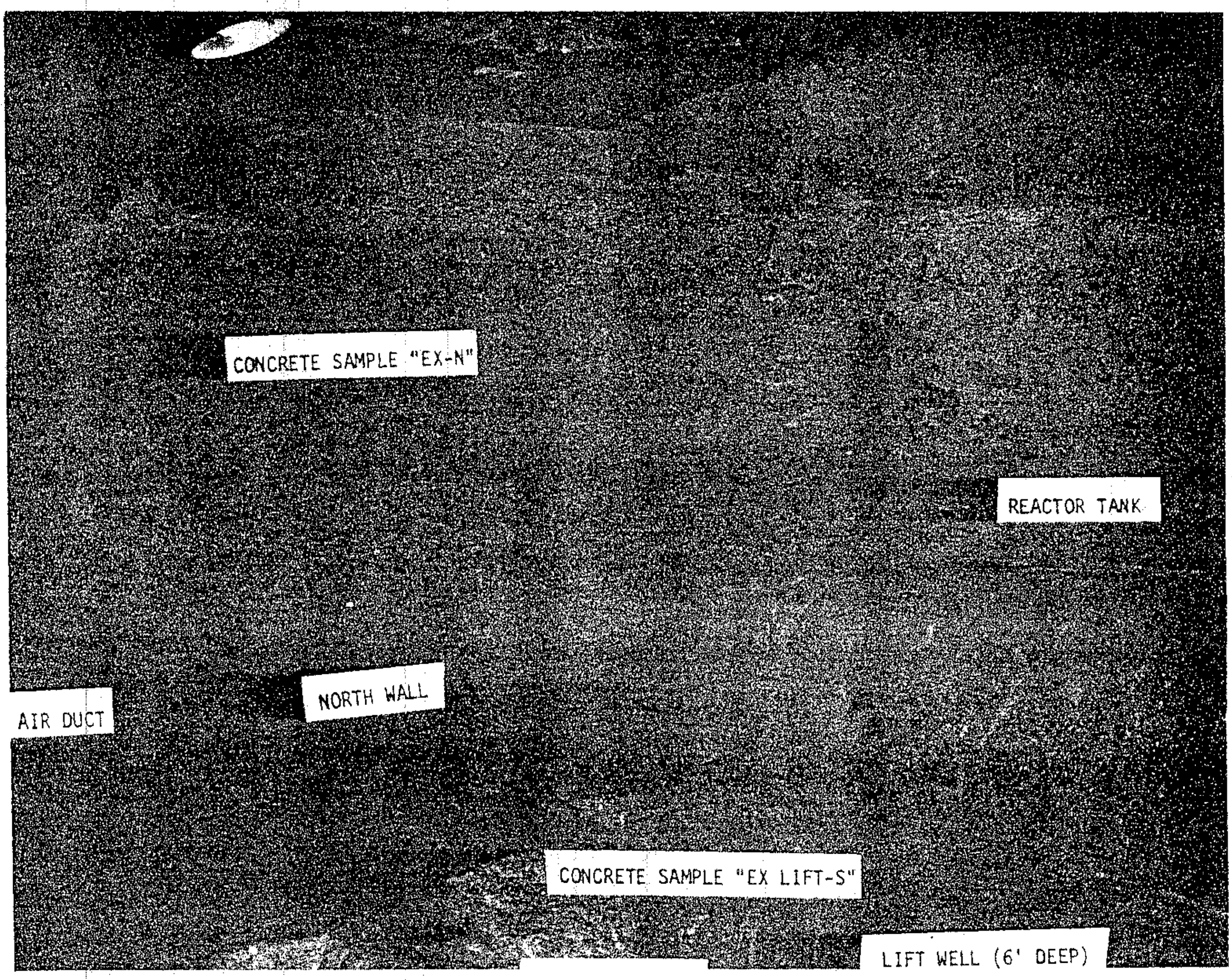
PORTAL VIEWED FROM EXPOSURE ROOM

Photograph 5

Incl 6



REACTOR TANK VIEWED FROM EXPOSURE ROOM Photograph 6



EXPOSURE ROOM

Photograph 7

Atomics International Division
Rockwell International

SUPPORTING DOCUMENT

NUMBER: N001-FDP-960-001

REV LTR/CHG NO.: A (SEE SUMMARY OF CHG)

PROGRAM TITLE: Decontamination and Disposition of DORF

DOCUMENT TYPE: Facilities Dismantling Plan

KEY NOUNS: Dismantling Plan, DORF

DOCUMENT TITLE: Facilities Dismantling Plan for DORF, Diamond Ordnance Radiation Facility

ORIGINAL ISSUE DATE:

GO NO.	S/A NO.	PAGE 1 OF TOTAL PAGES 51
04764	20100	REL. DATE 12-18-79

PREPARED BY/DATE: J. M. Harris 10/23/79 DEPT: D/731 MAIL ADDR: T034

IR&D PROGRAM? YES ☐ IF YES, ENTER TPA NO.

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R. J. Tuttle	22 Oct 79
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ABSTRACT

The Department of the Army has contracted with Energy Systems Group for the Decontamination and Dismantling of the Diamond Ordnance Radiation Facility (DORF). The reactor is a Triga type with pool, exposure room and support facilities. This plan is prepared to note the techniques and sequences of events to meet the contract requirements of DAAK-21-79-C-0136. The Diamond Ordnance Radiation Facility will be decontaminated for release for unrestricted use per NRC Regulatory Guide 1.86.

RESERVED FOR PROPRIETARY/LEGAL NOTICES

FORM 734-C REV. 2-74

REV	SUMMARY OF CHANGE	APPROVALS AND DATE
A	Page 11 - B.1. The sentence "a radiation survey of the reactor components that are scheduled for shipment to HEDL will be conducted," has been moved from its position as seventh sentence in the paragraph to its new position of third sentence in the paragraph. Page 17 - B.5. Last paragraph, "U. S. Army Environmental Health Agency" changed to "U. S. Army Environmental Hygiene Agency." Page 31 - Sentence under Total Contamination changed from, "0.1 mrad/hr beta-gamma measured through 7 mg/cm^2 absorber at 1 cm," to "0.1 mrad/hr average or 0.3 mrad/hr maximum beta-gamma measured through 7 mg/cm^2 absorber at 1 cm." Page 35 - J.1. Third sentence, "may be required," changed to, "shall be required." Page 37 - L.1. Last sentence, first paragraph. To was inserted between assigned and the, to read "...assigned to the radiologically..." Page 39 - M. Sentence added in 4th paragraph "Surgical gloves shall be protected with an overglove." Page 47 - T.3 Last sentence "...or as otherwise specified by HSRS." changed to, "or as otherwise specified." Page 50 - W.3 29 CFR 17, Part 1910.93 a changed to 29 CFR, Part 1910.1001. Page 50 - W.5 29 CFR 17, Part 1910.95 changed to 29 CFR, Part 1910.95.	W. D. Kittinger 12-13-79 RJ Tuttle 12/14/79 W.R. McClure 12/17/79 Rel. Date 12-18-79 CV

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FIGURES

TABLES

I. OBJECTIVE

The objective for dismantlement and radioactive decontamination of the Diamond Ordnance Radiation Facility (DORF) is to place it in a condition acceptable for release for unrestricted use. Reactor components will be packaged and shipped to the Department of Energy (DOE) at Hanford, Washington. All radioactive materials and components will be removed and decontaminated for release for unrestricted use, or packaged for disposal as radioactive waste and delivered to a licensed burial site. Areas of the facility and materals released for unrestricted use will be decontaminated to levels which are as low as reasonably achievable (ALARA), but in all cases to levels below those described in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86, Table 1. The methodology in decontamination for release for unrestricted use as stated in this guide will be followed. As part of the ALARA program, Rockwell has established the limits shown in Table 2 as their target for compliance with this contract. The limits are based on experience regarding levels that in most cases are reasonable achievable and can be effectively monitored.

TABLE 1

NRC REGULATORY GUIDE 1.86
ACCEPTABLE SURFACE CONTAMINATION LEVELS

Nuclide[a]	Average[b c]	Maximum[b d]	Removable[b c]
U-nat, U-235, U-238 and associated decay products	5000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	100 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm $\beta\gamma$/100 cm^2	15,000 $\beta\gamma$/100 cm^2	1000 dpm $\beta\gamma$/100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objectives of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

TABLE 2

ROCKWELL INTERNATIONAL/ENERGY SYSTEMS GROUP
CONTAMINATION LIMITS FOR DECONTAMINATION & DISPOSAL OF DORF

	TOTAL	REMOVABLE
Beta-Gamma Emitters	0.1 mrad/hr average [a] and 0.3 mrad/hr maximum [b] at 1 cm with 7 mg/cm^2 absorber	100 dpm/100 cm^2
Alpha Emitters	100 dpm/100 cm^2	20 dpm/100 cm^2

[a]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[b]The maximum contamination level applies to an area of not more than 100 cm^2.

II. SCOPE OF PLAN

The scope of this Dismantling Plan is to delineate the activities necessary to realize the objectives stated in Section I. The activities are categorized as: planning, monitoring, and control; radiological survey; dismantlement and disposal; and documentation.

III. PLANNING, MONITORING, AND CONTROL

The activities which comprise the dismantlement of DORF will be initiated, monitored and controlled by the Rockwell Site Manager at DORF. The site manager will also have the overall technical responsibility for the dismantling activities and will be the onsite interface for all contacts with the Army's Site Contracting Officer or his representative. The DORF D&D organization structure is shown in Figure 1. The Rockwell Radiation and Nuclear Safety representative will be responsible for radiological surveys and survey data analyses. Records of significant radiation surveys and analyses will be made available to the Contracting Officer or his representative.

A schedule listing the specific tasks and the proposed sequence for performance is presented in Table 3. The estimated level of manpower and milestones for these activities are included for information. The milestone schedule will serve as the criteria to measure progress in dismantling DORF.

The Operational Safety Plan on Decontamination and Disposal at DORF is attached as Appendix A. It contains the radiation safety, industrial hygiene, and industrial safety procedures in support of the activities described in this Dismantling Plan.

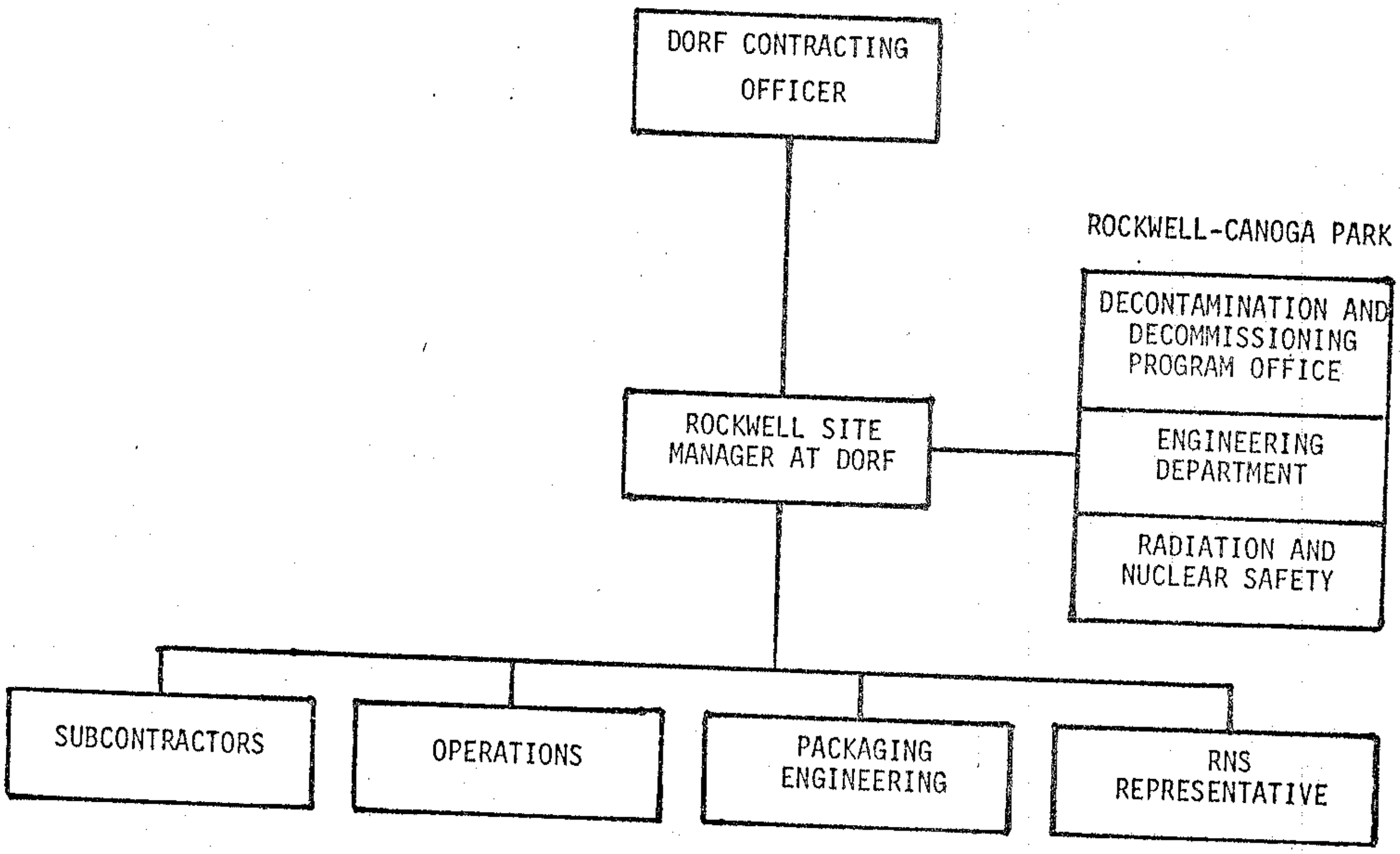
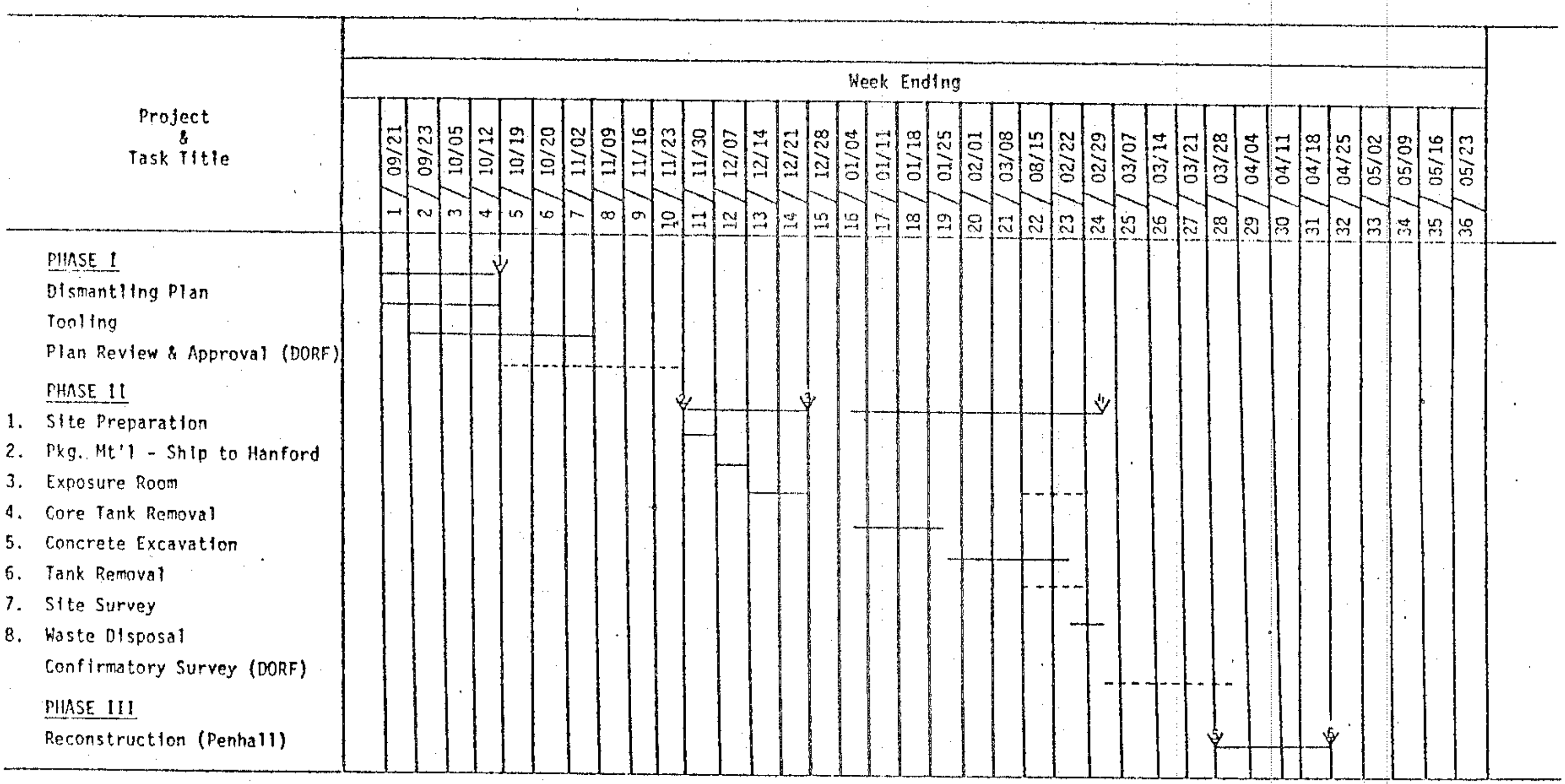


FIGURE 1. DORF D&D ORGANIZATION CHART

TABLE 3

DORF D&D PROJECT SCHEDULE



Milestone 1 - Complete Phase I, 10/14/79
Milestone 2 - Start Phase II, 11/26/79
Milestone 3 - Reactor Components Shipped to HEDL 12/21/79
Milestone 4 - Complete Phase II, 2/26/80
Milestone 5 - Start Phase III Within 30 Days of Survey Acceptance
Milestone 6 - Complete Phase III

IV. RADIOLOGICAL SURVEY

A radiological survey will be made to assess the extent of radioactivity present in the facility. This assessment will include the grounds which surround the facility to establish and record conditions at the site before beginning the dismantling activities.

Radiological surveys will be conducted before and during the Phase II work only to provide information for guidance in determining (1) what areas are radioactive, (2) when sufficient material has been removed to release these areas for unrestricted use, and (3) personnel surveillance.

The comprehensive radio-isotopic analysis appended to the Request for Quotation (RFQ) as Appendix IV was an estimate of the radioactive material remaining in the DORF structure as of the spring of 1978. This will serve as a guide and will be used as a qualitative indication of the presence of radioactivity.

V. DISMANTLEMENT AND DISPOSAL

The scope of work required to dismantle DORF is presented, followed by a description of the principal tasks required to accomplish the Phase II and III work defined in the RFQ. The tasks will be performed in the order shown in Table 3 if practicable. Overlap of the schedule tasks will occur as required to maintain continuity in the overall program.

A. DISMANTLEMENT SCOPE OF WORK

Activities required to accomplish the dismantlement of DORF include: (1) surveying and recording the radiological condition of the facility and surrounding grounds to define the existing condition; (2) the analysis and disposal of the core tank water per 10 CFR 20 limits; (3) the removing, packaging and shipping of the reactor components listed in Table 4 of this plan; (4) removing, packaging, and shipping to a licensed burial site the radioactive materials and components referenced in the RFQ and those generated during the dismantlement of DORF; (5) removing and disposing of the nonradioactive components or materials listed in Paragraph F.4.1 (a through g) of the RFQ (Note that (h) is included in Activity 3 above); (6) removing and delivering the jib-crane to the AURORA facility; and (7) the Health Physics support necessary to assure compliance with NRC Regulatory Guide 1.86 and 10 CFR 20.

B. PHASE II

1. Site Preparation

The site preparation task includes those activities required to move the ESG staff and their equipment to the site and to establish a base of operations. A radiation survey of the nonradioactive portions of the site will be conducted for documentation. A radiation survey of the reactor components that are scheduled for shipment to HEDL will be conducted. An analysis of the pool water to determine compliance with

TABLE 4

REACTOR COMPONENTS FOR SHIPMENT TO
DOE-HEDL, RICHLAND, WASHINGTON

Item No.	Description	Unit	Quantity
1	Core Support Structure, Upper Section	Each	1
2	Core Support Structure, Lower Section	Each	1
3	Top and Bottom Grid Plates	Each	1
4	Connecting Rods for Control Rods	Set	1
5	Control Rods	Set	1
6	Carriage Drive Motor	Each	1
7	Water Pump: 1.5 hp	Each	1
8	Incore Experiment Tube	Each	1
9	Ion Chamber Supports and Ion Chambers	Set	3
10	Carriage Support Rails	Set	1
11	Lead Shield Door Drives and Linkage	Set	1
12	Pool Cover Plates	Set	1
13	Fuel Storage Racks, Underwater	Each	8
14	Fuel Measurement Tool with Dial Micrometer	Each	1
15	Aluminum Water System Piping	Each	1
16	Water Pumps	Each	3
17	Demineralizers, 3 ft^3	Each	4
18	Flowmeters, 25 gpm	Each	2
19	Neutron Source, 10 curies, am-be	Each	1
20	Neutron Source Holder	Each	1
21	Pool Lights	Set	1
22	Carriage Positioning Potentiometer	Each	1
23	Carriage Umbilical Arm	Each	1
24	Fuel Element Location Diagram	Each	1
25	Water Box, 1 ft^3 Capacity	Each	1
26	Charcoal Filter, 1 ft^3 Capacity	Each	1

10 CFR 20 will be performed. The pool water will be discharged to the sanitary sewer as analysis permits. The appropriate limits are those listed in 10 CFR 20, Appendix B, Table 1, Column 2, as provided by 10 CFR 20.303. Should the water analysis show contamination above limits, the existing purification system will be used for cleanup.

2. Packaging and Shipping Reactor Components to HEDL

The electrical service for the reactor auxiliary systems will be disconnected from the relay and power distribution panels and the wiring will be removed. This will include power disconnects to the lead shield doors, carriage drive, and the diffuser pump.

All of the items listed in Table 4 will be removed, packed into weatherproof containers, and staged for transportation to the DOE, Hanford Engineering Development Laboratory (HEDL), Richland, Washington. The americium-beryllium neutron source will be placed in an approved Type A shipping container for shipment with the above items.

The water treatment system in the Filter Room will be removed after the pool water has been discharged to the sanitary sewer and a determination has been made that it will no longer be required. Water will be drained from the piping and filter medium and the water will be dispositioned based on radiation survey analysis. All of the items listed will be shipped to HEDL when they have been packaged and are available for shipment. Components will only be disassembled to the degree necessary to permit packing into reasonably sized containers.

All packaging will conform to the Department of Transportation (DOT) Specification Title 49 Code of Federal Regulations. Each package shall be monitored by the Health Physicist to determine its radioactive content and will be weighed to establish its shipping weight.

When all radioactive components have been removed from the facility, the areas which housed those components will be radiologically surveyed and the survey documented. Those areas which are above the release limits will be identified on a facility plot plan and scheduled for removal during the appropriate demolition task.

The jib crane will be removed from DORF and transported to the AURORA facility when it is no longer required to support dismantling activities.

3. Exposure Room

The exposure room will be stripped of its wood lining, lead shields, lead shield hoist, and other removable components. The material will be separated and dispositioned either to salvage or packaged for radioactive disposal.

Before starting activities in the exposure room, the floor drains will be plugged to reduce the potential for transporting radioactive materials into the sanitary sewer system.

The wood timber lining will be removed from the room using conventional techniques. Each timber will be surveyed to determine radioactivity and will be dispositioned according to Table 2 criteria. Material that is activated to levels that exceed Table 2 limits will be packed in strong, tight shipping containers, while material that is not activated will be set aside for salvage. The lead shields will be removed, surveyed and set aside for disposition. The lead shield hoist will be removed in its entirety and packed in a strong, tight shipping container. All other removable components will be removed from the room and will be dispositioned accordingly. When all removable material has been dispositioned, the exposure room will be vacuumed to remove remaining residue from the surfaces.

A detailed radiation survey of the exposure room will be conducted to establish a mapping of activity in the concrete. Selected areas will be sampled by core drilling to establish the extent of activation. The exposure room door and doorway will be included in the survey analyses. An excavation plan will be developed, for implementation during concrete excavation detailed in Section 5.

4. Core Tank Removal

All extraneous structures will be removed from the core tank, the lead shield doors will be drained of lead, the lead and doors will be removed from the core tank, the core tank will be stripped from the concrete, and the activated tar paper lining will be removed from the concrete surfaces. The materials will be surveyed when they are removed from the area and will be dispositioned accordingly.

The procedure to remove the lead shield doors will consist of drilling holes through the lower wall of each door to drain enough lead to permit them to be lifted with the 3.5-ton overhead crane. A dynamometer will be used to provide assurance that the weight of the load is within the crane limit. Each door will be lifted from the core tank and transferred to an area where the remaining lead can be removed. The doors and lead will be surveyed and dispositioned accordingly.

The procedure to remove the core tank will be to section the tank (by saw cutting) into vertical strips. Each strip will be pulled from the concrete by conventional techniques as determined by experience gained during the first and subsequent removal attempts. Because of the uncertainty associated with the adhesion of the tank to the concrete by virtue of tar paper, trial and error will be required. Leverage tools such as pry-bars, wedges, block and tackle, etc., will be used initially. If these techniques prove unsuccessful, then hydraulic or pneumatic techniques will be applied.

The activated tar paper lining will be removed from the concrete by scraping and/or by chipping away portions of the concrete. Where the concrete is also activated, the tar paper will be left for removal with the concrete.

All activated and contaminated materials will be packaged for disposal as detailed in the Waste Disposal section. Material will be size reduced, where practical, to reduce the volume of radioactive waste.

A detailed radiation survey will be conducted of the exposed concrete structure to establish a map of radioactivity. Selected areas will be sampled by core drilling to establish the extent of the activation. An excavation plan will be developed for the concrete structures. Implementation of the plan will be described in Section 5, "Concrete Excavation."

5. Concrete Excavation

Concrete will be removed from the pool cavity, exposure room, and exposure room door to the extent required to permit release of these structures for unrestricted use. Guidance for the amount of concrete to be removed will be determined by radiation survey and by the excavation plans developed after core tank removal and exposure room cleaning. When the detectable levels of radioactivity in the concrete are below the levels shown in Table 2, they will be considered to be in compliance with NRC Guide 1.86 (Table 1).

The extent of removal will be governed by the extent to which the structures are activated. Where activation is shallow, scabbling or chipping with pneumatic hammers will be used to break the concrete at the surface. High volume vacuum cleaners equipped with HEPA filtration will be used to remove the concrete and to control airborne contamination. Where penetration is several inches deep, jack hammers will be used. This operation will be aided by depth cutting with a concrete saw if necessary. If depth of activation is such that these techniques are not applicable, a hydraulic ram hoe or other devices will be used to break the concrete for removal.

Dust and particulate generation will be monitored by the Radiation and Nuclear Safety representative and control will be accomplished by use of a vacuum cleaner or water mist depending on the operation in progress. High volume air sampling will be conducted within the work area during operations which might produce airborne contamination. Personnel will be required to wear respirators whenever sampling indicates unacceptable levels of airborne contamination. Temporary structures will be built around the work area if necessary to control the spread of contamination.

Radiation survey data generated during the activated concrete excavation will be analyzed to provide a basis for compliance with Regulatory Guide 1.86 and ALARA (Tables 1 and 2). When the data indicates that compliance with these criteria have been met, concurrence will be solicited from the U.S. Army Environmental Hygiene Agency.

6. Site Survey

A final radiation survey will be conducted to verify the site condition. Surface smears and material samples will be selected by the Rockwell Site Manager with assistance from the Radiation and Nuclear Safety representative and the DORF Contracting Officer. These specimens will be sent to an independent laboratory for analysis. The specimens will be taken from representative areas of the buildings and excavations to confirm compliance with ALARA and Regulatory Guide 1.86. These data will provide independent analyses of the site condition and will form a basis for demonstrating that the facility can be released for unrestricted use.

7. Waste Disposal

Radioactive waste will be packaged when it is generated and will be staged in full load lots (~40-45,000 lb) for shipment. Shipments will be made under the exclusive use provision of Title 49 Code of Federal Regulations which permits low specific activity (LSA) waste to be packaged for shipment in strong, tight containers. Shipments will be monitored by the Radiation and Nuclear Safety Representative for conformance to DOT regulations. Radioactive waste will be delivered to a common carrier for delivery to a licensed disposal site as full load lots become available, or at the completion of Phase II.

Noncontaminated components and waste materials listed in the RFQ for disposal (F.4.1) will either be retained and used as backfill in the pool cavity or hauled to a local licensed landfill during Phase III.

C. PHASE III

A concrete wall will be erected between the exposure room and the pool cavity to provide a barrier between the two areas so the pool cavity can be used to hold backfilled concrete debris. The steel ramp structure and concrete parapet surrounding the pool cavity will be dismantled. The steel structures will be removed for salvage and the concrete support wall will be broken up and the debris will be placed into the pool cavity. The debris shall not be filled to a level above the main floor.

The air conditioning system inlet and exhaust ducts to the exposure room will be restored if necessary and made operable. Where practical, all electric outlets, air, water, and sewer lines will be retained in working order during the dismantling activities.

At the completion of Phase III, the accumulated demolition debris not deposited into the pool cavity will be removed from the facility area. The contractor's equipment will be removed and a reasonable effort will be made to clean the interior of the facility building of accumulated dirt and dust resulting from the demolition efforts.

VI. DOCUMENTATION

Documentation of the Dismantlement of DORF will consist of informal weekly progress reports, radiation survey reports, and a final report.

The informal weekly progress reports are primarily written to the Rockwell home office in Canoga Park, California, to keep them informed of the DORF site operations and to alert them to any changes that may impact schedule or cost. Copies of these reports will be sent to the Contracting Officer or his representative.

Radiation survey reports of significant data will be entered on a Rockwell Form 732-A, Health & Safety Analysis Report for distribution within the Rockwell organization. Copies of these reports will be sent to the DORF Contracting Officer or his representative.

At the completion of Phase II, a final report will be written to document the dismantlement of DORF. The report will describe the activities required to accomplish the work, problems encountered, solutions to the problems, and the current status of the facility and structures. The report will also include data, specified in units identical to those shown in Table 1, to show the effort made to reduce residual contamination to levels that are as low as reasonably achievable. It will describe the scope of the radiation survey, the general procedures followed to obtain the data, and any other pertinent information about the radiation survey data. A summary of the radioactive waste disposal information will be included to show the quantities of material removed from DORF.

The final report will be appended at the completion of Phase III to document the final status of the facility at the termination of this contract.

APPENDIX A
OPERATIONAL SAFETY PLAN
FOR DECONTAMINATION AND DISPOSAL OF DORF

I. PURPOSE

To delineate the radiation safety, industrial hygiene, and industrial safety procedures for the decontamination and disposition (D&D) of the Diamond Ordnance Radiation Facilities (DORF).

II. SCOPE

This plan applies to all operations at DORF involving the deactivation, dismantling, decontamination, and disposal of that nuclear facility.

The plan meets or exceeds the requirements set forth in Rockwell/ESG Standard Operating Policies, in applicable regulations and standards, in DOEM 0524, in the Williams-Steiger Occupational Safety and Health Act of 1970 (OSHA), 10 CFR 19 and 20, and in Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86.

III. RESPONSIBILITIES

A. HEALTH SAFETY AND RADIATION SERVICES

1. Radiation and Nuclear Safety

Radiation and Nuclear Safety shall establish requirements for design and operational procedures; and approve disposition of source and special nuclear material, and byproduct radioactive material.

Radiation and Nuclear Safety will designate and identify radiologically posted areas, radiological safeguards requirements, and radioactive materials; and will control the use and disposition of radioactive materials, and implement radiological safety standards.

Radiation and Nuclear Safety will perform field measurements of radiation and radioactive contamination levels, evaluate internal and external personnel radiation exposures, and evaluate radioactive material in the workplace environment.

Radiation and Nuclear Safety will maintain records as necessary to demonstrate compliance with ESG standards and applicable regulations. Included in these records will be a chronological log of information dealing with daily operations, conditions, and occurrences relating to radiological safety.

Radiation and Nuclear Safety will advise the Rockwell Site Manager and operations personnel on the safe performance of their assigned tasks.

Radiation and Nuclear Safety will evaluate operational conditions to determine requirements for personnel monitoring and protective devices such as film badges, breathing zone air samplers, bioassay, protective clothing, and respiratory protection devices.

The Radiation and Nuclear Safety (RNS) representative assigned to the project will conduct the radiological surveillance program and will maintain sufficient familiarity with program operations and facility conditions to be aware of those areas which may require increased surveillance or corrective action.

2. Industrial Hygiene and Safety

Radiation and Nuclear Safety will provide the services necessary to control personnel exposures to toxic chemicals and harmful physical agents and to control mechanical and electrical hazards. RNS Representative will maintain surveillance of the occupational environment to identify, evaluate, and control conditions pertinent to health and safety, and to assure compliance with the requirements of DOE Manual, OSHA, 10 CFR 19 and 20, as appropriate.

B. ROCKWELL SITE MANAGEMENT AT DORF

The Rockwell Site Manager is responsible for the safety of all personnel within facilities under the jurisdiction of the DORF D&D Program.

The Site Manager will ensure that all personnel employed at or visiting the facility know and understand the rules and regulations governing work with radioactive materials and will assure compliance with these rules. The Site Manager will carry out the responsibilities charged to "Operating Supervision" and will provide safe conditions at the facility, in conformance with applicable regulations and standards, and under the guidance of Radiation and Nuclear Safety.

The Site Manager will establish the requirements for the packaging of radioactive waste, collecting of packaged waste, and will arrange for disposal by land burial.

Rockwell Program Management and the Site Manager will coordinate radiological and industrial hygiene and safety problems with Radiation and Nuclear Safety as appropriate.

C. OPERATIONS PERSONNEL

Operations personnel are responsible for compliance with all rules governing work with radioactive and hazardous materials as outlined by this procedure and as established by Radiation and Nuclear Safety and D&D Program Management. Operations personnel are responsible for taking every reasonable precaution to minimize radiation exposures to themselves and to fellow workers and to prevent the unnecessary release of radioactive material.

D. CONTRACTOR PERSONNEL

Contractor personnel are responsible for compliance with all safety rules and requirements established by Radiation and Nuclear Safety and for responding to specific instructions from the RNS Representative with regard to radiation safety and industrial hygiene.

IV. ADMINISTRATIVE SAFEGUARDS

A. PROCEDURAL CONTROL

Any changes to the radiation safety or industrial hygiene and safety procedures must be jointly authorized by Radiation and Nuclear Safety and the Site Manager following evaluation of the proposed changes by the RNS Representative. Revised procedures will be distributed to all personnel directly affected by the change.

Operations involving potential radiological hazards or potential industrial safety hazards will be reviewed in advance by Radiation and Nuclear Safety.

B. METHODS OF REPORTING DAY-TO-DAY CONDITIONS

Day-to-day operational safety conditions will be observed by the assigned Radiation and Nuclear Safety representative, who will report all recognized hazardous conditions and each instance of noncompliance with regulatory directives to the Site Manager and the workers involved. Radiological data (film badge and bioassay results, radiation and contamination survey results, air sampling reports, etc.) will be maintained by the RNS Representative. Whenever these data indicate the need for corrective action, the RNS representatives will contact the Site Manager to arrange for such action. Industrial hygiene and safety conditions observed by the RNS representative will also be communicated to the Site Manager. A summary of incidents and data will be reported to the Site Manager, the Radiation and Nuclear Safety Office, and the Rockwell Program Management Office on a weekly basis.

V. GENERAL RADIATION AND INDUSTRIAL HYGIENE AND SAFETY PROCEDURES

Certain radiological and industrial safety controls and procedures are independent of operations in the facilities, and are required to provide facility surveillance and radiological and industrial safety protection commensurate with the ESG contract and regulatory agency standards.

A. AREA DESIGNATION, RADIOLOGICAL SAFETY CONTROL

All areas are designated as either radiologically posted or unposted. A posted radiological area is an area, defined by physical barriers, which is posted with prescribed caution signs or labels for purposes of radiation protection. Signs used to designate posted radiological areas must comply with applicable regulations. There are six posted area classifications as defined below:

1. Radiation Area

A Radiation Area is an area subject to radiation from encapsulated radioactive materials and/or radiation machines within the area, or to radiation from any source outside the area; where there exists radiation at such levels that an individual could receive in any one hour a dose to the whole body in excess of 5 millirem, or in any five consecutive days a dose in excess of 100 millirem.

Each Radiation Area will be posted with a sign meeting all regulatory requirements including the radiation symbol and the words "CAUTION - RADIATION AREA." Where appropriate, indications of the radiation level will be included in the area posting.

2. Radiation Area - Radioactive Contamination

A Radiation Area - Radioactive Contamination is an area in which work with and/or storage of unencapsulated material is permitted with the provision that the radioactive material concentration in air is not likely to exceed 25% of the appropriate occupational exposure limit. Each Radiation Area - Radioactive Contamination will be posted with signs meeting all applicable regulatory requirements including the radiation symbol and the words "CAUTION - RADIATION AREA - RADIOACTIVE CONTAMINATION."

3. Radiation Area - Airborne Radioactivity

A Radiation Area - Airborne Radioactivity is an area in which the radioactive material concentration in air is likely to exceed 25% of the applicable regulatory standard for occupational exposure.

Each Radiation Area - Airborne Radioactivity will be posted with signs meeting all applicable regulatory standards including the radiation symbol and the words "CAUTION - RADIATION AREA - AIRBORNE RADIOACTIVITY."

4. High Radiation Area

A High Radiation Area is an area accessible to individuals in which there exists radiation at such levels that an individual could receive in any one hour a dose to the whole body in excess of 100 millirem. Each High Radiation Area will be posted with signs meeting all applicable regulatory requirements including the radiation symbol and the words "CAUTION - HIGH RADIATION AREA."

5. Radiation Area - Radioactive Materials

A Radiation Area - Radioactive Materials is an area in which work with and/or storage of encapsulated materials is permitted.

Each Radiation Area - Radioactive Materials will be posted with signs meeting all applicable requirements, including the radiation symbol, and the words "CAUTION - RADIATION AREA - RADIOACTIVE MATERIALS." Federal and State regulations also require that storage containers and localized areas in which radioactive materials are present in certain amounts will be posted with signs containing the radiation symbol and the words "CAUTION - RADIOACTIVE MATERIALS." It should be noted that these containers and areas may or may not be located within posted areas. Radiation and Nuclear Safety will advise operating supervision as to the amounts of radioactive materials in containers or localized areas which require such signs.

6. Restricted Access Area

A Restricted Access Area is an area identified by Radiation and Nuclear Safety as requiring special safety precautions for entry and requiring inspection immediately prior to entry by any person. Each

Restricted Access Area will be posted with signs with the following words in yellow over a red background:

"WARNING - RESTRICTED ACCESS AREA - OBTAIN PERMIT
FROM OPERATIONAL SAFETY PRIOR TO ENTRY"

Any area meeting more than one of the above criteria will be posted with all of the applicable signs.

B. AREA DESIGNATION, INDUSTRIAL SAFETY CONTROL

Operations posing potential hazards shall be identified by appropriate caution or warning signs. The signs shall conform to specifications in 29 CFR 17, Section 1910.145. Examples of posting are:

1. Hard Hat Area

A Hard Hat Area will be established wherever personnel are working at different elevations and there is a potential of being hit by falling objects.

2. Eye Protection Area

An Eye Protection Area will be established where a hazard due to flying objects exists.

3. No Smoking Area

No Smoking Areas will be established where explosives, flammable liquids, or gases may be present.

4. Open Excavations

Open Excavations will be protected by appropriate physical barriers.

5. Obstructions

Obstructions will be made clearly visible by the use of yellow and black striping.

C. RADIOLOGICAL SURVEY FREQUENCY

Routine radiation and contamination surveys will be performed in work areas at a frequency to be determined by the assigned Radiation and Nuclear Safety representative in accordance with established procedures. Additional surveys may be required to determine the effectiveness of contamination control procedures. The requirement for these surveys will be established on the basis of initial experience with those tasks which may pose significant personnel radiation or airborne contamination exposure.

D. RADIOACTIVE CONTAMINATION LIMITS

Evaluation of levels of radioactive contamination will be required in order to determine:

1) The adequacy of the level of decontamination performed on the facilities;
2) The extent of required excavation or other demolition of activated structures; and
3) The disposition of equipment, materials, and scrap.

Facilities and equipment will be evaluated for removable and total (fixed plus removable) contamination by means of wipe surveys and instrument surveys. Activated structures will be evaluated for radioactive concentrations by sampling or surveying with detection instruments.

Removable contamination limits for radiologically posted and unposted areas are described in Table A-1. The upper limit of allowable contamination listed in the table is that level which, if reached, requires immediate cessation of operations, immediate decontamination must be effected and measures taken to prevent recurrence. The action limits specified in Table A-1 are the upper limits of the amount of general area contamination tolerable in posted and unposted areas. General contamination in an area in excess of the action limit requires prompt decontamination.

TABLE A-1
REMOVABLE CONTAMINATION LIMITS
(dpm/100 cm^2)

Area	Activity	Upper Limit	Action Limit
Unposted Areas and Radiation Areas	Beta	1,000	100
	Alpha	200	20
Contamination Areas Airborne Radioactivity Areas	Beta	50,000	5,000
	Alpha	20,000	200
Restricted Access Areas	Beta	Not Defined	Not Defined
	Alpha	Not Defined	Not Defined

The levels of contamination which will be considered acceptable for unconditional release of equipment or facilities are as follows:

Removable Contamination

20 dpm/100 cm^2 alpha

100 dpm/100 cm^2 beta-gamma

Total Contamination

100 dpm/100 cm^2 alpha

0.1 mrad/hr average or 0.3 mrad/hr maximum beta-gamma measured through 7 mg/cm^2 absorber at 1 cm

Water

3×10^{-7} μCi/ml beta-gamma

3×10^{-8} μCi/ml alpha

Soil (If Subject to Contamination)

100 ρCi/g gross detectable beta-gamma

10 ρCi/g alpha

Where practicable, items may be decontaminated to levels lower than the acceptable limits.

During demolition activities, all scrap generated will be evaluated for radioactive contamination prior to release to normal waste channels or packaging for disposal by land burial.

E. SURVEY REPORTS

The original copy of radiation, contamination survey, and special radioanalysis reports will be forwarded promptly to the Site Manager and Radiation and Nuclear Safety supervision. These reports will indicate contamination and radiation levels at specific locations throughout the facility. Copies of these survey reports will be retained indefinitely by Radiation and Nuclear Safety.

Radiation and Nuclear Safety will post or have posted such signs as are necessary for the clear identification of potential radiological hazards. To assure that the posting of radiological hazards is current, periodic surveys will be conducted by the Site Manager and the RNS Representative. Signs which have been approved by Radiation and Nuclear Safety will be used to indicate radiological hazards in the facility. No such signs will be removed without the approval of Radiation and Nuclear Safety. In addition, warning signs relative to hazardous conditions and/or special safety requirements may also be posted.

G. FACILITY VENTILATION

The DORF facility ventilation systems will be used to control airborne contamination. If greater control is necessary in localized areas, a system will be constructed.

Direction of air flow from areas of lower contamination to areas of higher contamination will be maintained at all times.

Exhaust from areas in which airborne contamination potential is present will be directed through prefilters and high efficiency particulate air (HEPA) filters.

Filter replacement will be performed when pressure differential across HEPA filters exceed 6 in. of water, or when indicated by reduced air flows. Prefilters will be replaced when pressure differentials across the filters exceed 1 in. of water.

Where practical, a minimum of six air changes per hour will be provided in areas posted as airborne radioactivity areas.

Ventilation systems will provide once-through air with no provision for recirculation.

H. EVALUATION OF AIRBORNE CONTAMINATION

Airborne contamination will be evaluated to assure that no individual is exposed to airborne radioactive or toxic material in excess of regulatory limits.

1. Air Monitoring

Air monitoring for airborne radioactive material will be performed by means of continuous air monitors in such areas as deemed necessary by the RNS Representative.

2. Air Sampling

Air sampling for airborne radioactive or toxic material could be performed by the following methods:

a) Continuously or intermittently by Gast Vacuum Pump air sampling units located at various points throughout a facility. Data from these samples will be evaluated and recorded weekly or daily as indicated by the potential for airborne activity.
b) Special, "hi-volume grab samples" at the discretion of the Radiation and Nuclear Safety representative.
d) Toxic gas detectors, such as the "length of stain" type will be used as indicated by the potential for such exposure.

I. LIMITS FOR AIRBORNE RADIOACTIVITY LEVELS

Every reasonable effort will be made by the use of engineering safeguards to maintain airborne contamination levels at less than 10% of the applicable limits described in DOEM 0524 and 10 CFR 20. In the event airborne contamination levels approach or exceed the applicable limits, the appropriate respiratory protective devices will be utilized to control the exposure.

The applicable limits for airborne contamination levels in the radiologically posted areas are the limits described in Column 1, Table I, Appendix B, 10 CFR 20. These limits will apply to occupational exposure for 40 hours in any seven consecutive days which translates to a time-integrated exposure for seven consecutive days. In the event any employee receives a time-integrated exposure to airborne radioactive materials in excess of 25% of the allowable exposure in seven consecutive days, as indicated by lapel air sampling, appropriate respiratory protection will be required to prevent exposures in excess of the limit. Specific protection factors will be applied to specific types of respirators. Protection factors are applied to airborne concentrations to determine the concentration inhaled by the wearer, according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Airborne Concentration}}{\text{Protection Factor}}$$

Applicable protection factors for air purifying respirators will be 10 (0.1% toxic gas or vapor concentration) for half-face masks and 50 (0.5% toxic gas or vapor concentration) for full-face masks. Only limited use of atmosphere supplying respirators is anticipated. If required, they will be used only by persons specifically qualified and trained in the use of such devices.

At the discretion of the RNS Representative, certain specific operations may require the use of respiratory protective devices strictly on the basis of the potential for exposure to airborne contaminants. Such operations will be identified as work progresses.

In the event that air sampling indicates airborne radioactive material in concentrations greater than the occupational limits, all persons entering the facility will be required to wear lapel air samplers and, if necessary, appropriate respiratory protection devices if the use of such devices is authorized by the RNS Representative.

J. PERSONNEL MONITORING DEVICES

1. Film Badges

Film badges will be worn by all persons entering radiologically posted areas. Film badges will normally be exchanged at the end of each calendar quarter, or in the case of persons with greater exposure potential, at the end of each month. Special film badges and direct reading dosimeters shall be required in addition to the regular personal badge for radiation exposure control during work in High Radiation Areas. The special badges will be processed as required to evaluate cumulative radiation exposure. An exposure report sheet will be provided to supervision listing the reported radiation exposure for each person assigned to the program. Radiation exposure to personnel will be maintained to as-low-as-practicable levels. During any calendar quarter the occupational dose to the whole body of radiation workers shall not exceed 3 rems, as modified by the lifetime occupational exposure limit of 5 (N-18) rems, where "N" equals the individual's age in years at his last birthday. Whenever practicable, dismantling tasks will be planned to utilize remote tooling or shadow shielding to reduce the personnel exposure associated with the performance of the task. Personal film badges will be distributed to job-site personnel by the RNS representative. Visitors film badges will also be located at the job site for issuance by the RNS representative. A signout sheet will be provided for use in the issuance of the visitor badges. All visitors entering a radiologically posted area will complete the signout sheet and obtain a visitor's badge prior to entry. The badge will be returned following the visit, with the exception that visitors anticipating multiple entries may keep the badge for the balance of the calendar quarter.

All film badges used for the DORF program will contain beta-gamma sensitive film packets with the appropriate shields for radiation quality assessment.

2. Dosimeters

Dosimeters may be issued in conjunction with film badges during certain operations at the discretion of the RNS representative to provide an additional control on planned radiation exposures.

3. Extremity Monitoring

Whenever operations are performed which pose a potential for significant extremity exposure, extremity monitoring will be performed. Finger ring film badges or thermoluminescent dosimeters will be utilized for extremity monitoring.

K. AREA RADIATION MONITORING SYSTEMS

1. Area Film Badges

Area film badges will be mounted at selected locations throughout those facilities under the jurisdiction of the D&D program. These film badges will provide a record of integrated radiation levels for the exposure period at these locations. Area badges will be exchanged once each quarter and records of the badge exposures will be maintained by Radiation and Nuclear Safety.

E. BIOASSAY

1. Requirement

Bioassay, principally by means of urinalysis, will be utilized as a means of assessing internal radiation exposure of personnel. A baseline specimen will be obtained from each worker assigned to work in the radiologically posted areas. During the initial period of actual facility decommissioning, specimens may be collected at frequencies of 1 week to 1 month (depending on the nature of the work). Following the initial period, the collection frequency may be reduced, assuming engineering safeguards against airborne radioactivity are demonstrated to be effective. Specimens will then be submitted at least once each calendar quarter, with the exception that specimens will be submitted once each 6 months by persons not routinely assigned to the radiologically posted areas.

Special bioassay specimens, including urine and fecal specimens, will be submitted at the discretion of the RNS representative or Radiation and Nuclear Safety Management whenever there is reason to believe that personnel may have been subjected to internal exposure.

Whenever the analysis of a routine or special bioassay specimen indicates radioactivity present in excess of the minimum detection limit of the analysis, resampling will be performed at a frequency no greater than biweekly.

Invivo lung counting or whole body counting may be used to provide direct evaluation of internal deposition of radioactivity for purposes of confirming urinalysis data, or of providing further evaluation of suspected exposures.

Radiation and Nuclear Safety will notify the Site Manager of the names of employees for whom bioassay specimens are due. The Site Manager will assure that those employees pick up a specimen bottle on the date indicated and collect and return the specimen as directed on the bottle.

2. Analysis

Bioassay specimens will be accumulated by Radiation and Nuclear Safety and shipped to a vendor laboratory for appropriate analysis. Radiation and Nuclear Safety will notify the Site Manager and Rockwell Program Management of any significantly positive results of bioassay analysis. In the event urinalysis indicates excretion rates which are indicative of the presence in an employee of greater than 50% of a maximum permissible body burden, that employee will be restricted from further work in radiologically posted areas until such time as two consecutive urinalyses submitted at least 5 days apart each indicate less than 25% of a maximum permissible body burden.

3. Incidents and Injuries

Any injury, no matter how small, received while working in a radiologically posted area must be reported immediately to the Site Manager or the RNS representative. Medical services will be obtained as required. The RNS representative will conduct wound monitoring, as necessary.

Employees with open cuts, abrasions, etc., will be restricted from work in radiologically posted areas unless specific approval is given by Radiation and Nuclear Safety. All incidents suspected, or known to have caused internal deposition of radioactivity must be reported immediately to the RNS representative.

M. PROTECTIVE CLOTHING AND EQUIPMENT

All persons entering a radiologically posted area in which unencapsulated radioactive material is processed will be required to don protective clothing at the change line located outside the entrance to these areas. The items of protective clothing required for entrance into these areas include, as appropriate:

1) Red-trimmed laboratory coat or coverall
2) Plastic or canvas show covers
3) Respirators.

Protective clothing and equipment for protection against potential hazards other than ionizing radiation will be prescribed on a case by case basis.

Respirators will only be fitted and issued by the RNS representative. No employee will be allowed to work in areas in which respirators are required unless he has been fitted and has completed the Rockwell/ESG Respiratory Protection training course within the past 12 months, including appropriate medical evaluation.

The RNS representative will establish respirator exchange frequencies as indicated by individual requirements. In addition to the protective clothing required for entry, certain additional items of clothing, such as skull caps or red-trimmed coveralls, may be required for certain operations posing high potential for contamination. Surgeons gloves will be required for operations involving direct handling of contaminated equipment. Surgical gloves shall be protected with an overglove. Persons exiting radiologically posted areas will remove their protective clothing at the change line and place the items of clothing in the drums, racks, or hangers provided as appropriate. Respirators will be returned to the plastic bag in which they were issued pending re-use or return to the respirator maintenance laboratory. Immediately upon exiting these areas, all persons will monitor their hands and feet with the count rate meter provided there. They will then proceed to the nearest washroom and wash their hands.

N. HANDLING OF CONTAMINATED PROTECTIVE CLOTHING

All reusable items of protective clothing will be removed from the facility for decontamination and reissue. Disposable items will be collected and disposed of as radioactive waste. Laundry drums, lined with 50-gallon plastic bags, will be provided at the change line for the accumulation of contaminated laboratory coats, canvas shoe covers, and coveralls. The contaminated laundry will be collected as the bags are filled and will be processed through a licensed vendor.

Waste drums, lined with plastic bags, will be provided at the change line for the accumulation of disposable items such as caps, plastic shoe covers, and surgeons gloves. This waste will be packaged as the bags are filled and will be processed for ultimate disposal.

O. INSTRUCTION OF PERSONNEL

Prior to beginning work in the radiologically posted areas, all employees will be indoctrinated with regard to radiation and industrial safety rules.

Employees whose regular assignments include for the first time work in radiologically posted areas, must complete a training course covering the general aspects of working with radioactive materials. This course will include (a) a description of the properties and potential hazards of radiation and radioactive material; (b) the basic principles of radiation protection; (c) the requirements of applicable Standard Operating

Policies and applicable regulations; (d) safe handling practices; and (e) emergency procedures.

P. EMPLOYEE QUALIFICATIONS

The Site Manager will furnish to Radiation and Nuclear Safety the names of all persons who will be assigned to work in the radiologically posted areas. Subsequently, whenever additional employees are to be assigned to work in these areas, Radiation and Nuclear Safety will be notified prior to each assignment. Radiation and Nuclear Safety will review the qualifications of persons assigned to work in the radiologically posted area and establish that these persons are fully qualified "radiation workers" and that they have sufficient familiarity with the operations in the posted areas to allow them to work safely in these areas. Included in the required qualifications or preparations for assignment to work in these areas are:

1) Personal film badge assignment
2) Bioassay baseline sample
3) Inclusion on periodic bioassay roster
4) Medical baseline examination
5) Inclusion on periodic medical examination roster
6) Completion of radiation worker training course
7) Completion of respirator training course
8) Successfully fitted with an approved respiratory protective device
9) Completion of facility indoctrination
10) Completion of required special training
11) No precluding physical limitations or radiological restrictions
12) NRC Form 4 or equivalent on file with Radiation and Nuclear Safety.

Rockwell Program Management or the Site Manager will also notify Radiation and Nuclear Safety of those persons whose assignments in posted areas are being terminated.

Q. INSTRUMENTATION

Radiation and Nuclear Safety will establish the requirements for radiological instrumentation, provide the instruments from general inventory if available, request calibration and repairs as required and instruct operations personnel in the use of these instruments as required.

Personnel monitors will be provided at change lines and in change rooms. Each of these monitors will consist of an alpha or beta-sensitive (as appropriate) detector, a count rate meter, and an audible "poppy-type" signal. These monitors will be inspected by Instrument Repair at least once each 3 months.

Continuous air monitors will be provided as required. These monitors will sample air through a filter media at a rate of about 1 cfm and will continuously monitor the particulate radioactive material collected on the filter media. The monitors will provide a ratemeter display of activity levels and an audible alarm which will actuate automatically in the event the radioactive material collected on the filter exceeds a preset level. These monitors will be serviced and calibrated at least once each 3 months.

Beta-gamma and alpha sensitive counting systems will be provided for use by the RNS representative in evaluating air samples and surface contamination samples for radioactivity. These systems will be serviced and calibrated at least once each 6 months.

Various types of beta-gamma and alpha sensitive portable radiation survey instruments will be provided for use by the Radiation and Nuclear

Safety representative in the day-to-day surveillance of operations in radiologically posted areas. All portable radiation survey instruments will be serviced and calibrated at least once each 3 months, or at shorter time intervals if recommended by the manufacturer.

R. DECONTAMINATION REQUIREMENTS

The requirements for decontamination in day-to-day operations will be determined by the RNS representative and communicated to the Site Manager who will assure that the required decontamination is performed.

1. Personnel Decontamination

In the event radioactive contamination is detected or suspected to be present on the skin or hair of an employee, the RNS representative will evaluate the degree of contamination and direct the decontamination efforts. In the event the contaminated employee is injured, the Site Manager will arrange for medical services. The RNS representative will direct or perform decontamination of the employee to acceptable limits using prescribed methods, unless it becomes apparent that further decontamination efforts will cause significant skin damage. In this case, the RNS representative will ask that further decontamination be accomplished under the direct supervision of a licensed practicing physician.

2. Equipment Decontamination

In the event that equipment, components, materials, etc., are found to be contaminated in excess of the appropriate limits, the RNS representative will promptly notify the Site Manager who will effect the required decontamination by operations personnel.

3. Area Decontamination

In the event the floors or walls of an area are found to be contaminated in excess of the appropriate limits, the RNS representative will notify the Site Manager and request decontamination. The RNS representative will coordinate decontamination efforts with operations personnel as necessary.

S. REMOVAL OF EQUIPMENT FROM RADIOLOGICALLY POSTED AREAS

All equipment or materials moving into unposted areas from any radiologically posted area must be surveyed for radiation and radioactive contamination levels. No item may be moved into any unposted areas if it is contaminated in excess of the limits established for such areas as shown in Table A-1. The radiation and contamination levels will be assessed by the RNS representative immediately prior to the transfer of the item. Required decontamination will be performed by the operations personnel.

In case of packaged items, the outer surfaces of the package will be surveyed for radiation and contamination levels, and these surfaces must be free of contamination in excess of the limits for unposted areas as shown in Table A-1. Packaged contaminated items will comply with the provisions of Title 49 Code of Federal Regulations and will be tagged with a completed radioactive materials tag prior to transfer into radiologically unposted areas.

T. RESTRICTED ACCESS AREA ENTRY PERMIT (FORM 719-L)

Varying degrees of control, consistent with the hazard involved, are exercised over posted areas by Radiation and Nuclear Safety. The Form 719-L is a means of restricting access to posted areas on the basis of personnel and potential hazards. The highest degree of hazard is associated with an area in which the active contamination and radiation levels are of such significance that special rigid entry controls and safety precautions are necessary.

1. <u>Subcontractors</u>

Subcontractor personnel who have cause to work within any radiologically posted area must have a completed Form 719-L which will be submitted by the Site Manager and approved by the RNS representative prior to the start of work. Groups representing the same contractor and who work in the same general area need only one tagged area entry permit.

All contractor personnel entering any radiologically posted area must obtain a film badge prior to entry. Radiation and Nuclear Safety shall determine if previous entries into posted areas had been made during the current calendar year and if so, shall ascertain from personnel monitoring records the dose received, and plan radiation exposures accordingly.

Contractor personnel performing work in a radiological posted area will be surveyed prior to breaks, lunch and quitting time by means of portable battery operated or ac survey instruments. If the instrument survey indicates contamination, decontamination will be effected immediately.

Radiation and Nuclear Safety will attend any operation involving contractor personnel in a posted area to the extent necessary to ensure that such personnel perform their duties in such a manner as not to cause the release of radioactive material or become unduly exposed to radiation. Should such an event occur, work will be stopped until appropriate surveys have been performed and necessary corrections effected.

All tools and equipment used by contractor personnel in radiologically posted areas must be surveyed and found to be free of contamination before they may be removed. The removable contamination limit for tools and equipment shall be 20 dpm/100 cm^2 alpha activity and 100 dpm/100 cm^2 beta. Fixed contamination shall be undetectable with appropriate portable survey instruments.

Contractor personnel shall not be exposed to concentrations of radioactive material in air and water greater than 10% of the maximum permissible concentrations as listed in 10 CFR 20, "Standards for Protection Against Radiation," under Table I of Appendix B unless they are qualified as radiation workers as described below.

Whole body dose to contractor personnel will be limited to 500 mrem/year. An exception to the standard 500 mrem/year will be made if an affidavit, signed by a representative of the contractor, authorizes their employees to be considered radiation workers, in which case the employee will be required to execute an NRC Form 4 or equivalent authorizing Rockwell/ESG to obtain occupational radiation exposure histories.

Cumulative records or radiation exposures will be maintained by Radiation and Nuclear Safety to ensure that personnel are not exposed in excess of applicable standards.

If a contractor employee receives a radiation exposure in excess of 25 mrem, Rockwell/ESG will notify the contractor of the dose within 30 days following the determination of such exposure.

Contractor personnel under 18 years of age will be limited to 125 mrem/calendar quarter.

Approved visitors will be considered in the same category as outside contractors and must have completed a Form 719-L prior to performing work in radiologically posted areas, or visiting such facilities for extended periods during which they are not under continuous escort.

2. Rockwell Employees - Other Than Personnel Assigned to the DORF D&D Program

Personnel who have cause to enter the radiologically posted areas for maintenance or repair purposes will complete a Form 719-L prior to entry.

3. Restricted Access Areas

All personnel who have reason to enter certain rigidly controlled areas, "Restricted Access Areas," such as radioactive exhaust system filter plena or liquid waste holdup tanks, will complete a Form 719-L prior to entry into these areas. A minimum of two persons will be assigned to perform operations in these areas, or as otherwise specified.

4. Preparation of Form 719-L

When a Form 719-L is required, the individual or manager of the group requesting entry will fill out his portion of the form and give it to the RNS representative who will outline the pertinent radiological safety instructions, sign the form, and return it to the originator.

The originator will obtain the signature of the Site Manager and distribute copies as required.

The entire working crew will initial the Form 719-L, signifying receipt and comprehension of instructions on the form.

Copies of completed Forms 719-L will be kept on file indefinitely by Radiation and Nuclear Safety.

U. EMERGENCY CONDITIONS

1. Ventilation Loss or Airborne Radioactivity Alarms

In the event of a radioactive exhaust system failure, or of other evidence of loss of airflow in ventilated areas, personnel will leave these areas and await evaluation of the facility conditions by Radiation and Nuclear Safety.

In the event of actuation of the "variable warble" and/or bell alarm of continuous air monitors, personnel present will evacuate the facility and await evaluation by Radiation and Nuclear Safety.

V. RADIOACTIVE WASTE MANAGEMENT

All radioactive waste will be collected, evaluated, processed, and shipped for disposal to a licensed radioactive waste disposal site.

1. Solid Waste

Low level solid radioactive waste will be packaged into standard containers such as steel DOE Specification 17-H 55-gallon drums or wood type box DOT Specification 19-A or low specific activity, strong, tight containers. In the case of low level waste such as concrete rubble which is generated in large volumes, special containers may be designed. Low-level solid radioactive waste generated in support of decontamination operations (i.e., plastic shoe covers, surgeons gloves, kim-wipes, miscellaneous plastic, etc.) will be collected in drums lined with plastic bags. When the bags are filled, they will be final packaged for disposal.

High-level radioactive waste will be packaged and shipped in containers such as reusable lead-shielded casks, one-way concrete shielded containers, or DOT approved overpacks.

2. Liquid Waste

Liquid radioactive waste will be solidified. In the event there is other liquid radioactive waste such as acids or corrosives, they will be neutralized in drums and then solidified for disposal by land burial. Solidification will consist of cementation.

W. INDUSTRIAL SAFETY REQUIREMENTS

Industrial safety requirements are described by Rockwell/ESG Health and Safety Procedures, SOP's ANSI, AEC Manaul, and OSHA.

1. Hoisting and Rigging

Hoisting and rigging operations shall be conducted in compliance with the requirements of 29 CFR 17, Part 1910, Subpart N, and the ANSI B30 Series.

Equipment used in material handling shall be proof-loaded and maintained per PL Series 8.

All personnel engaged in hoisting and rigging shall be qualified by appropriate experience and training.

2. Explosives

Explosives use and handling shall be in accordance with Federal, State, and local regulations including 29 CFR 17, Part 1910.109, Health and Safety Procedure G-16, and Section XXV, EM 385-1-1, "General Safety Requirements Manual," of the Corp of Engineers with the exception of Paragraph 25.B.04.

3. Insulation Removal

The removal of insulation material containing asbestos will be done in compliance with 29 CFR, Part 1910.1001.

Warning signs, OSHA approved, will be displayed at each location where the airborne concentration of asbestos fibers may exceed the allowable exposure limits. The airborne concentration will be verified by environmental sampling.

Insulation will be removed in a manner that will minimize the generation of airborne dust. Wet methods will be used if practical. Approved respirators will be worn during all asbestos insulation removal operations. If the presence of asbestos in a material is questionable, it will be assumed to be present.

4. Burning, Cutting, and Welding

Burning, cutting, and welding will be performed only as authorized by the Site Manager. Additional requirements may be imposed by the RNS representative as necessary to protect against toxic and/or radioactive vapors and fumes.

5. Noise

Personnel exposed to noise in excess of the limits specified in 29 CFR, Part 1910.95 will be required to wear approved ear protection. The noise level for an 8-hour exposure is 90 dbA. Personnel exposures to noise over 90 dbA shall be evaluated by the RNS representative. Personnel exposures to noise in excess of 115 dbA are prohibited without approved ear protection.

6. Confined Space Entry

Confined space entires will be made in compliance with Health and Safety Procedure G-19. All Class 2 entires will require the preparation of a Form 719-L, Rev. 5-70, "Restricted Access Area Entry Permit," which will designate the required control measures.

7. Contractor Safety

Contractors will conduct operations in compliance with Federal, State, and local codes, standards and regulations as applicable. Contractors are subject to compliance with Rockwell International and ESG regulations as indicated in Form 511-C which describes Contractor Safety Requirements at Rockwell International facilities.

HSWP-QHP 23 September 1977

SUBJECT: Review of Plan for the Decommissioning of DORF, dated 30 Jun 77

Mr. James McGarrity
Chairman, Reactor Safeguards Committee
Harry Diamond Laboratory
2800 Powder Mill Road
Adelphi, MD 20783

The Plan for the Decommissioning of the Diamond Ordnance Radiation Facility, dated 30 Jun 77, has been reviewed per your request, and the following changes are recommended:

Section II A, add:

6. Walter Reed Army Medical Center Health Physics Officer (WRAMC HPO). The WRAMC HPO within his duties as WRAMC Radiation Protection Officer has the authority to monitor all operations during decommissioning to assure the safety of WRAMC personnel and the general public.

Section II B, add:

5. WRAMC

WRAMC will provide support services during decommissioning as described in the Intraservice Agreement between Commander, WRAMC and Commander, HDL. This support does not include Health Physics and dosimetry support of contract operations and personnel.

Section III A.4: Change to read "The reactor pool water will be discharged to the sanitary sewer provided the radioactivity in addition to that released by WRAMC is within the standards set by 10CFR20."

Section VI A.2.a(3): Change to read "Contractors will provide personnel monitoring for their own personnel."

Section VI A.2.b., add:

(4) Whole body badge: Neutron.

Section VI A.3a: Change line 1 to read "WRAMC Health Physics will supply the film badges except for contractor personnel."

Section VI A.3c: Change all to read "Contractors will maintain their own records of radiation exposure."

Section VI B: Liquid Waste Processing. Add to end of paragraph: "All discharges to the sanitary sewer will be coordinated with WRAMC HPO.

Section VI C.1: Plant Surveys. Add in line 9 after "Additional Surveys and activation analysis:" "These additional surveys and activation analysis will have to be performed by contract or personnel other than WRAMC Health Physics."

Section VI C.2: Contamination Control Plan. Add: "(For DORF Personnel and Equipment Only) contractor will be responsible for the Contamination Control Plan in his job area."

BOBBY R. ADCOCK
LTC, MSC
Health Physics Officer

Lt Col. Adcock
WRAMC - HPO

HARRY DIAMOND LABORATORIES

ADELPHI, MARYLAND

THE PLAN

FOR

THE DECOMMISSIONING OF THE DIAMOND ORDNANCE RADIATION FACILITY

AT

FOREST GLEN SECTION

WALTER REED ARMY MEDICAL CENTER

30 JUNE 1977

PREPARED BY:

WALTER L. GIESELER
DORF-PIC

Reviewed
~~APPROVED~~ BY:

PAUL CALDWELL
CHIEF, NUCLEAR EFFECTS
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JAMES M. McGARRITY
CHAIRMAN, REACTOR SAFEGUARDS COMMITTEE

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APPENDICES

I. Introduction

A. Background. The Harry Diamond Laboratories (HDL) operates the Diamond Ordnance Radiation Facility (DORF) which utilizes a research reactor with associated experimental equipment. The facility occupies a single remote building on 4.2 acres of the Forest Glen Annex of the Walter Reed Army Medical Center (WRAMC) near Silver Spring, Maryland. An intraservice agreement between the Commanding Officer, WRAMC and Commanding Officer, HDL establishes the WRAMC support services for DORF.

The reactor is the familiar General Atomic Company TRIGA Mark F, moderated by light water and mounted on a track support carriage assembly which can be moved through a 15,000 gallon capacity pool. The reactor core consists of 85 (maximum 87) fuel elements, four control rods, neutron source, and miscellaneous neutron detectors. The fuel elements are composed of zirconium hydride moderator homogeneously combined with 20% enriched uranium fuel. The control system consists of borated graphite safety, shim, regulating and pulse control rods, having either solid aluminum or fuel followers. Experiments are conducted in a 20 x 20 x 8 foot high fast neutron exposure room adjacent to pool, the pool itself, and within the core.

The facility was orginally developed in late 1959 and began operations in September 1961. Modifications applied since then include (1) replacement of the aluminum clad fuel elements with stainless steel clad elements (1964), (2) automatic SCRAM timing (1969), (3) replacement of the poison-followed transient rod with an aluminum follower (1964), (4) replacement of aluminum follower control rods with fuel-followed control rods (1971), and (5) replacement of reactor instrumentation with up-to-date instrumentation (1973).

The reactor has the capability of the following modes of operation:

1. Steady-state operation up to 250 kW.

2. Square-wave operation up to 250 kW.

3. Pulse operation resulting in up to a maximum peak power of 2000 MW with a pulse width of 9.5 ms at half maximum.

B. The decision to decommission the DORF reactor is the culmination of an Army reactor utilization study begun in mid-1975 to examine the requirement for the three Army research reactors. This study was done by HDL Nuclear Weapons Effects Program Office (NWEPO) which investigated the following alternatives:

1. Operation of the three reactors in their present locations.

2. The shutting down of one pulse reactor.

3. The consolidation of the Aberdeen Pulse Reactor with the Diamond Ordnance Radiation Facility (DORF) at either Forest Glen, Md. or Aberdeen Proving Ground, Md.

4. The consolidation of the DORF with the White Sands Missile Range (WSMR) pulse reactor with the closing of the Aberdeen Pulse Reactor.

5. Closing down of all Army effects research reactors with experiments conducted at either ERDA or other service reactor sites.

6. The closing of one pulse reactor and the DORF facility.

In addition, an Army Scientific Advisory Panel Ad Hoc Group on Pulse Reactors was asked to review the technical capabilities of the Army's two pulsed reactor facilities at WSMR and APRF and to review documentation including the Study for Requirements for Nuclear Weapons Effects Research Reactors prepared by NWEPO, HDL. The Ad Hoc Group also visited DORF and although they were not asked to address the DORF facility, their conclusion was that if any of the three reactors should be closed, DORF would be the logical candidate. This conclusion is based on the availability of TRIGA type reactors at the Armed Forces Radiological Research Institute and other facilities nationally. In January 1977 the HDL were directed by Headquarters, US Army Materiel Development and Readiness Command to decommission DORF with an initial closure date of 1 July 1977. The closure date was subsequently changed to 1 October 1977 so that planned experiments could be completed.

C. The primary objective of the decommissioning of the DORF is: (1) to remove the Special Nuclear Material (SNM), i.e., reactor fuel elements, and to return it to US Energy Research and Development Administration (ERDA) for disposal; (2) to remove all radioactive material from the facility and ship to a Nuclear Regulatory Commission (NRC) licensed burial site; and (3) to decontaminate and prepare the facility building for alternate use. Upon completion of decommissioning the facility building will be given to Walter Reed Army Medical Center (WRAMC) for their use. The purpose of this Decommissioning Plan is to outline the method of accomplishing this objective in a safe manner which will allow least exposure to personnel or contamination to the environment.

D. In decommissioning the DORF, the removal and shipment of the fuel elements is relatively straightforward but time consuming. There are approved TRIGA fuel shipping casks available, however they may not be available until early 1978 or later. Because of the Army's difficulty in obtaining indemnification, the shipment of the fuel will be performed by ERDA. The precise details of all the decommissioning and dismantling operations cannot be specified since tasks to be accomplished at a latter period are dependent upon results and judgement of earlier operations. For example, after the removal of the wood lining in the exposure room, the amount of concrete to be removed will depend upon its radioactivity. Therefore, this Plan will not give detailed procedures which may change at a latter date, but will give firm objectives which must be accomplished to complete the decommissioning operations. The detailed procedures and hazard analysis will be attached to the plan as appendices.

II ORGANIZATION AND RESPONSIBILITIES

A. The health and safety aspects of the Department of the Army nuclear reactor systems is a command responsibility. The specific command responsibilities are stated in AR 385-80, Chapter 2. The organization chart for the DORF Decommissioning is shown in Fig. 1.

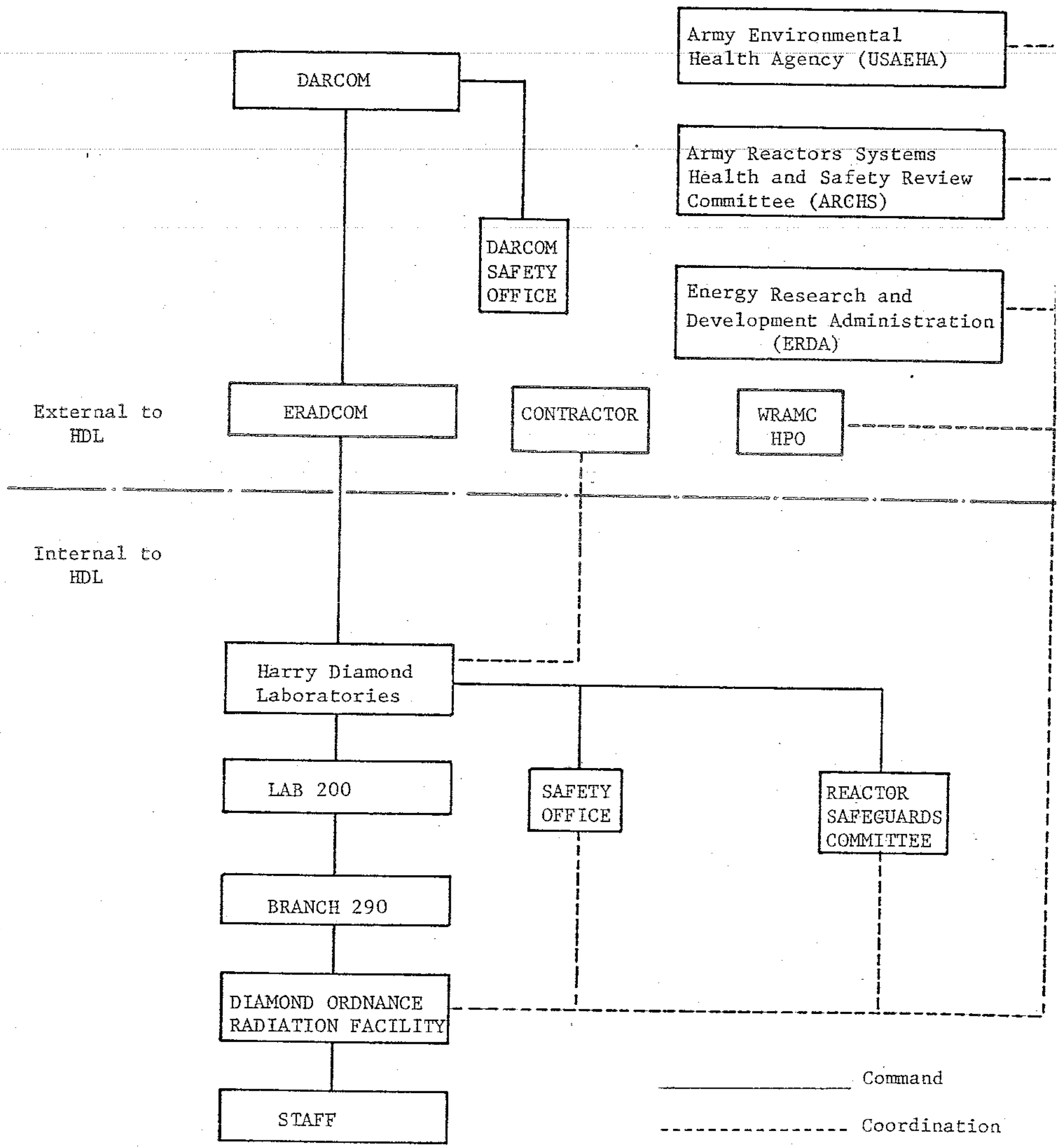
1. Responsible Commander. The Commanding Officer, Harry Diamond Laboratories is the responsible commander. He has ultimate responsibility for the operation of the reactor. He shall comply with the requirements of AR 385-80 and insure that all operations of the DORF reactor systems are conducted in a safe manner.

2. Reactor Safeguards Committee. The Reactor Safeguards Committee (RSC) is appointed by and reports to the Commanding Officer, HDL. The committee consists of personnel who collectively provide a broad spectrum of experience in reactor technology. The RSC will review the plans and procedures of decommissioning.

3. Chief, Nuclear Radiation Effects Laboratory (Lab 200). The Chief, Lab 200 is the direct line of authority from the reactor commander to the HDL commanding officer.

4. Reactor Commander. The Chief, Nuclear Effects Simulation Technology Branch (Br. 290), is the reactor commander as defined in AR 385-80. He has the direct responsibility for the safe, competent and efficient operation and use of the facility. These responsibilities will continue through DORF decommissioning.

5. Physicist-in-Charge. The Physicist-in-Charge (PIC) is the Facility Supervisor. He has the direct and immediate responsibility for the facility. He is responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the DORF Technical Specifications. These responsibilities will continue until the DORF decommissioning is concluded.



ORGANIZATION FOR DECOMMISSIONING

Figure 1

B. Support from Other Commands and Agencies

1. Army Reactors Systems Health and Safety Review Committee (ARCHS)

The Safety Director, Office of Chief of Engineers, has been delegated the responsibility of surveillance and general guidance in all health and safety matters relating to Army nuclear reactor systems, which includes the decommissioning of DORF. The Army Nuclear Reactor Systems Health and Safety Review Committee (ARCHS) was established to assist in discharging this responsibility. The ARCHS will review and approve this decommissioning plan, safety analysis and other applicable documentation.

2. US Army Environmental Health Agency (USAEHA)

Following decommissioning and in-house radiological surveys, the USAEHA will be requested to conduct an independent radiological survey of the DORF site.

3. US Energy Research Development Administration (USERDA)

ERDA will be requested to supply the fuel transfer casks and the transportation of the fuel elements to the disposal site.

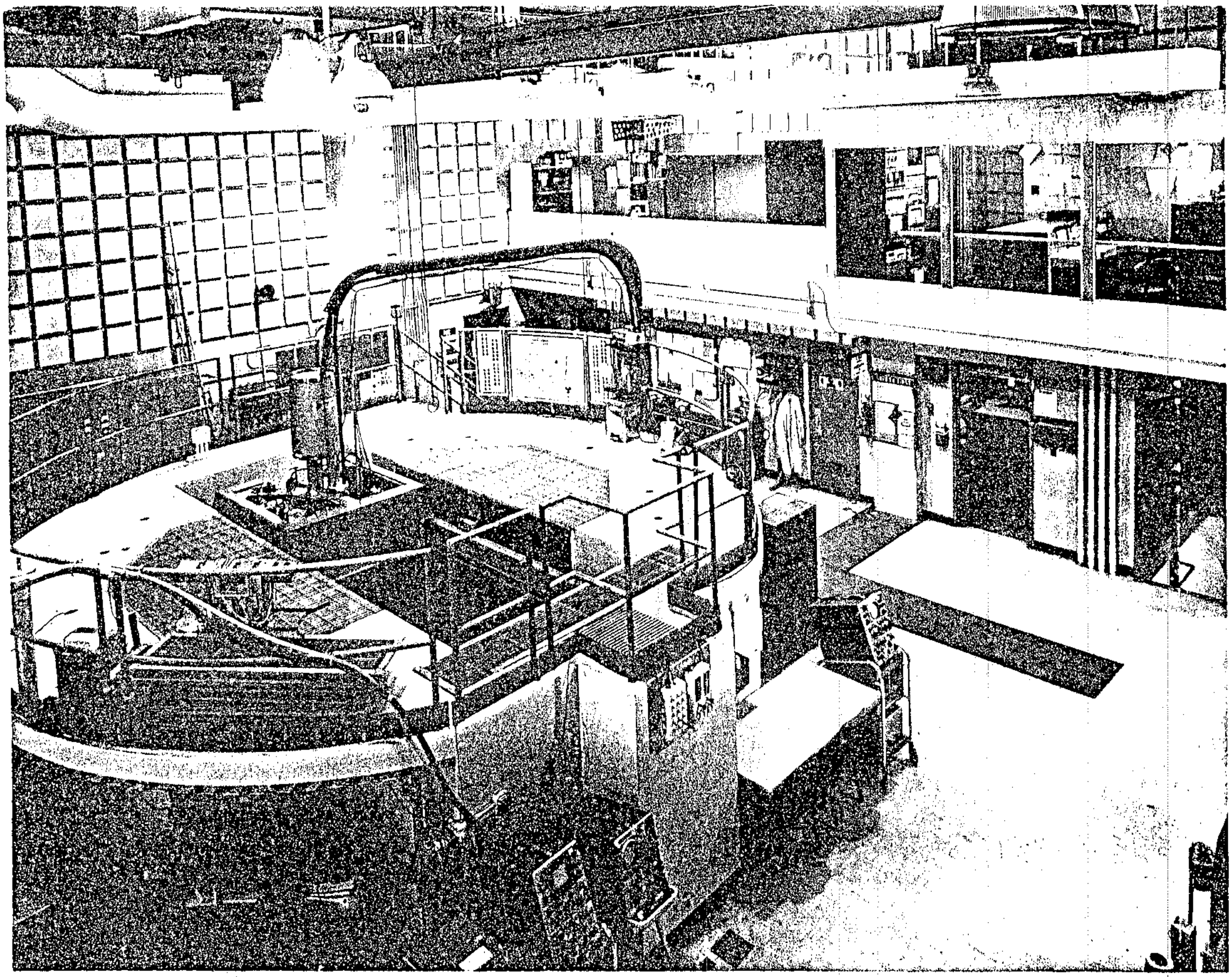
4. Civilian Contractor

A contract will be negotiated for the dismantlement and decontamination of the facility, the packing and shipment for disposal of radioactive waste, and the preparation of the building for alternate use.

III OBJECTIVES

A. Summary of Tasks

The facility decommissioning will begin immediately after the approval of this Decommissioning Plan. The decommissioning operation will be completed when all accessible areas of the facility are radiologically safe and unrestricted for personnel occupancy. The Energy Research and Development Administration ERDA has been asked to negotiate the fuel cask and shipment contract. This approach is used because ERDA is indemnified and the Army is not. The fuel elements will be sent to the chemical reprocessing plant in Idaho or to other TRIGA reactor facilities specified by the University Relation Division, ERDA. The dismantling, removal of radioactive equipment, and building decontamination will be done by contract. The Decommissioning Plan will be prepared by HDL personnel and approved by the Army Reactor Systems Health and Safety Review Committee (ARCHS) before implementation. The overall layout of the reactor is shown in Figures 2 through 5. Generally, the decommissioning operation will proceed as follows:



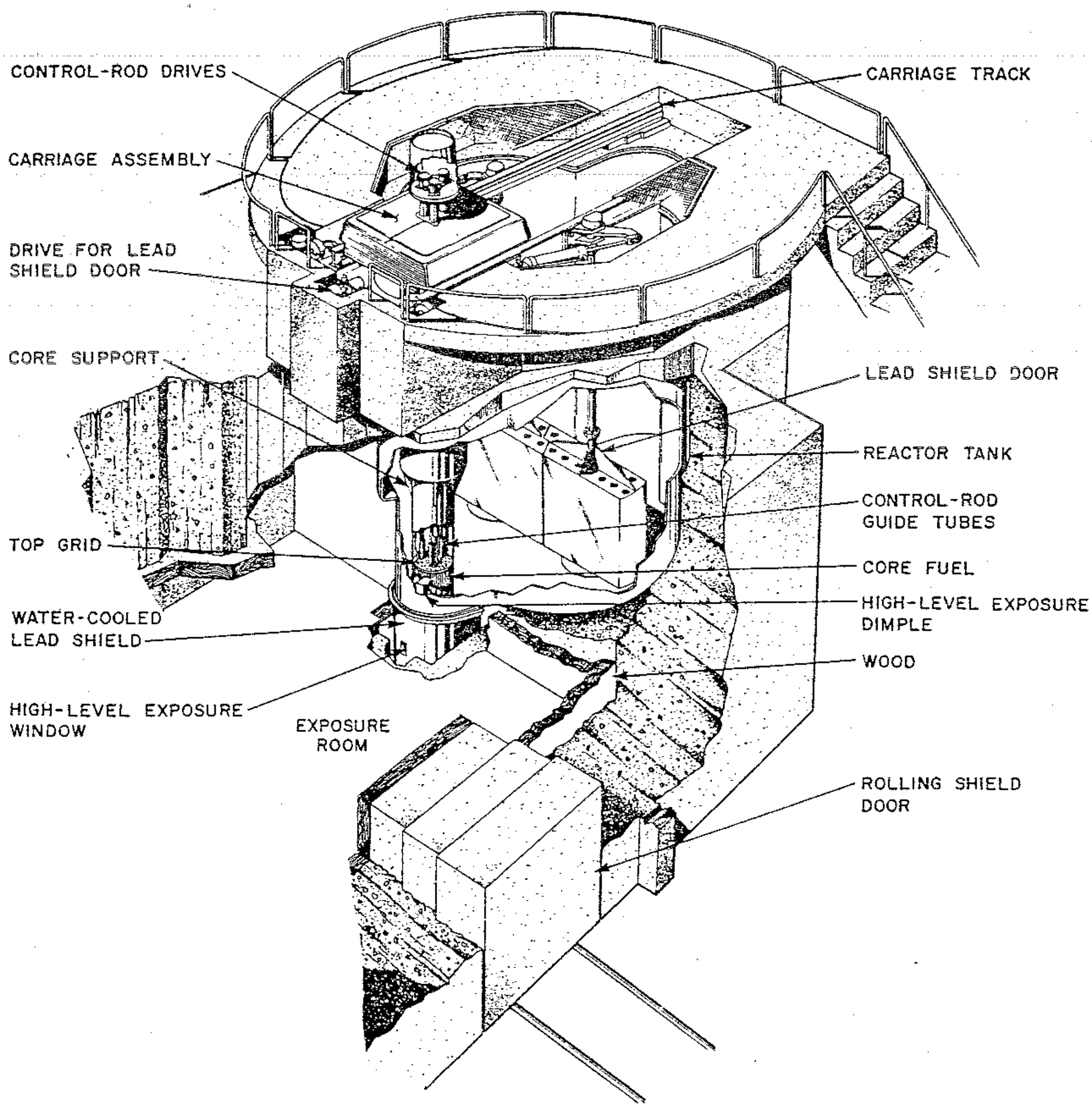


Fig. 5-1 -- Perspective view of DORF reactor

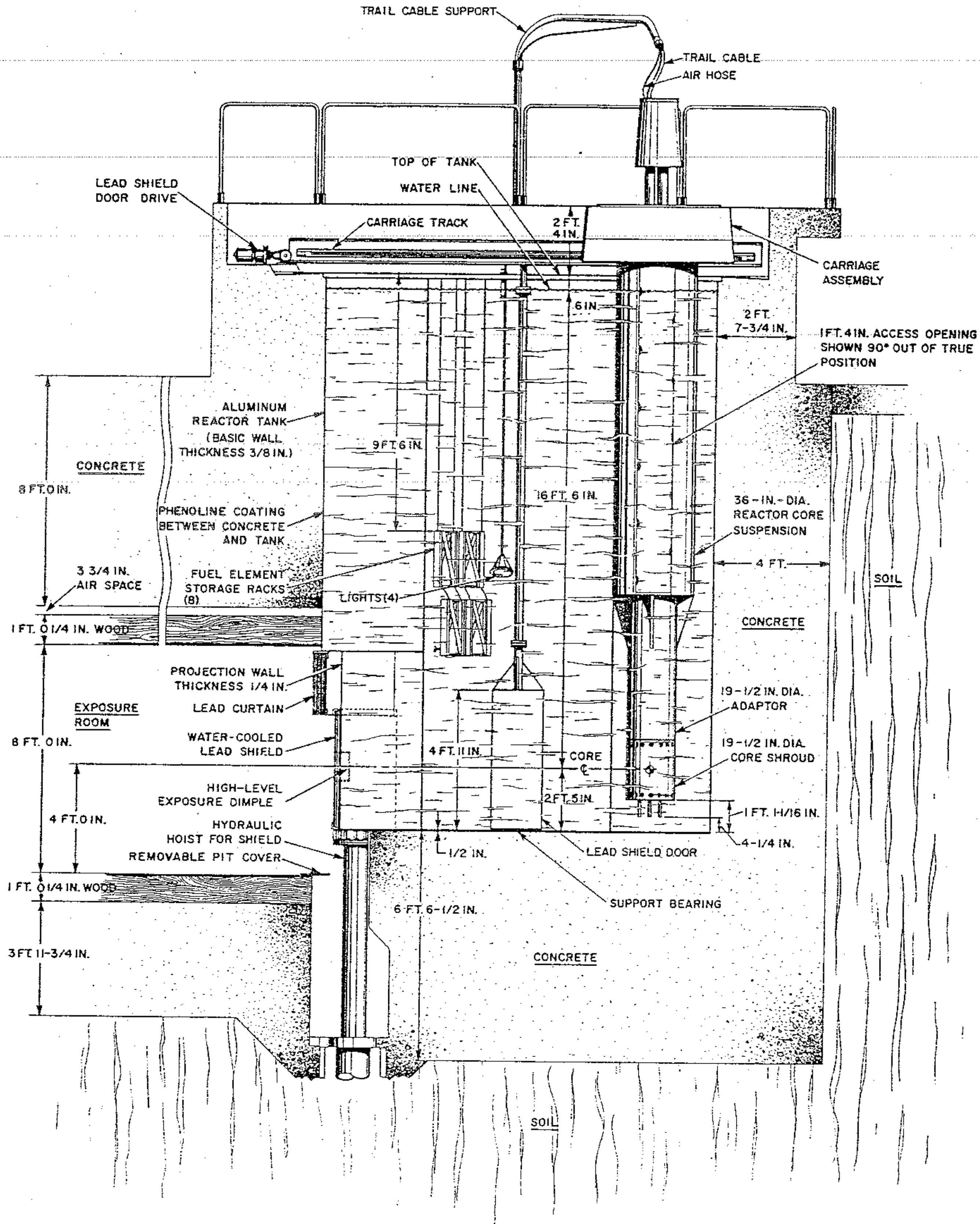


Fig. 5-3 -- Sectional elevation of DORF reactor

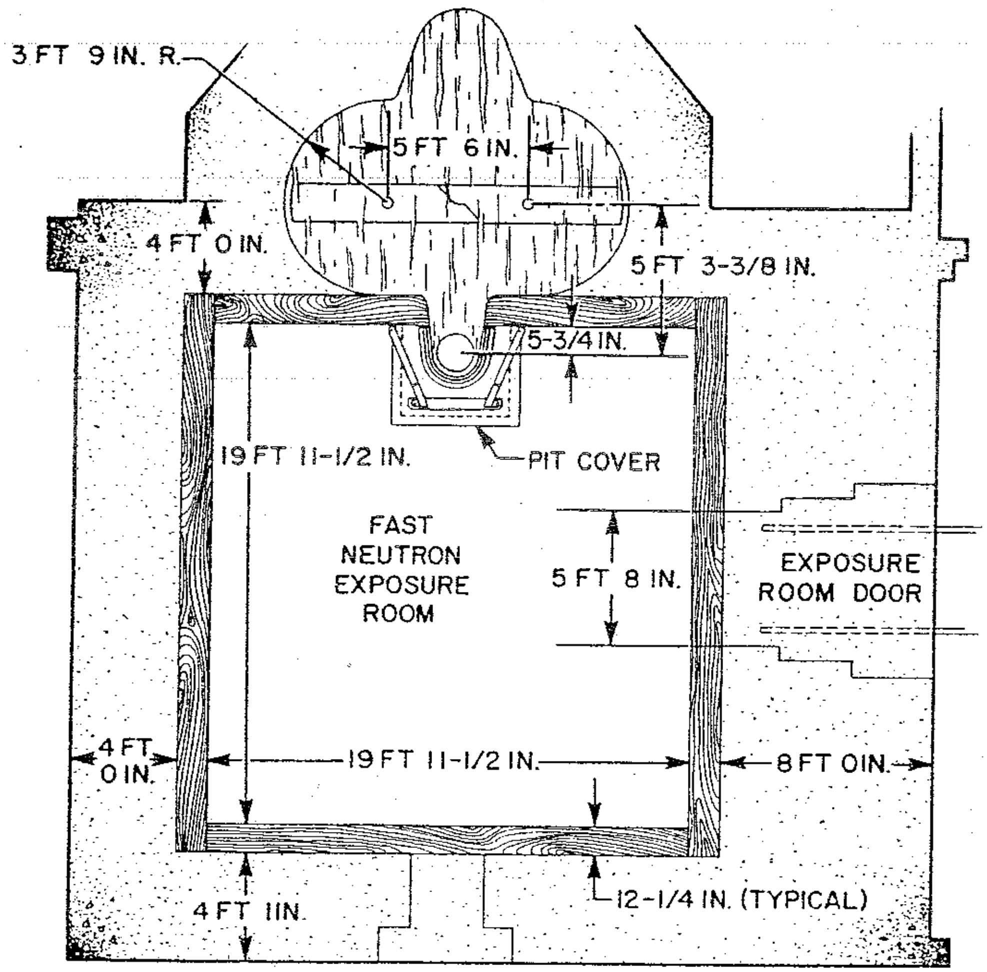


Fig. ~~7-2~~ 5 -- Plan view of DORF exposure room

1. The reator fuel will be transferred from the reactor core to the in-pool storage racks. (Note: This will be done immediately after operations cease on 1 Oct 1977. These are approved standard pool storage for the fuel elements and storage will assure reactor non-operability).

2. The fuel elements and fuel-followed control rods will be packaged in approved fuel shipping casks and shipped to the US Energy Research and Development Administration (ERDA), Idaho Reprocessing Plant. There is a possibility that some of the irradiated fuel elements will be shipped to other TRIGA reactor facilities. The loading will be performed by DORF personnel. The fuel shipment will be performed by ERDA on a cost recoverable basis. Unirradiated fuel elements will be transferred to the Armed Forces Radiological Research Institute (AFRRI).

3. Standard control rods and other radioactive reactor structures will be shipped from the facility for disposal at NRC licensed burial site. Selected reactor structures will be shipped to interested TRIGA reactor owners.

4. The reactor pool water will be discharged to the sanitary sewer provided the radioactivity is within the standards set by 10CFR20.

5. The reactor carriage, core support structure, and pool shield doors will be removed and activated sections will be shipped for disposal at a NRC licensed burial site.

6. The aluminum pool tank will be removed. Radioactive sections will be shipped from the facility site to an approved burial site.

7. Sufficient activated concrete beneath the aluminum liner will be removed to comply with acceptable surface contamination levels in NRC regulatory Guide 1.86, Termination of Operating Licenses for Nuclear reactors. (See Table 1)

8. Wood liner in the exposure room will be removed and shipped for disposal at a NRC licensed burial site.

9. Activated concrete in exposure room and exposure room plug door will be removed and shipped for disposal at a NRC licensed burial site. Sufficient concrete will be removed to comply with contamination levels in Table

10. Exposure room lead shield and hoist and lead curtain will be removed and activated sections shipped for disposal at a NRC licensed burial site.

11. All activated structures will be decontaminated or removed for disposal

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

12. Reactor console and associated equipment will be removed by HDL personnel and shipped to AFRRI.

13. Building will be prepared for alternate use by removing parapet and restoring floor to bay level.

14. Remove three 5000 gallon waste water hold-up tanks and reconnect sewer system. Tanks to be shipped for disposal as non-radioactive waste.

15. A final site survey will be made by an independent agency to insure that none of the areas will exceed NRC guidelines for unrestricted use.

IV TRAINING

A. DORF Personnel. Fuel element handling will be done by DORF personnel who have the experience and training in this operation. Special training and briefings will be conducted prior to transfer of the fuel elements to shipping casks. Other special training will be conducted, as appropriate, for those tasks not performed during normal reactor operations. Removal of large items requiring rigging techniques will require personnel (contractor) who are familiar with these operations. Training will be on-site and will be the responsibility of the Physicist-in-Charge.

B. NON-DORF Personnel. All non-DORF personnel associated with the decommissioning will receive a briefing which will include reactor facility entrance and exit procedures, radiation areas and exclusion areas, the identification and meaning of various radiation signs, controlled areas, and use of radiation monitoring devices.

C. SUPPORT UNITS. Periodic training of support units such as the fire department personnel and WRAMC military police is their own responsibility. These units have participated in DORF quarterly emergency drills and will be apprised, and briefed when necessary, of decommissioning operations.

V. QUALITY ASSURANCE

A. The DARCOM requirements for quality assurance at reactor facilities is contained in DARCOM Supplement 1 to AR 385-80. The surveillance requirements and management procedures in the DORF Technical Specifications, and ENRADMON and Health Physics Plans are designed to assure that adequate control over activities affecting all aspects of the reactor system operations are maintained. Additional quality assurance during decommissioning will be implemented, as necessary, in accordance with the above DARCOM Supplement and present DORF procedures and plans.

VI. RADIATION PROTECTION

A. Personnel Monitoring

1. Basic Radiation Protection Standards

a. The basic radiation protection standards are prescribed in Army Regulation 40-14, and Walter Reed Army Medical Center Regulation 40-10. Every effort will be made to maintain the radiation dose equivalent as far below the following Radiation Protection Standards as practicable. Positive efforts will be carried out to fulfill this objective; and, determination of necessity will be weighed against the benefits to be expected.

b. Basic Radiation Protection standards adopted for the control of occupational exposures to ionizing radiation include:

(1) The accumulated dose equivalent of radiation to the whole-body; head and truck; active blood-forming organs; gonads; or lens of the eye will not exceed:

(a) 1.25 rem in any calendar quarter, nor

(b) 5 rem in any 1 calendar year

(2) The accumulated dose equivalent of radiation to the skin of the whole-body (other than hands and forearms); cornea of the eye; and bone will not exceed:

(a) 7.50 rem in any calendar quarter, nor

(b) 30 rem in any 1 calendar year

(3) The accumulated dose of radiation to the hands and wrists or the feet and ankles will not exceed:

(a) 18.75 rem in any calendar quarter, nor

(b) 75 rem in any 1 calendar year

(4) The accumulated dose of radiation to the forearms will not exceed:

(a) 10 rem in any calendar quarter, nor

(b) 30 rem in any 1 calendar year

(5) The accumulated dose equivalent or radiation to the thyroid; other organs; tissues; and organ systems will not exceed:

(a) 5 rem in any calendar quarter, nor

(b) 15 rem in any 1 calendar year

(6) Individual(s) under 18 years of age, females known to be pregnant, and occassionally exposed individual(s) will not be exposed to a whole-body dose equivalent of more than:

(a) 2 millirem in any 1 hour, nor

(b) 100 millirem in any 7 consecutive days, nor

(c) 500 millirem

(d) more than 10 percent of the values in (2), (3), (4) and (5) above, for other areas of the body

(7) Individuals over 18 years of age but who have not yet reached their 19th birthday may be occupationally exposed to ionizing radiation provided that they do not exceed 1.5 rem dose equivalent to the whole-body in any calendar quarter, nor 3 rem in the 12 consecutive months prior to their 19th birthday.

2. <u>Monitoring Devices</u>

a. A film badge is the primary dosimeter device for monitoring personnel exposure. Personnel selected for personnel monitoring will include:

(1) Individuals who are likely to be exposed to sufficient radiation from all occupational exposures to receive an accumulated dose in excess of ten (10) percent of the applicable basic Radiation Protection Standard.

(2) Those other individuals selected by the Health Physicist on duty.

(3) Contractors will provide dosimeters for their own personnel.

b. The following types of personnel monitoring devices will be available:

(1) Whole Body badge: Sensitive to beta, x-ray and gamma radiation and worn to measure the exposure received by the whole body

(2)

(2) Wrist badge: Same as the whole body except that it is provided with a wrist band so that it can be used to measure the dose to the hands

(3) Pocket ionization chambers will be used to provide a means of obtaining rapid indication of accumulated dose over short periods of time. This chamber enables individuals to monitor their own accumulated dose.

3. Documentation

a. WRAMC Health Physics will supply the film badges. Records are kept on WRAMC Form 119 (Film Badge Application and Record of Occupational External Radiation Exposure). Health Physics will exchange film badges and transmit the film packets, along with photodosimetry reports, to Lexington-Bluegrass Army Depot for monthly development and exposure evaluation. Records of exposure will be maintained as follows:

(1) The Lexington-Bluegrass Army Depot maintains permanent records of all exposures and returns the Photodosimetry Report (DA 3484) to the WRAMC Health Physics Officer.

(2) The WRAMC Health Physics Officer maintains DD Form 1141 for all military and civilian personnel assigned or attached to WRAMC.

b. Cumulative daily Pocket ionization chamber dose readings will be recorded on WRAMC Form 705, "Pocket Dosimeter Log".

c. In general, contractors will maintain their own records of film badge exposures. However, they will be supplied pocket dosimeters at the direction of the Health Physicist and exposure records will be maintained on Form 705.

NO

4. Use of Monitoring Devices

Use of personnel monitoring devices is contained in WRAMC Regulation 40-10, Annex E (PERSONNEL MONITORING). During the DORF decommissioning the following general guidelines will be followed:

a. Film badges: Each person who occupies a controlled area will wear a film badge, (beta, gamma) unless specifically exempted by the Health Physicist. These badges will be worn at all times within the controlled area and left at the facility when leaving. Neutron and wrist badges will be worn at the direction of the Health Physicist.

b. Pocket dosimeters; Each person who occupies a controlled area will wear a self-reading pocket dosimeter to provide an immediate indication of accumulated dose over short periods of time.

c. Contract specifications will require, as a minimum, that the contractor will supply the above (a and b) monitoring devices for their personnel.

5. Administrative Control

a. Staff Personnel: The fuel elements will be removed from the reactor pool to the shipping casks by DORF staff personnel who have had experience in handling TRIGA Fuel elements. All personnel will be briefed on tasks and hazards that exist. Rehearsals and "dry runs" will be made to reduce radiation exposure and to reduce hazards where applicable.

b. Contractors and Visitors:

(1) Contractor work will begin after the fuel elements have been removed and shipped from the facility. Therefore, the major source of radiation hazard will be gone when they begin the facility dismantling and decontamination. However, the contractors will be throughly briefed on the location of the remaining radioactive material and hazards involved, also use and control of monitoring devices and safety precautions to be utilized.

(2) Visitors will be accompained at all times when within the controlled area. The existing entrance sign-in-log and personnel dosimeter monitoring procedures will be followed.

B. Liquid Waste Processing

All liquid waste at the facility, except comode and urinals, drain into 15,000 gallon hold-up tanks. Before discharge to the sanitary sewer, the tank affluent is sampled and analyzed for radioactivity to insure that the maximum permissible concentration (MPC) listed in 10CFR20 is not exceeded.

C. Radiological Surveys

1. Plant surveys. The procedures for radiological surveys are contained in the DORF Health Physics Plan. These procedures insure that the facility is provided with adequate routine radiological daily, weekly and monthly checks. Particular attention will be made of radioactive dose rates during fuel transfer to shipping casks and during dismantlement of radioactive structures to insure minimum exposure to personnel. Additional surveys and activation analysis will be done during concrete removal to determine when sufficient material is removed so that the area can be made an unrestricted area. The Health Physicist will inform and advise persons of anticipated or existing radiation hazards and these hazards will be posted when required.

2. Contamination Control Plan. Control points will be established at the point of entry of any contaminated area. The procedures established in the Health Physics Plan will be followed. Materials and tools removed from the contaminated area will be surveyed. It is the goal of the decommissioning to remove sufficient activated materials so that the residual radiation levels do not exceed the values in Table 1.

3. Effluent Monitoring. Air borne effluents will continue to be monitored by the existing stack monitor as required by the DORF Technical Specifications. Local air borne activities will be monitored using a Staplex Air Sampler as directed by the HP. All building air is exhausted through the absolute filters and out the stack.

VII DOCUMENTATION AND RECORDS

A. Logs

1. Decommissioning log. A log will be kept of all events, as they occur, during decommissioning. The purpose of the log is to present as a record of all the events, drawings, sketches and other information that are required to perform decommissioning tasks. As a minimum, the following will be recorded:

a. Daily job status

b. Initiation and completion of each task

c. Personnel injuries

d. Accidents

e. Schedule delays

f. Facts bearing on any problem areas

g. Major shipments (incoming and outgoing)

h. Weekly summary: Work progress should be summarized in the log on a weekly basis.

2. Health Physics Log. A health physics log will be maintained recording the chronological decommissioning events. The health physicist on duty will be responsible for maintaining the log. As a minimum the following will be recorded:

a. Remarks pertaining to ~~daily~~ monitoring of decommissioning tasks to include any special monitoring performed.

b. The number and weight of radioactive waste containers filled

c. Status of radioactive waste in storage awaiting shipment

d. Amount (gallons and activity) of liquid waste discharge (to include hold-up tanks)

e. Amount and average concentration of airborne activity discharged.

f. Movement of all highly radioactive equipment, components of structures.

g. All incidents pertaining to radiological safety

h. Facts bearing on any problem areas

i. Issuance and termination of all radiation work permits (RWP)

j. Shipment of radioactive samples (quantity and description) and analysis results when received from Health Physics or a contractor.

3. Quality Control and Assurance. Quality control and quality assurance inspections and tests, where applicable, will be the responsibility of the Physcisit-in-Charge. Results of inspections, quality control and quality assurance observations will be kept in the decommissioning log.

4. Disposition. After completion of the decommissioning task, all logs will be forwarded to the HDL Technical & Administration Support Office for disposition.

B. Reports

1. Final Inspection Report. The reactor Commander (Chief Branch 290) shall be responsible for preparing a report on the final inspection of the decommissioned DORF.

2. Final Decommissioning Report. The Reactor Commander (Chief Branch 290) will be responsible for the preparation of the final decommissioning report. This report will contain all the aspects of the decommissioning tasks and contain the safety analysis of the facility in its ultimate status. The report will be forwarded through Commander, DARCOM (ATTN: DRCSF-P) to Chief of Engineers (ATTN: DAEN-SON).

3. Monthly Decommissioning Progress Report. A monthly decommissioning progress report will be prepared by the Physicist-in-Charge and forwarded to the Commander DARCOM (ATTN: CRCSF-P) by the 15th day of the month immediately following the reporting period.

C. Records

1. Fuel Accountability. The DORF fuel elements (Special Nuclear Material) accountability will be transferred from the US Army, Harry Diamond Laboratories to ERDA. The fuel elements will be sent to the Idaho Reprocessing Plant or to other TRIGA reactor plants as determined by the University Relations Division, ERDA.

2. Files and Records. After completion of decommissioning, all DORF operations logs and records will be transferred to HDL Technical and Administration Support Office for retention or final disposition in accordance with AMC Supplement to AR 385-80.

VIII EMERGENCY PLANS

A. DORF Emergency Plan

The Emergency Plan required by the Technical Specifications will be in effect through the Decommissioning operations. The purpose of the DORF Emergency Plan is to establish the procedures to be followed in the event of an accident or emergency at the reactor facility. Accordingly, these procedures are to insure the protection of the health and safety of personnel under accident or emergency conditions. The objective of the Emergency Plan is to provide complete and coherent procedures for dealing with emergency and accident situations. The potential emergency conditions which may arise at the DORF are classified into two categories:

(1) The Design Basis Accident and, (2) varying degrees of lesser accidents. After the SNM fuel has left the facility site, the Design Basis Accident considerations would no longer be in effect.

B. OTHER PLANS AND SOPS

The following plans and standard operating procedures will be in effect during decommissioning where applicable:

1. Technical Specifications for the DORF, 13 Aug 1973

2. Health Physics Plan for the DORF Reactor, 18 March 1975

3. DORF ENRADMON PLAN, 1 Feb 1977

4. Operations Procedures for DORF, HDL Pamphlet 70-5

5. DORF Emergency Plan 1973

6. Physical Security Plan, DORF, 25 Sept 1975

7. SOP No. 4 Area Radiation Monitoring System

8. SOP No. 6 Procedures for Measuring Fuel Elements and Fuel Following Control Rods

9. SOP No. 8 Criteria for Replacement of Pre and Absolute Filters in Exhaust System

10. SOP No. 12 Procedures for Special Nuclear Material (SNM) Inventory

11. SOP No. 13 Personnel Access Control

12. WRAMC Health Physics Regulations, 40-10

IX CONTRACTING

A. Nuclear Fuel Shipment

The shipment of the DORF Special Nuclear Material (SNM) is governed by AR 55-5, Department of Transportation Regulation and Nuclear Regulatory Commission Regulation 10CFR71. U.S. Energy Research and Development Administration has been asked to contract for the shipping cask and for the shipment of the fuel elements to the reprocessing plant in Idaho. A licensed Battelle, Columbus BM-1 shipping cask for the standard TRIGA fuel elements is available in November 1977, or if this date cannot be met early 1978. A different cask will be used for the fuel followed control rods and instrumented fuel elements because of their longer length. Some of the fuel elements may be shipped to other TRIGA reactor facilities.

B. Solid Radioactive Waste Shipments

1. Control Rods. The control rods will be shipped with the solid radioactive waste. The contact dose rates from these rods is not expected to exceed 100 mr/hr.

2. Solid Radioactive Waste. The shipment of solid radioactive waste will be contracted to a commercial carrier. Disposal will be in accordance with AR 755-15 at an NRC licensed burial site. It is expected that solid radioactive waste will be low level.

X. DECOMMISSIONING TASKS

A listing of the decommissioning tasks is tabulated in Appendix 2.

APPENDIX 1

NRC REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

June 1974

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

NRC

REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

A. INTRODUCTION

Section 50.51, "Duration of license, renewal," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each license to operate a production and utilization facility be issued for a specified duration. Upon expiration of the specified period, the license may be either renewed or terminated by the Commission. Section 50.82, "Applications for termination of licenses," specifies the requirements that must be satisfied to terminate an operating license, including the requirement that the dismantlement of the facility and disposal of the component parts not be inimical to the common defense and security or to the health and safety of the public. This guide describes methods and procedures considered acceptable by the Regulatory staff for the termination of operating licenses for nuclear reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

When a licensee decides to terminate his nuclear reactor operating license, he may, as a first step in the process, request that his operating license be amended to restrict him to possess but not operate the facility. The advantage to the licensee of converting to such a possession-only license is reduced surveillance requirements in that periodic surveillance of equipment important to the safety of reactor operation is no longer required. Once this possession-only license is issued, reactor operation is not permitted. Other activities related to cessation of operations such as unloading fuel from the reactor and placing it in storage (either onsite of offsite) may be continued.

A licensee having a possession-only license must retain, with the Part 50 license, authorization for special nuclear material (10 CFR Part 70, "Special Nuclear Material"), byproduct material (10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material"), and source material (10 CFR Part 40, "Licensing of Source Material"), until the fuel, radioactive components, and sources are removed from the facility. Appropriate administrative controls and facility requirements are imposed by the Part 50 license and the technical specifications to assure that proper surveillance is performed and that the reactor facility is maintained in a safe condition and not operated.

A possession-only license permits various options and procedures for decommissioning, such as mothballing, entombment, or dismantling. The requirements imposed depend on the option selected.

Section 50.82 provides that the licensee may dismantle and dispose of the component parts of a nuclear reactor in accordance with existing regulations. For research reactors and critical facilities, this has usually meant the disassembly of a reactor and its shipment offsite, sometimes to another appropriately licensed organization for further use. The site from which a reactor has been removed must be decontaminated, as necessary, and inspected by the Commission to determine whether unrestricted access can be approved. In the case of nuclear power reactors, dismantling has usually been accomplished by shipping fuel offsite, making the reactor inoperable, and disposing of some of the radioactive components.

Radioactive components may be either shipped offsite for burial at an authorized burial ground or secured and/or shielding material. Means such as a remote-readout intrusion alarm system should be provided to indicate to designated personnel when a physical barrier is penetrated. Security personnel that provide access control to the facility may be used instead of the physical barriers and the intrusion alarm systems.

b. The physical barriers to unauthorized entrance into the facility, e.g., fences, buildings, welded doors, and access openings, should be inspected at least quarterly to assure that these barriers have not deteriorated and that locks and locking apparatus are intact.

c. A facility radiation survey should be performed at least quarterly to verify that no radioactive material is escaping or being transported through the containment barriers in the facility. Sampling should be done along the most probable path by which radioactive material such as that stored in the inner containment regions could be transported to the outer regions of the facility and ultimately to the environs.

d. An environmental radiation survey should be performed at least semiannually to verify that no signficant amounts of radiation have been released to the environment from the facility. Samples such as soil, vegetation, and water should be taken at locations for which statistical data has been established during reactor operations.

e. A site representative should be designated to be responsible for controlling authorized access into and movement within the facility.

f. Administrative procedures should be established for the notification and reporting of abnormal occurrences such as (1) the entrance of an unauthorized person or persons into the facility and (2) a significant change in the radiation or contamination levels in the facility or the offsite environment.

g. The following reports should be made:

(1) An annual report to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, describing the results of the environmental and facility radiation surveys, the status of the facility, and an evaluation of the performance of security and surveillance measures.

(2) An abnormal occurrence report to the Regulatory Operations Regional Office by telephone within 24 hours of discovery of an abnormal occurrence. The abnormal occurrence will also be reported in the annual report described in the preceding item.

h. Records or logs relative to the following items should be kept and retained until the license is terminated, after which they may be stored with other plant records:

(1) Environmental surveys,

(2) Facility radiation surveys,

(3) Inspections of the physical barriers, and

(4) Abnormal occurrences.

4. DECONTAMINATION FOR RELEASE FOR UNRESTRICTED USE

If it is desired to terminate a license and to eliminate any further surveillance requirements, the facility should be sufficiently decontaminated to prevent risk to the public health and safety. After the decontamination is satisfactorily accomplished and the site inspected by the Commission, the Commission may authorize the license to be terminated and the facility abandoned or released for unrestricted use. The licensee should perform the decontamination using the following guidelines:

a. The licensee should make a reasonable effort to eliminate residual contamination.

b. No covering should be applied to radioactive surfaces of equipment or structures by paint, plating, or other covering material until it is known that contamination levels (determined by a survey and documented) are below the limits specified in Table I. In addition, a reasonable effort should be made (and documented) to further minimize contamination prior to any such covering.

c. The radioactivity of the interior surfaces of pipes, drain lines, or ductwork should be determined by making measurements at all traps and other appropriate access points, provided contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement should be assumed to be contaminated in excess of the permissable radiation limits.

d. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated in excess of the limits specified. This may include, but is not limited to, special circumstances such as the transfer of premises to another licensed organization that will continue to work with radioactive materials. Requests for such authorization should provide:

(1) Detailed, specific information describing the premises, equipment, scrap, and radioactive contaminants and the nature, extent, and degree of residual surface contamination.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/100 cm^2

[a]Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

Appendix 2

Decommissioning Tasks

The DORF decommissioning tasks are divided into three phases: (1) Reactor fuel removal and shipment, (2) Facility decommissioning, and (3) Preparation of building for alternate use and Post-decommissioning tasks.

Phase I Tasks: Reactor Fuel Removal and Shipment

1. Prepare decommissioning Plan.
2. Obtain ERDA agreement to ship and dispose reactor fuel.
3. Prepare fuel handling Safety Analysis Report (FHSAR).
4. Obtain ARCHS approval of FHSAR.
5. Negotiate rental of licensed shipping casks (either HDL or ERDA).
6. Prepare Appendix A data, "Description of Specification Material and Designation of Processing Batch Size" and "Idaho Chemical Processing Plant Fuel Receipt Criteria", and send to ERDA, Idaho.
7. Remove fuel elements, prepare for shipment.
8. Transportation of fuel elements to ERDA, Idaho or to TRIGA reactor facilities.

Phase II Tasks: Facility Decommissioning

1. Prepare dismantlement plan.
2. Obtain ARCHS approval of dismantlement plan.
3. Establish acceptable surface contamination levels for release of premises for unrestricted use.
4. Prepare specifications for dismantlement contract.
5. Award dismantlement contract.
6. Radioactive equipment and structure removal:

   a. Removal of core support structure
   b. Removal of pool lead shield doors
   c. Pool water discharge to sanitary sewer. (water analysis to assure within 10CFR20 limits)
   d. Removal of aluminum tank liner and activated concrete under the liner
   e. Removal of wood timber lining and concrete in exposure room and plug-door
   f. Removal of lead shield hoist and curtain in exposure room
   g. Activation analysis/dose rate measurements at bottom of pool and in the exposure room to assure acceptable residual contamination levels.
   h. Packaging of radioactive material for shipment to disposal site
   i. Removal of water treatment system and activated sections of exposure room air-conditioning ducts.

7. Transfer reactor console instrumentation and other ancillary equipment to the Armed Forces Radiological Research Institute (AFRRI).

8. Remove jib crane hoist.

9. Remove machine shop equipment and 100 kW emergency generator (to be surplused or transferred to other users).

10. Prepare procedures or discriptions of method for accomplishing the dismantling tasks where applicable.

11. Description of the biological shield and estimates of the content and extent of induced radioactivity.

12. Contamination control:

(1) Dust collection and absolute filtering procedures.
(2) Respirator and protective clothing requirements.
(3) Air Sampling Procedures.
(4) Other Health Physics requirements or procedures.

13. Removal of three 5000 gallon water hold-up tanks.

Phase III <u>Preparation of Building for Alternate Use</u>

1. Post decommissioning radioactivity survey prior to covering/ resurfacing former activated areas.
2. Preparation of building for alternate use:

NO
yes

a. Dismantle concrete parapet to floor level.
b. Fill-in pool hole and resurface to bay floor level.
c. Fill-in lead-shield hoist hole in exposure room and restore floor to level of entrance.
d. Remodel and reactivate air-conditioning system.
e. Remodel electrical distribution and sewer system where applicable.
f. Remove office and laboratory trailer.
g. Remove all other HDL equipment and furniture.
h. Reconnection of sewer lines at hold-up tank area, fill hole and relandscape.

3. Post decommissioning inspection by Independent Agency.

4. Post decommissioning I.G. Inspection and final decommissioning report.

Appendix III

TRIGA Fuel Shipping Cask Analysis

The shipping casks to be used to ship the DORF fuel are the Battelle Memorial Institute BMI-1 cask and another cask not yet determined. The BMI-1 cask will be used to ship the standard length DORF fuel elements. Another cask must be used for the fuel follower control rods (FFCR) and the thermocouple instrumented fuel elements (TCFE) because of the longer overall length.

This appendix contains the BMI-1 certificate of compliance and the cask safety analysis as follows:

1. Certificate of Compliance No. 5957, Revision 1, of the BMI-1 shipping package.

2. Safety Analysis Report for shipment of the TRIGA fuel by the University of Arizona.

3. Supplement No. 1 to request for license to transport irradiated TRIGA fuel in BMI-1 Shipping Cask, June 1972.

4. University of Arizona TRIGA loading-to-critical experiment.

5. Summary of initial Criticality Experiment for Torrey Pines TRIGA Mark III Reactor Startup.

6. Supplement No.1 to request for license to transport irradiated TRIGA fuel in BMI-1 Shipping Cask, June 1973 (replacement pages 12 through 20 to item 3).

7. Analysis of Shells of Revolution Subjected to Symmetrical and Nonsymmetrical Loads.

8. DORF reactor start-up.

APPENDIX IV

RADIO-ISOTOPIC ANALYSIS OF RADIOACTIVE MATERIAL IN THE DORF STRUCTURE BEFORE DECOMMISSIONING

INTRODUCTION

This report documents information of the amount and type of radioactive material that will be present in the structure and building of the Diamond Ordnance Radiation Facility after removal of the reactor fuel in the spring of 1978. Such information is required for decommissioning plans and must be supplied to the Army Reactor Committee for Health and Safety (ARCHS) prior to their approval of such plans. The information is also needed by the waste-disposal area directorate who must budget for specific volumes and radioactive levels. Finally, the isotopic composition of the radioactive waste is necessary for labeling containers at the time of shipment.

The first section of this report is a summary for those who need only the final results on type, location and amount of residual radioactivity. Section two describes the investigative procedures, discusses the possible sources of radioactivity and the properties of the radioactive isotopes found. Graphs of isotopic analyses and calculations, which convert detector response to specific activities, are included in this section. The second section also provides the detailed calculations of volumes, weights and total radioactivity in the various sections of DORF. The final section contains recommendations based on things discovered during this study.

SUMMARY

The radioactivity that will remain at DORF after the fuel removal in the spring of 1978 has been carefully estimated based on criteria, measurements and necessary assumptions documented in this report. A concise summary of that radioactivity is given in Table I. The most predominant radioactive isotopes in the concrete are cobalt-60 and europium-152 and -154. The most predominant isotopes in lead are antimony-124 and silver-110 . The wood and steel (mainly in the lead-shield hoist) are not very radioactive and are easy to dispose of. The aluminum itself is almost non-radioactive but there is a radioactive Phenoline

liner which tends to stick to the aluminum. Its radioactivity comes from cobalt-60 and zinc-65. All of these radioactive isotopes have half-lives in excess of 60 days.

TABLE I. Summary[1/] of total radioactivity to be expected from materials in the DORF structure after core removal.

Material	Mass (lbs)	Volume (ft^3)	Radioactivity (millicuries)
CONCRETE	82,170	412	36.24
(If whole plug door included).	(170,050)	(850)	(36.24)
LEAD	55,753	112	13.34
ALUMINUM	2,288	15	75.71
WOOD	34,944	1344	0.33
STEEL	2,662	5.5	0.03
GRAND TOTAL	177,817 lbs (89 tons)	1889 ft^3	0.126 Curies

[1/] This represents a summation of the values given in Table X.

IDENTIFICATION OF THE RADIOACTIVITY

Isotope Identification:

The principal method of identification was gamma-radiation spectroscopy with a germanium lithium-drifted detector, or Ge(Li) crystal. The crystal is housed inside a very low-activity-lead cave lined with wood. Numerous background analyses confirm that for photons with energies greater than 140 keV, samples with low activities (two to three times background) can be successfully analyzed for specific photon energies. A plot of a multichannel analyzer spectrum of the background is given in Fig. 1. The principal higher energy peaks in the background spectrum are the 511-keV gammas associated with annihilation radiation and the 1461-keV peak from ^{40}K, a radioactive isotope which is found naturally in almost all "non-radioactive" materials.

The method of analyses provides for very good resolution of the photon energies in the range 140 keV to 2500 keV at approximately 9.7 keV per channel of 256 total channels. The electronic equipment is sufficiently stable over counting periods of 50,000 seconds to permit energy assignment within two percent. Graphs of the gamma spectra of the various materials investigated are shown. (See Fig. 2 through 5).

The method does not provide for the ultimate in accuracy for determining specific activity. The crystal efficiency (disintegrations per count as a function of energy) can only be accurately assigned for a well-defined geometry. The samples in the present situation varied in size and shape. Therefore, they were suspended above the crystal so that their centers of mass were approximately three centimeters from the active volume of the detector and efficiencies were determined with calibrated point sources. The error associated with this procedure is estimated to be no greater than 50%, based on a volume integration of point-source response at points in space representative of the sample size. For the task at hand such accuracy is sufficient.

Rational of Sample Selection

The job was to identify the radioactive content and quantity of materials that will have to be removed from the DORF site so that it can be certified, by post-decommissioning radioactive survey, as an unrestricted area for possible public use. This survey, to be conducted by the Army Environmental Health Agency (AEHA), must be accomplished prior to any filling, sealing or burying activities. This presented two problems. How can we identify the radioactivity in presently inaccessible areas, such as below the reactor pool, before the reactor fuel and higher-level radioactive structures have been removed? What amount of material will have to be removed from walls and floors to reach an acceptable AEHA level?

The first problem was attacked as follows. Representative samples of all the material types are accessible in the exposure-room area. Because of the significantly larger thermal-neutron cross sections of materials and the fact that the DORF-TRIGA reactor is zirconium-hydride moderated and water-cooled reactor, the thermal component of the spectrum is the dominate source of induced radioactivity. As will be discussed later, the predominance of radioactive europium confirms this. Therefore, isotopic analyzes of exposure room samples are representative of those in presently inaccessible areas. Furthermore, with facility dosimetry data for the various locations, we can estimate the residual radioactivity in remote locations with significantly different flux exposure levels.

The second problem of how much material to remove is more complex because we do not have good guidance on the amount of radioactivity in volume that can remain. NRC Regulation 1.86, the current guide, clearly specifies levels for removal surface contamination but is, at best, vague on volume activity and how to detect it. The criteria set for the analysis in this report are as follows:

(1) Once the reactor support structure has been removed there will be no high-level radioactive waste remaining in the DORF structure. Our analyses confirm this.

(2) Based on existing allowable concentrations of radioactive materials in water and a specific activity proportional to material density, we can set an allowable specific activity of 2×10^{-5} microcuries per gram as the maximum permissible concentration of radionuclides in water when it is known that Sr 90, I 129, (I 125, I 126, I 131, Table II only), Pb 210, Ra 226, Ra 228, Cm 248, and Cf 254 are not present.* Since the density of water is one g/cm^3 and there are 28317 cm^3/ft^3, 2×10^{-5} $\mu Ci/g$ corresponds to 0.57 $\mu Ci/ft^3$ of water.

(3) It is assumed that the radioactivity is distributed in the material to be removed in proportion to the incident thermal fluence (flux-time product) and attenuated exponentially according to thermal-neutron relaxation lengths, (i.e., the inverse of microscopic removal cross sections for broad beams). Half-life decay is taken into consideration for the period until spring 1978. Therefore, the depth of material to be removed, D in centimeters, is determined

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* 10 CFR20, note to Appendix B

by relative fluence level at the surface, ϕ/ϕ_o, and

$$D = L \ln \frac{\phi/\phi_o \times A}{0.57\mu Ci/ft^3} \quad (1)$$

where A is the activity in $\mu Ci/ft^3$ estimated from this study. The values of relaxation length are given in Table II.

TABLE II. Material densities and relaxation lengths, L

Material	Density	Relaxation Length
Concrete	2.35 g/cm^3	1.6 cm
Lead	11.0 g/cm^3	4.2 cm
Wood	0.42 g/cm^3	2.9 cm

Measured Radioactivity

Samples taken from the DORF exposure room were concrete, wood, aluminum, lead and a tar-paper-like liner installed between the aluminum pool tank and the concrete pool base. Although the aluminum itself has very little residual radioactivity (less than $8 \times 10^{-6} \mu Ci/gm$ for the sections counted), the Phenoline paper (i.e., the tar-paper liner) has the highest specific activity of all the materials examined. Since this liner tends to stick to the aluminum, for all practical purposes the aluminum tank exhibits this activity.

Tables IV through IX give a breakdown of the isotopic composition of the radioactivity in the various samples. Tables IV and V are composed of several additional pages that serve as detailed examples of the methods of analyses and are self explanatory when reference is made to the graphs of the multichannel-analyzer output. Figs. 1 through 5 are the multichannel-analyzer gamma spectra for the various types of samples. The energy of the photopeaks is related to

TABLE III. GAMMA* SPECIFIC ACTIVITY AND THE NUMBER OF MICROCURIES PER UNIT OF MATERIAL FOR VARIOUS RADIOACTIVE MATERIALS FROM THE DORF EXPOSURE ROOM

	Type of Material	Location in Exposure Room	Specific Activity (d/s·g)	Activity per unit of material
1.	PHENOLINE PAPER	On aluminum tank near exposure room end of pool	636	3.6 μCi/sq ft
2.	CONCRETE	From front part of room about 4 feet from reactor	78	140 $\mu Ci/ft^3$
3.	CONCRETE	Very near reactor at exposure room end of tank	141	252 $\mu Ci/ft^3$
4.	LEAD	From curtain above the movable lead shield	72	0.62 μCi/lb
5.	LEAD	From brick in middle of the movable lead shield	205	1.30 μCi/lb
6.	WOOD	From very near reactor and concrete sample #3, above.	1.3	0.40 $\mu Ci/ft^3$

*From gross beta plus gamma analyses, the beta-to-gamma activity of all these different materials is approximately 1.8.

spectra for the various types of samples. The energy of the photopeaks is related to start the channel number (abcissa) by the following equation:

$$E(\text{keV}) = (\text{channel} + 2.5) \times 9.69 \quad \pm 2\% \tag{2}$$

For clarification, the gamma-ray peaks are indentified by isotope and their energies in keV (and in parenthsis) are given for most of the peaks.

The specific activity (d/s·g) for each measured sample is compared in Table III. This table also provides the number of microcuries per unit most practical for that type of material. This latter information is used in Table X to determine the total radioactivity in the volumes of radioactive materials at DORF.

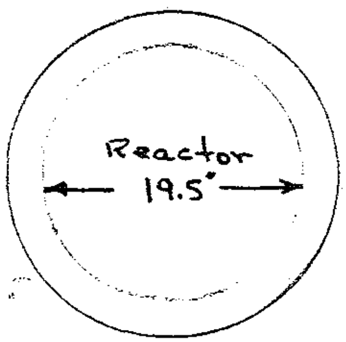
CONCRETE EXCAVATION

The maximum observed concrete-volume activity was 252 μCi/ft³. If we assume that the maximum fluence at any location is 100 times larger than at this monitored exposure room location, Eq. (1) ... the excavation-depth formula ... requires

$$D = 1.6 \ln\left(\frac{100 \times 252}{0.57}\right) = 17.1 \text{ cm or } 6.7 \text{ inches}$$

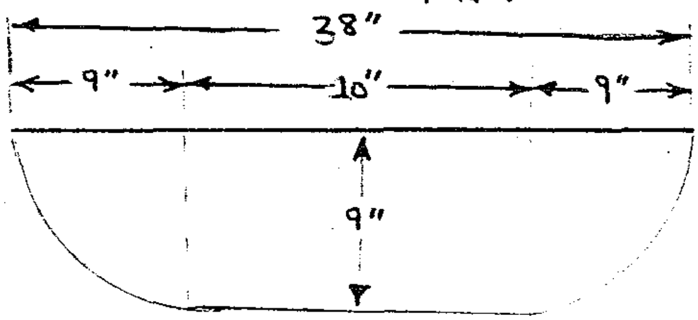
of course, fast neutrons will also penetrate and thermalize but the factor of 100 is already very conservative so excavation depths of 9 to 10 inches in the immediate vicinity of the reactor and at the end positions of the pool should adequately remove the radioactivity. It is assumed that because of H_2O attenuation that the flux will be decreased by a factor of 400 at the 38 inch extremes.

Excavation below reactor in Pos #3 & radiography position:



Excavation diameter

38"



Volume: $\pi r_1^2 l + \frac{1}{4}(2\pi^2 R r_2^2)$

$\pi[(10^2)9 + \frac{1}{2}(\pi(14.5)(9^2))]$

$V = (2827 + 5795) \div 1728 = \mathbf{2.52\ ft^3}$

<u>Activity in concrete below Pos. 3</u>: Activity will not exceed measured maximum of 252 μCi/ft³ by more than a factor of two. Assign 500 μCi/ft³.

$$5.0 \times 500 = 2500\ \mu Ci$$

<u>Activity in concrete below radiography position:</u>

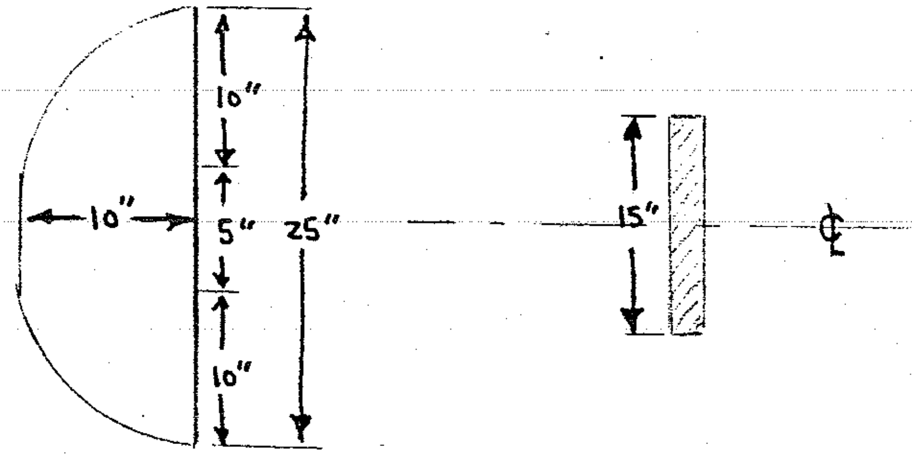
Assign 250 μCi/ft³ (conservative)

$$5.00 \times 250 = 1250\ \mu Ci$$

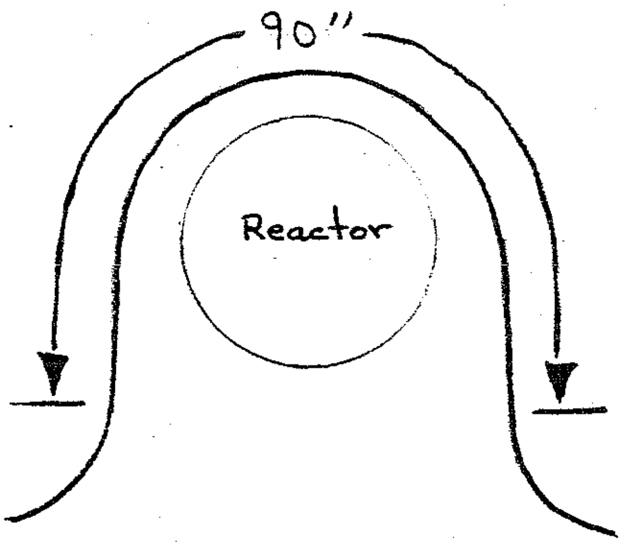
Excavation from tank-support walls in Pos. #1 & #3:

(active fuel length is 15 inches)

FOR 1&3:



Volume for Pos. 3 end of pool:



Volume $L \times (\frac{1}{2}\pi r^2 + t \times d)$

$90\left(\frac{\pi}{2}(10^2) + 5 \times 10\right)$

$V = 90(157 + 50) \div 1728 = \mathbf{10.8\ ft^3}$

Volume for Pos. 1 end of pool: Because tank projects into the exposure room, L for pos. 1 in taken as 40" rather than 90" and $V = \frac{4}{9}(10.8) = \mathbf{4.8\ ft^3}$

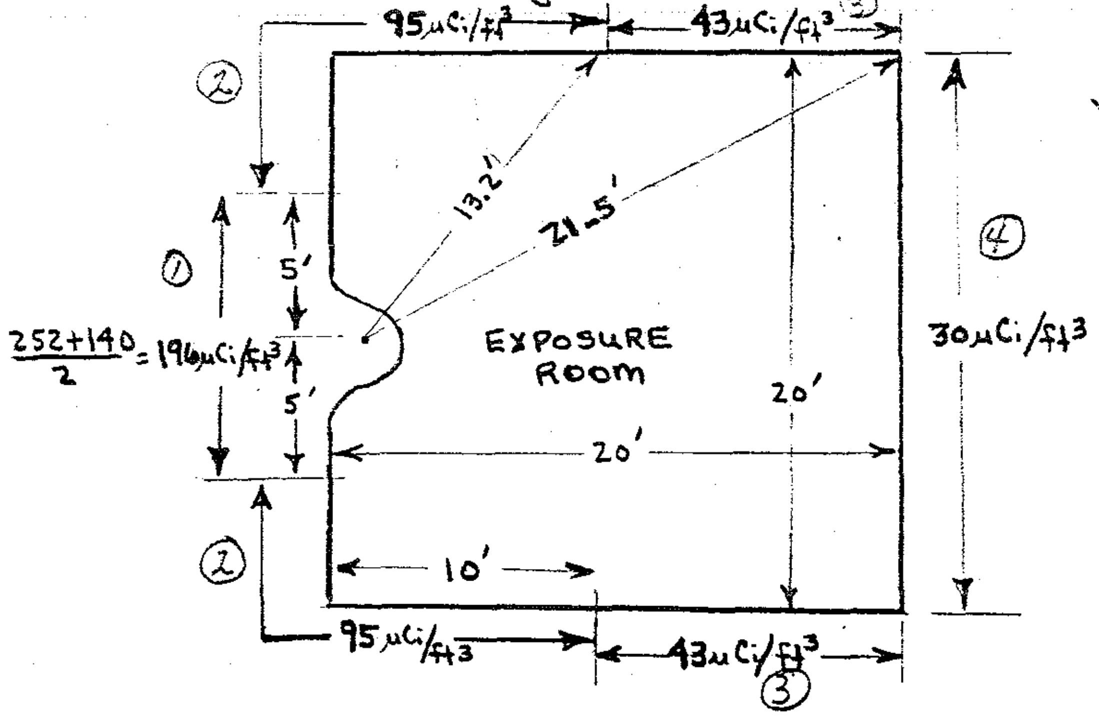
Activity for wall excavation in Pos. 3 & 1: Consistant with conservative estimate for concrete below reactor in Pos. 3, assign $500\,\mu Ci/ft^3$

Activity for Pos. 3 end: $500 \times 10.8 = 5400\,\mu Ci$

Activity for Pos. 1 end: $500 \times 4.8 = 2400\,\mu Ci$

Excavation of concrete from walls, floor and ceiling of the DORF exposure room.

The following diagram shows the estimated volume activity in the concrete of the exposure room walls. Because of thermalization of fast flux by the wood and scatter from the walls, the effective thermal flux is assumed to fall off more nearly as 1/R than as $1/R^2$.



room heighth 8 feet.

Excavation depths:

No.	D (inches)	Assign Activity/ft³
1	3.7 "	196 μCi/ft³
2	3.2	95 "
3	2.7	43 "
4	2.5	30 "

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IV. CONCRETE SAMPLE FROM "FRONT" OF EXPOSURE ROOM

$M = 2.47\,g$ $T = 54,252$ seconds ($MT = 1.34 \times 10^5$)

ISOTOPE	Half Life $T_{1/2}$	Activity dis/s.g.	% of Total Activity
^{60}Co	5.27y	31.66	40.55 %
^{152}Eu - ^{154}Eu	12.2y – 16y	30.15	38.62 "
^{46}Sc	84 days	5.44	6.97 "
^{134}Cs	2.07 y.	5.06	6.48 "
^{65}Zn	245 d	2.29	2.93 "
^{182}Ta	115 d	1.19	1.52 "
^{124}Sb	60 d	0.63	0.81 "
* Annihilation	—	1.65	2.12 "
	TOTAL:	78.07 d/s.g	(100.00)

Nat. Bkg in concrete
(^{226}Ra series)
^{214}Bi 1622y 0.84 d/s.g.
(i.e., concrete is 93 times its own natural background)

Activity per cubic foot of concrete

$(12 \times 2.54)^3 = 28317\ cm^3/ft^3$ } $6.65 \times 10^4\ g/ft^3$
ρ concrete $= 2.35\ g/cm^3$

$$\text{Activity} = \frac{6.65 \times 10^4 \times 78.07}{3.7 \times 10^{10}\ d/s \cdot Ci} = 1.40 \times 10^{-4}\ Ci \text{ or } \boxed{140 \mu Ci/ft^3}$$

Estimate per "ml" of H_2O
$(78.07)/3.7 \times 10^{10} \times 2.35 = 8.98 \times 10^{-10}$ or $8.98 \times 10^{-4}\ \mu Ci/$[illegible]

* Note: All activities except annihilation radiation (511 kev) were determined from sample with background substracted.

BLE IV-A CONCRETE SAMPLE FROM FRONT OF EXPOSURE ROOM

WEIGHT = 2.47 g (BKG. SUBSTRACTED FOR EQUAL TIME: 54,252 seconds)

* Indicated more that one isotope has a peak in this channel   9.71±0.05 kev/ch+2.5

[illegible] E & $T_{1/2}$	keV ENERGY	CHANNEL	PEAK GROSS COUNTS	COMPTON BKG COUNTS	PEAK NET COUNTS	e c/d	Total d	
Eu – ^{154}Eu	$122^{152,154}$	10	23500	3364	20136	2.56×10^{-2}	305440	
2y – 16year)	$245 \& 248^{154}$	23	4889	2030	2859	1.16×10^{-2}	246465	
	344^{152}	33	8441	1756	6685	8.10×10^{-3}	825308	
	$411^{154,152}$	40	1894	1370	524	6.65×10^{-3}	78797	
	488^{152}	47.8	1658	1340	318	5.58×10^{-3}	56989	
	689^{154}	68.8	1366	1230	136	3.90×10^{-3}	34872	
	$720 \& 723.1^{154}$	72 *	1357	1240	117	3.63×10^{-3}	32231	
	779^{152}	78	2562	1250	1313	3.35×10^{-3}	391940	
	$867 \& 873^{154}$	87	1788	1325	463	2.95×10^{-3}	156949	
	966^{152}	97	2457	1150	1307	2.66×10^{-3}	491353	
	1006^{154}	101	1130	940	190	2.57×10^{-3}	73930	
	~~1092^{152}~~	~~110~~	~~1728~~	~~975~~	~~753~~	~~2.45×10^{-3}~~	~~307347~~	delete because of Zir
	1277^{154}	129	321	135	186	1.95×10^{-3}	95385	
	1412^{152}	143	1220	50	1170	1.74×10^{-3}	672413	
	1457^{152}	146.7	73	32	41	1.67×10^{-3}	24551	
	1528^{152}	155	59	18	41	1.60×10^{-3}	25625	
	1595^{154}	162.5	60	17	43	1.52×10^{-3}	28289	
						TOTAL =	4.048×10^{6}	
^{60}Co (5.27 years)	1173.1	119	4857	589	4268	2.11×10^{-3}	2.023×10^{6}	
	1332.4	135	4187	126	4061	1.86×10^{-3}	2.219×10^{6}	
						TOTAL =	4.242×10^{6}	
^{134}Cs (2.07 year)	563 & 569	56	1580	1300	280	4.72×10^{-3}	5.932×10^{4}	
	605	60 *	1612	1293	319	4.36×10^{-3}	7.317 "	
	795 & 802	80	2101	1300	801	3.25×10^{-3}	24.646 "	
	1040	105 *	870	840	30	2.50×10^{-3}	1.200 "	
	1168	119 *	(370)	120	250	2.15×10^{-3}	11.628 "	
	1365	139	370	60	310	1.81×10^{-3}	17.127 "	
						TOTAL =	6.785×10^{5}	
Zn (245 day)	1115.4	110 *	1728	975	753	2.45×10^{-3}	307347	
						TOTAL	3.073×10^{5}	

(2 page table)

TABLE IV-A. (continued)

ISOTOPE	keV ENERGY	channel	Gross	Bkgs	Net	ε c/d	Total
^{46}Sc (84 days)	889.4	89 *	2491	1400	1091	2.90×10^{-3}	3.762×10^{5}
	1120.3	113	2330	(1550)	780	2.21×10^{-3}	3.529×10^{5}
						TOTAL	7.291×10^{5}
^{124}Sb (60 day)	602.6	60 *			200	4.40×10^{-3}	45.454×10^{3}
	646	65 *			25	4.17×10^{-3}	5.995 "
	722.8	72 *			25	3.63×10^{-3}	6.887 "
	968	97 *	1120	1100	20	2.66×10^{-3}	7.518 "
	1045	105 *	870	853	17	2.50×10^{-3}	6.800 "
	1691	172	41	28	13	1.43×10^{-3}	9.091 "
	2088	207	25	22	3	1.15×10^{-3}	2.609 "
						TOTAL =	8.435×10^{4}
^{182}Ta (115 day)	100.3	8	4577	4102	475	3.10×10^{-2}	15.322×10^{3}
	152.4 & 156	13.6	3054	2700	354	1.42×10^{-2}	24.930 "
	198.4	17.5 *	2620	2450	170	1.50×10^{-2}	11.333 "
	223	20.5	2348	2210	138	1.35×10^{-2}	10.222 "
	1121	113 *	870	810	60	2.21×10^{-3}	27.149 "
	1189	119	(710)	650	60	2.09×10^{-3}	28.708 "
	1221.6	123	337	252	85	2.05×10^{-3}	41.463 "
						TOTAL =	1.591×10^{5}
Annihilation	511	50	4100	3100	1,000	4.50×10^{-3}	2.22×10^{5}
						TOTAL	2.22×10^{5}
Additional analysis of "natural background in concrete: ^{214}Bi (1622 years*) * from ^{226}Ra series							
	1731	177	36	27	9	1.39×10^{-3}	6.475×10^{3}
	1767	180	39	29	10	1.24×10^{-3}	8.064 "
	2119	217	23	14	9	1.13×10^{-3}	7.965 "
	2204	226	32	23	9	1.08×10^{-3}	8.333 "
	2293	234	28	12	16	1.03×10^{-3}	15.534 "
	2447	250	36	25	11	9.40×10^{-4}	11.702 "
		Actual about 2x this →				TOTAL	[illegible] $\times 10^{4}$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE V. TAR-PAPER* SAMPLE FROM FRONT OF EXPOSURE ROOM

* CALLED PHENOLINE COATING

$m = 3.69$ g $\quad T = 20{,}183$ s $\quad (mT = 7.45 \times 10^4)$

ISOTOPE	HALF-LIFE $\tau_{1/2}$	Activity dis/s.g.	% of Total Activity
^{65}Zn	245 day	185.2	29.1 %
^{60}Co	5.27 year	363.8	57.2 %
^{152}Eu	12.2 year	86.8	13.7 %
	TOTAL:	635.8 d/s·g	100.0 %

From gross counts, this sample was $423 \div 11 = 38.5$ times more active than the natural background in the Ge(Li) crystal cave. NOTE: It is $635.8/78.1 = 8.1$ times more radioactive than the concrete from the front part of the room.

Activity per square foot of paper:
(assuming a "1/8-inch thickness")

$(12 \times 2.54)^2 \times 0.125 \times 2.54 = 295$ cm^3 per square foot

$\rho = 0.7$ g/cm^3

$$\text{Activity} \quad \frac{635.8 \text{ d/s}\cdot\text{g} \times 206.5\text{g}}{3.7 \times 10^{10} \text{ d/s}\cdot\text{Ci}} = \boxed{3.55\,\mu\text{Ci/sq. foot}}$$

TABLE V-A. TAR-PAPER SAMPLE FROM POS #1 END OF ALUMINUM TANK (IN EXPOSURE ROOM).

TIME = 20,183 sec.

423 samples/100
11 bkg

M = 3.6875 grams

20183 000004 002000

		1	2	3	4	5	6	7	8	9
000000	020183	000004	065213	016877	013805	014030	014273	013571	013285	013616
000010	048361	011699	011727	011230	011878	010214	010054	009864	009489	009693
000020	009953	009577	009525	013469	008729	008657	008524	008433	008402	008066
000030	008237	008377	007927	019884	007346	007546	007144	006983	006859	006789
000040	007630	006823	006536	007492	006603	006469	006496	006540	006508	006670
000050	011253	006750	006323	006526	006497	006456	007013	006515	006682	006515
000060	007831	006514	006633	006741	006702	006907	006732	006904	007032	006989
000070	006667	006954	007301	007106	007305	007392	007381	007520	009923	007635
000080	18622	008056	008383	008368	008992	008968	009128	010116	009901	012019
00090	10301	009768	008668	008394	008263	008354	008177	009248	006260	005804
000100	005583	005638	004958	004642	004680	004475	004422	004334	004357	004701
000110	005809	004901	009792	037642	002790	002421	002438	002329	007215	028249
000120	001547	001386	001187	001140	000822	000787	000644	000546	000530	000822
000130	000530	000744	000621	000538	000840	027671	000259	000203	000178	000189
000140	000194	000178	000271	002053	000157	000170	000186	000296	000244	000122
000150	000132	000137	000141	000132	000135	000193	000148	000144	000137	000121
000160	000126	000137	000143	000130	000094	000106	000124	000107	000116	000104
000170	000114	000116	000092	000124	000107	000099	000118	000084	000094	000087
000180	000094	000080	000089	000100	000093	000115	000099	000102	000088	000080
000190	000082	000107	000089	000090	000090	000090	000095	000092	000078	000081
000200	000093	000089	000093	000095	000085	000116	000081	000082	000079	000068
000210	000090	000064	000065	000062	000069	000070	000053	000076	000058	000074
000220	0066	000064	000068	000052	000064	000070	000059	000076	000054	000062
0230	00072	000062	000049	000079	000056	000047	000041	000045	000043	000034
000240	000033	000027	000023	000019	000027	000016	000012	000013	000012	000016
000250	000013	000006	000009	000011	000016	000019 16				

TABLE ___

ISOTOPIC ANALYSES OF TAR-PAPER SAMPLE FROM ON TANK WALL AT EXPOSURE-ROOM END AT APPROXIMATELY CORE CENTERLINE.

COUNT TIME = 20,183 sec.; WEIGHT = 3.6875 g

SOURCE ISOTOPE / $T_{1/2}$	ENERGY* (keV)	CHANNEL #	PEAK GROSS COUNTS	COMPTON BKG. COUNTS	PEAK NET COUNTS	XTAL EFF. (c/d)	TOTAL d
^{65}Zn (245 day)	1115.4	113	37642	3700	33942	2.45×10^{-3}	1.38×10^{7}
						TOTAL =	(1.38×10^{7})
^{60}Co (5.27 year)	1173	119	28249	1700	26549	2.11×10^{-3}	1.26×10^{7}
	1332	135	27671	750	26921	1.86×10^{-3}	1.45×10^{7}
						TOTAL =	(2.71×10^{7})
^{152}Eu (12.2 year)	121.8	10	48361	12600	35761	2.50×10^{-2}	1.43×10^{6}
	244.6	23	13469	9140	4329	1.16×10^{-2}	0.30 "
	344.2	33	19884	7700	12184	8.10×10^{-3}	1.50 "
	411.0	40	7690	6790	900	6.65×10^{-3}	0.14 "
	443.9	43	7492	6500	992	6.60×10^{-3}	0.15 "
	778.6	78	9932	7550	2382	3.35×10^{-3}	0.71 "
	964.1	97	9248	7200	2048	2.66×10^{-3}	0.77 "
	1086.0	110	5809	4800	1009	2.45×10^{-3}	0.41 "
	1407.5	143	2053	210	1843	1.74×10^{-3}	1.06 "
						TOTAL =	(6.47×10^{6})

* Energy is approximately (±1.5%) 9.69 ×(CHANNEL+2.4) keV/channel

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VI. CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE ROOM

$M = 11.63\ g$ $T = 40{,}000\ sec$ $(MT = 4.65 \times 10^5)$

ISOTOPE	HALF-LIFE $T_{1/2}$	Activity dis/s·g	% of Total Activity
^{60}Co	5.27 y	63.2	44.5 %
^{152}Eu-^{154}Eu	12.2y – 16y	55.4	39.0 "
^{46}Sc	84 day	10.5	7.4 "
^{65}Zn	245 day	4.0	2.8 "
^{182}Ta	115 day	3.1	2.2 "
^{134}Cs	2.07 y	1.3	0.9 "
^{124}Sb	60 day	0.2	1.2 "
Annihilation	—	2.8	2.0 "
	TOTAL	140.5 d/s·g	(100.0)

Note: This concrete sample is 1.8 times more radioactive than the sample from the front of the room.

Activity per cubic foot of concrete:

Density: $6.65 \times 10^4\ g/ft^3$

$$\text{Activity} = \frac{6.65 \times 10^4 \times 140.5}{3.7 \times 10^{10}\ d/s \cdot Ci} = 2.52 \times 10^{-4}\ Ci \text{ or } \boxed{252\,\mu Ci/ft^3}$$

Estimate per "ml" of H_2O: $1.62 \times 10^{-3}\ \mu Ci/g$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VII. LEAD FROM BRICK IN MOVABLE LEAD SHIELD

M = 2.531 g T = 40,000 sec (MT = 1.01×10^5)

ISOTOPE	HALF-LIFE ($T_{1/2}$)	ACTIVITY d/s·g	% of TOTAL ACTIVITY
^{124}Sb	60 days	181.8	71.0%
^{110m}Ag	253 day	67.3	26.3%
^{152}Eu-^{154}Eu	12.2y - 16y	1.5	0.6%
Annihilation	——	5.4	2.1%
	TOTAL	256.0 d/s·g	(100.0%)

Activity per pound of lead:
454 g/lb

$$\text{Activity} = \frac{454 \times 256}{3.7 \times 10^{10}} = 3.14 \times 10^{-6} \text{ or } 3.14\ \mu Ci/lb$$

NOTE: About 44% of this radioactivity will decay by spring 1978 because of the ^{124}Sb contribution. Therefore Eff. Activity = 1.76 μCi/lb.

Futhermore, the top and bottom quarter sections of the lead shield should have activities more like that of the lead curtain (Eff. Act = 0.62 μCi/lb) so a better value for the whole

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE VIII. LEAD FROM CURTAIN

M = 0.602 T = 40,000 sec ($MT = 2.41 \times 10^4$)

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY (d/s·g)	% of TOTAL ACTIVITY
^{124}Sb	60 days	4.96	62.1%
^{110m}Ag	253 days	2.28	28.5%
^{46}Sc	84 days	0.45	5.6%
$^{152}Eu-^{154}Eu$	12.2y-16y	0.05	0.7%
Annihilation	——	0.25	3.1%
		7.99 d/s·g	(100.0)

Activity per pound of lead:
(454 g/lb)

$$\text{Activity}: \frac{454 \times 7.99}{3.7 \times 10^{10}} = 0.98 \times 10^{-7} = 1.0\,\mu Ci/lb$$

NOTE: About 38% of this activity will have decayed by spring 1978.
Therefore: Eff Activity = $\boxed{0.62\,\mu Ci/lb}$

GAMMA SPECIFIC ACTIVITY ANALYSES

TABLE IX. WOOD FROM FRONT OF EXPOSURE ROOM

$M = 36.58$ g $T = 8967$ sec. ($MT = 3.28 \times 10^5$)

ISOTOPE	HALF-LIFE $T_{1/2}$	ACTIVITY d/s.g	% of TOTAL ACTIVITY
^{152}Eu-^{154}Eu	12.2y-16y	0.588	46.7 %
^{60}Co	5.27 y	0.534	42.4 %
^{46}Sc	84 day	0.055	4.4 "
^{129m}Te	41 day	0.040	3.2 "
^{127m}Te	110 day	0.026	2.1 "
Annihilation	——	0.017	1.2 "
	TOTAL	1.26 d/s.g	(100.0)

Activity per cubic foot of wood

$$28317 \text{ cm}^3/\text{ft}^3 \times 0.42 \text{ g/cm}^3 = 1.189 \times 10^4 \text{ g/ft}^3$$

$$\text{Activity}: \frac{1.26 \times 1.189 \times 10^4}{3.7 \times 10^{10}} = 4.04 \times 10^{-7} \text{ or } \boxed{0.40 \mu Ci/ft^3}$$

Less than 10% of this activity will decay by spring 1978.

NOTE: From WRAMC Health Physics survey of this sample we find there is a β/γ ratio of approximately two.

TABLE X. Summary of radioactive material types, locations, weights, volumes and activity levels at DORF after reactor-core removal.

MATERIAL	LOCATION AT DORF	DENSITY (lbs/ft³)	WEIGHT (lbs)	VOLUME (ft³)	ACTIVITY LEVEL	TOTAL ACTIVITY OF VOLUME (millicuries)
1. CONCRETE	PLUG DOOR TO EXPOSE ROOM	200	87,880	439	————	Negligible
	• (Front 12 inches of door)	"	9,750	49	140 μCi/ft³	6.86
2. "	• WALLS, CEILING & FLOOR OF EXPOSURE ROOM	"	70,920	355	(see separate section for detail)	27.49
3. "	• BELOW TANK IN ROOM	"	1,500	7.5	252 μCi/ft³	1.89
4. LEAD	• MOVABLE SHIELD (51"x72"x2")	708	3,010	4.3	1.30 μCi/lb	3.91
5. "	• CURTAIN ABOVE SHIELD	"	1,847	2.6	0.62 μCi/lb	1.15
6. "	• ON FRONT WALL EXPOSURE RM.	"	10,896	15.4	0.40 μCi/lb	4.36
7. "	• LEAD SHOT IN POOL DOORS	"	40,000	(90)	0.10 μCi/lb	4.00
8. ALUMINUM	• ALUMINUM POOL DOORS	168.5	1,305	7.8	0.10 μCi/ft³	0.01
9. "	• CORE SHROUD	"	168	1.0	75 mCi/ft³	75.00
10. "	• TANK WALL: 5/16" thick with phenoline liner	"	815	6.2	(3.6 μCi/ft²)*	0.70
11. WOOD (dry fir)	• FRONT EXPOSURE ROOM	26	3,328	128	0.40 μCi/ft³	0.05
12. "	• SIDES EXPOSURE ROOM	"	6,656	256	0.25 μCi/ft³	0.06
13. "	• BACK EXPOSURE ROOM	"	4,160	160	0.15 μCi/ft³	0.02
14. "	• FLOOR & CEILING " "	"	20,800	800	0.25 μCi/ft³	0.20
15. STEEL	• SHIELD HOIST	484	2,662	5.5	0.01 μCi/lb	0.03

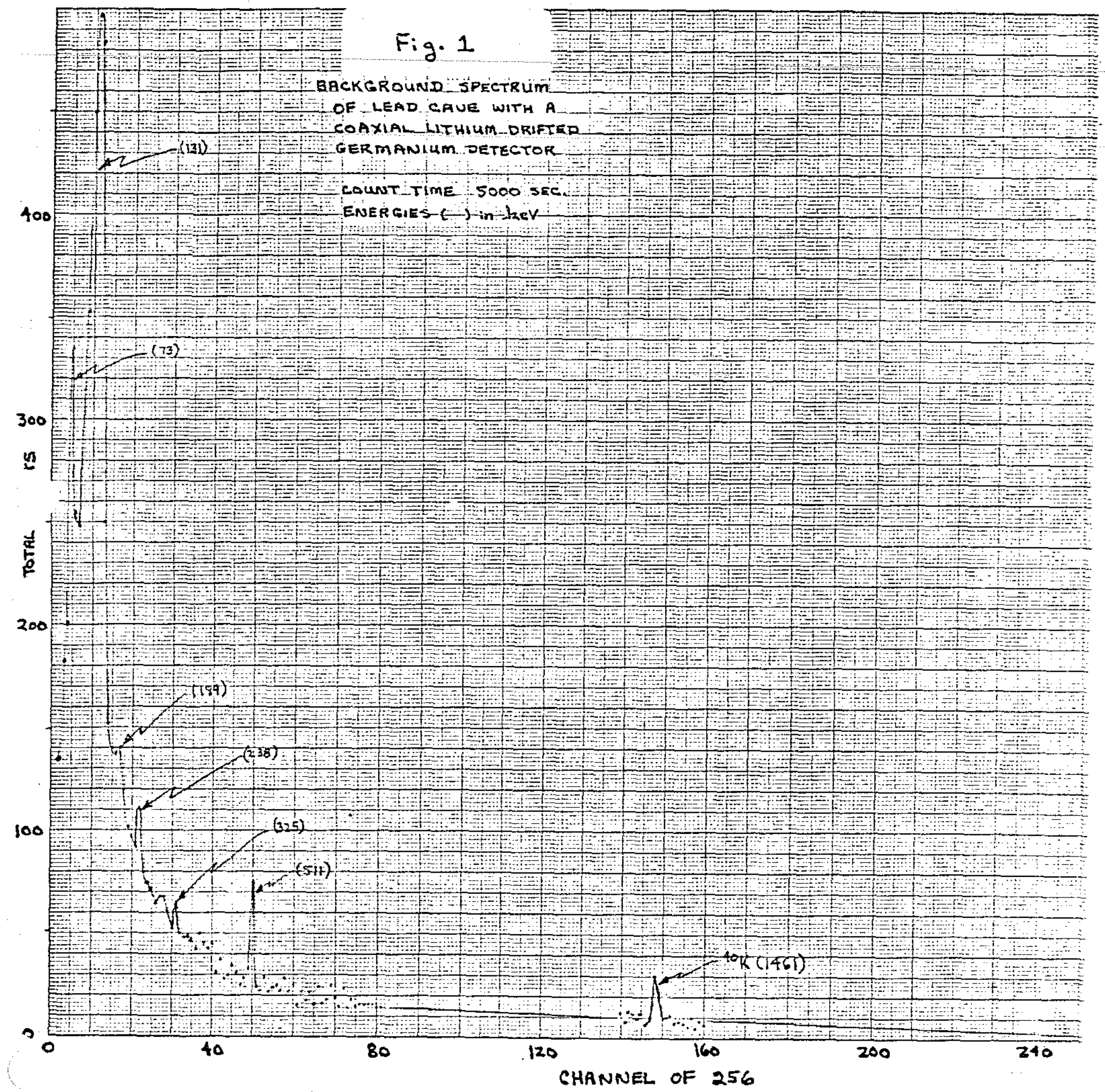
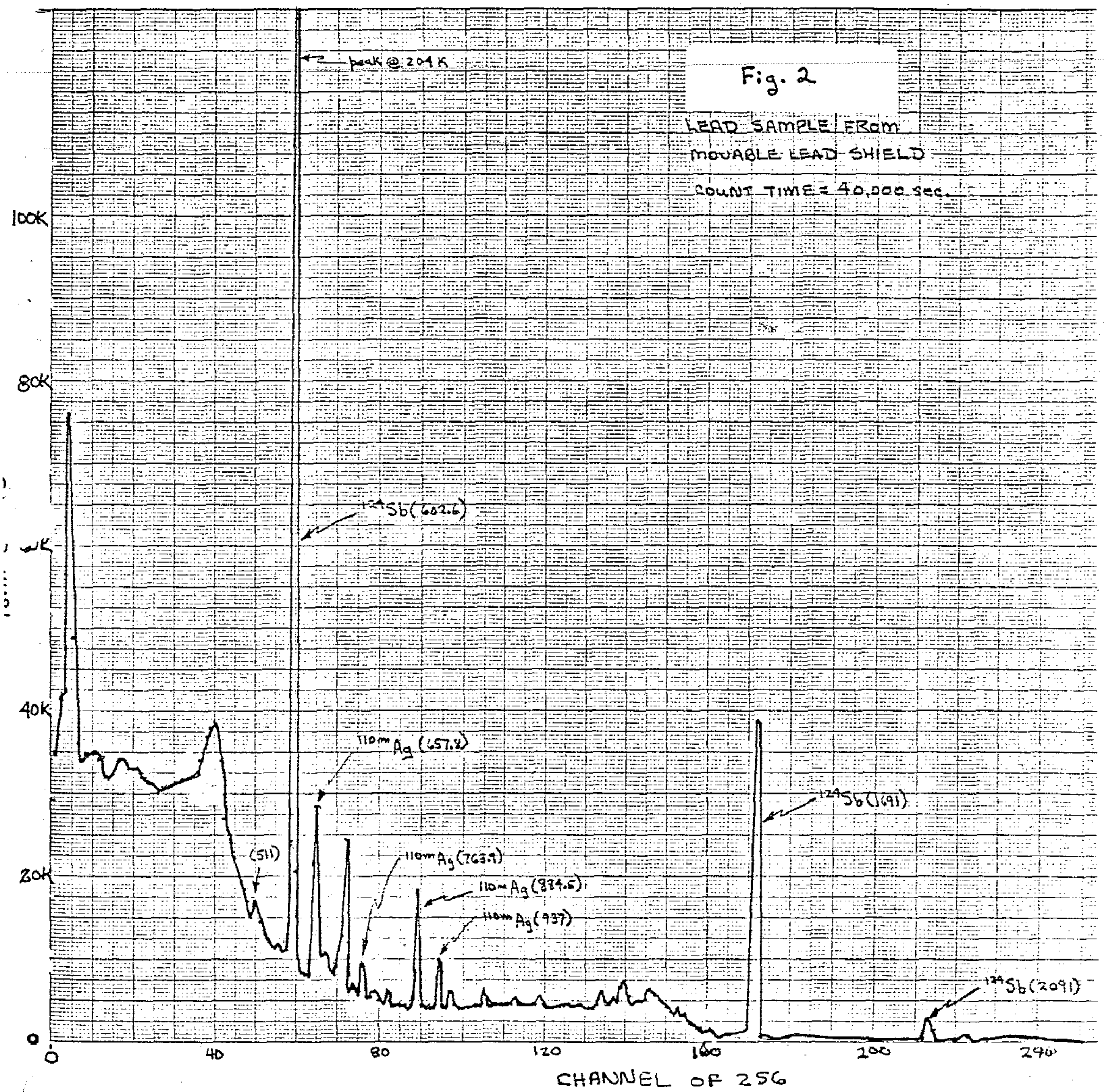


Fig. 1
BACKGROUND SPECTRUM
OF LEAD CAVE WITH A
COAXIAL LITHIUM DRIFTED
GERMANIUM DETECTOR
COUNT TIME 5000 SEC.
ENERGIES ( ) in keV
(131)
(73)
(159)
(238)
(325)
(511)
40K (1461)
TOTAL
400
300
200
100
0
40
80
120
160
200
240
CHANNEL OF 256


peak @ 204 K
Fig. 2
LEAD SAMPLE FROM
MOVABLE LEAD SHIELD
COUNT TIME = 40,000 sec.
100K
80K
40K
20K
0
124Sb (602.6)
110mAg (657.8)
(511)
110mAg (763.9)
110mAg (834.6)
110mAg (937)
124Sb (1691)
124Sb (2091)
0
40
80
120
160
200
240
CHANNEL OF 256

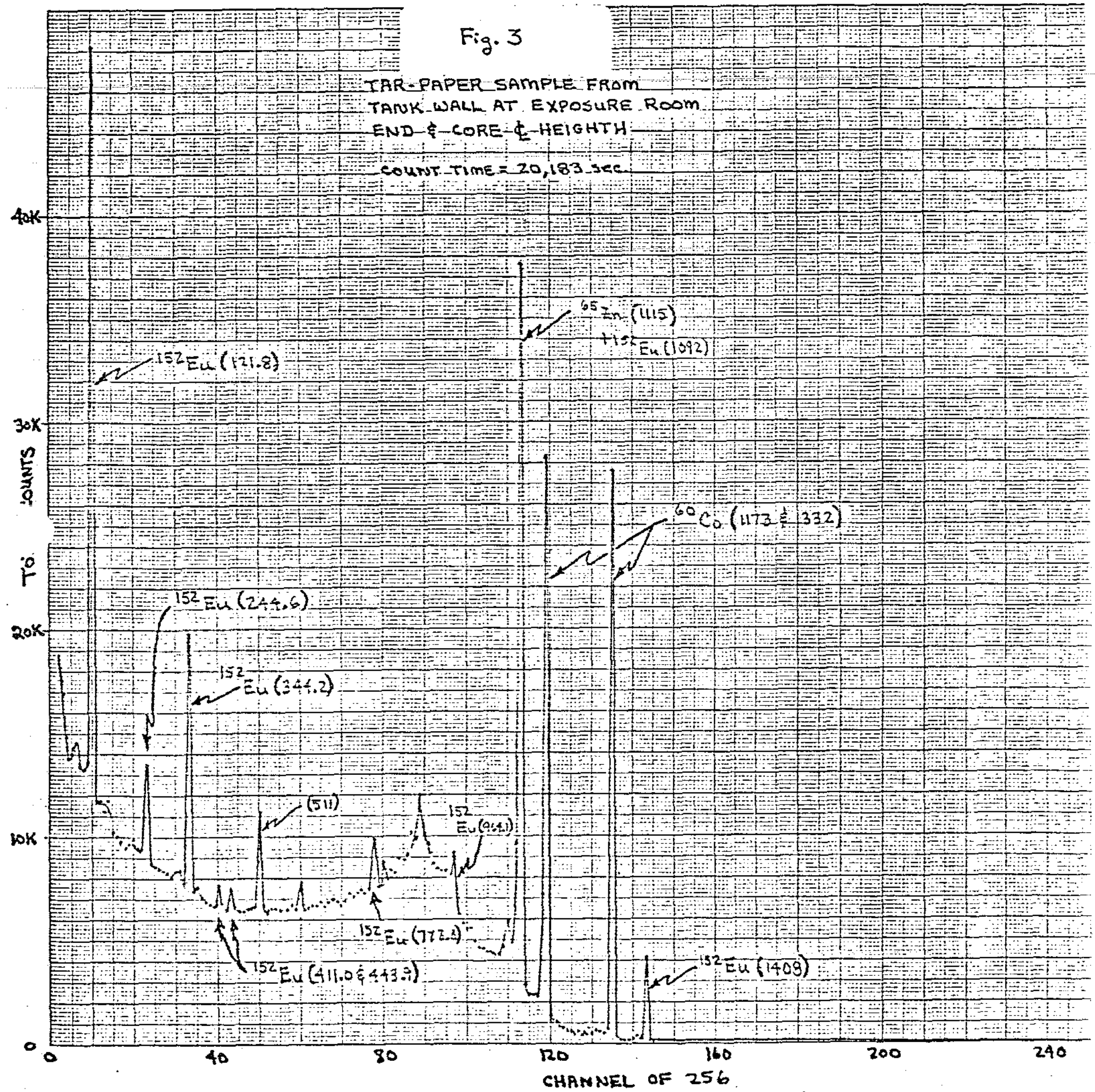


Fig. 3
TAR-PAPER SAMPLE FROM
TANK WALL AT EXPOSURE ROOM
END & CORE ℄ HEIGHTH
COUNT TIME = 20,183 SEC.
40K
30K
20K
10K
0
TOTAL COUNTS
^{152}Eu (121.8)
^{65}Zn (1115)
+ ^{152}Eu (1092)
^{60}Co (1173 & 1332)
^{152}Eu (244.6)
^{152}Eu (344.2)
(511)
^{152}Eu (964.1)
^{152}Eu (778.9)
^{152}Eu (411.0 & 443.9)
^{152}Eu (1408)
0
40
80
120
160
200
240
CHANNEL OF 256

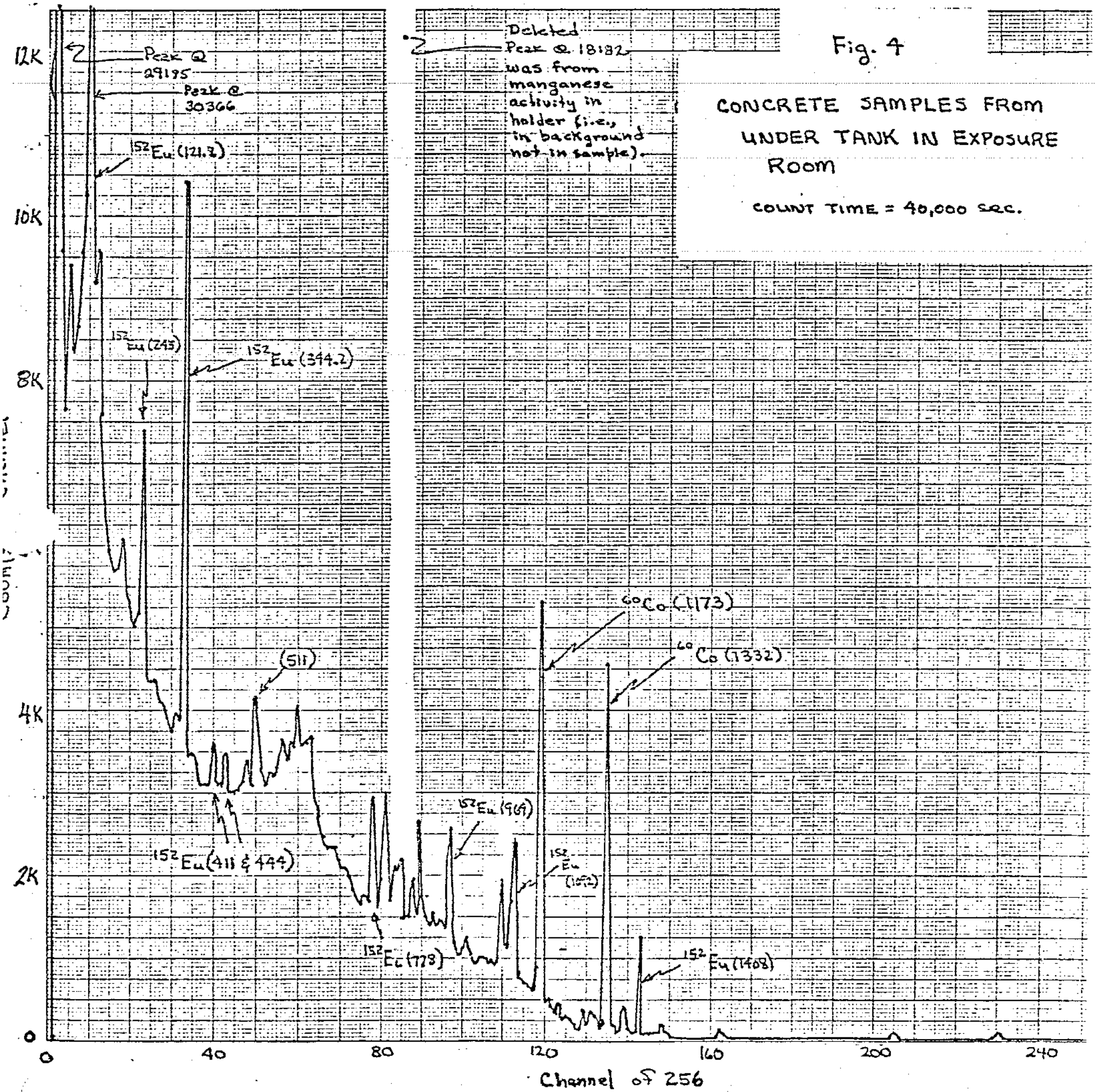


Fig. 4
CONCRETE SAMPLES FROM UNDER TANK IN EXPOSURE Room
COUNT TIME = 40,000 sec.
Peak @ 29195
Peak @ 30366
Deleted Peak @ 18182 was from manganese activity in holder (i.e., in background not in sample).
152Eu (121.2)
152Eu (245)
152Eu (344.2)
(511)
152Eu (411 & 444)
152Eu (778)
152Eu (964)
152Eu (1052)
60Co (1173)
60Co (1332)
152Eu (1408)
12K
10K
8K
4K
2K
0
0
40
80
120
160
200
240
Channel of 256

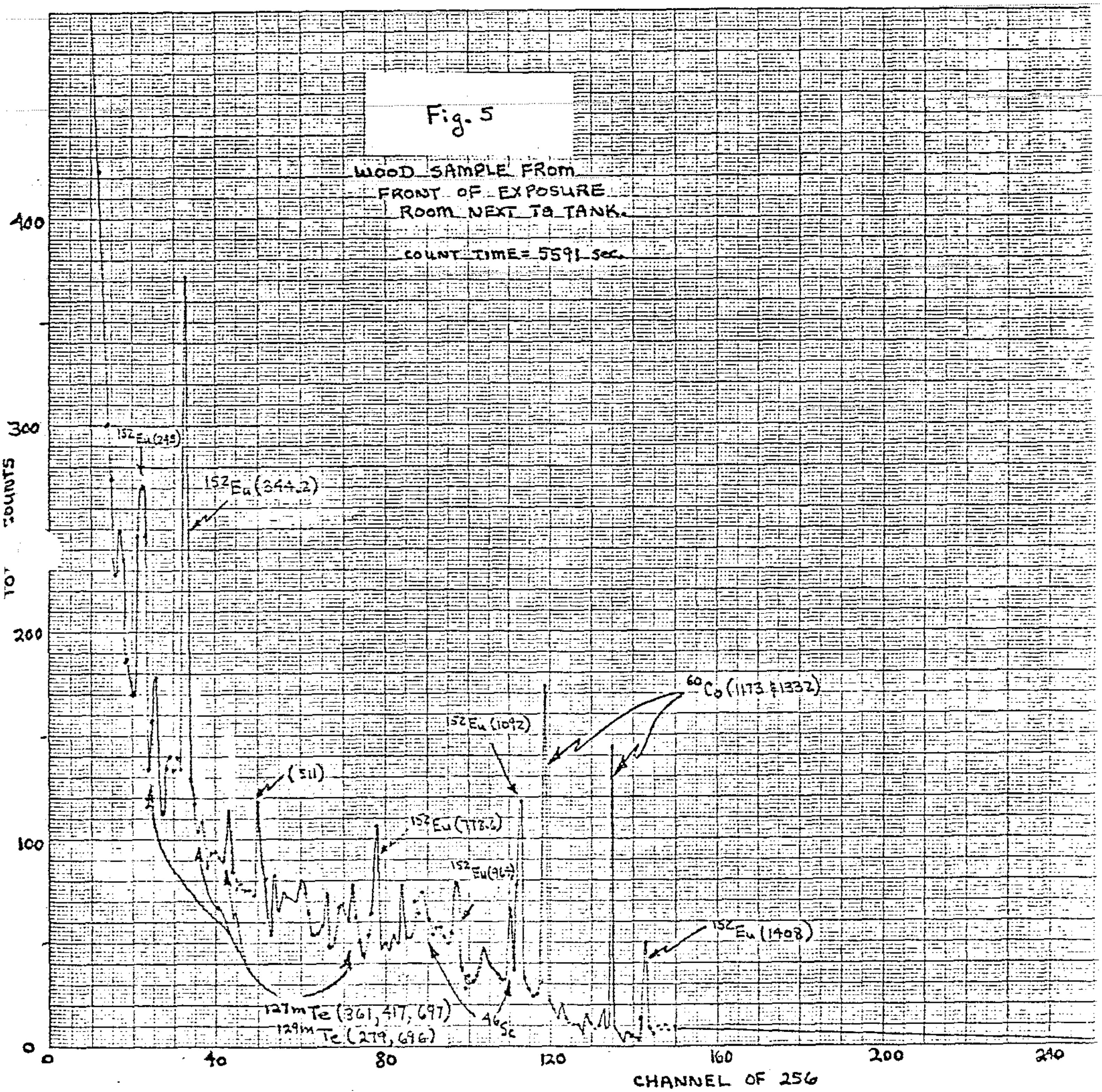


Fig. 5
WOOD SAMPLE FROM FRONT OF EXPOSURE ROOM NEXT TO TANK.
COUNT TIME = 5591 sec.
TOTAL COUNTS
CHANNEL OF 256
0
40
80
120
160
200
240
100
200
300
400
152Eu (245)
152Eu (344.2)
(511)
152Eu (778.2)
152Eu (964)
152Eu (1092)
60Co (1173 & 1332)
152Eu (1408)
127mTe (361, 417, 697)
129mTe (279, 696)
46Sc

CONCLUSIONS & RECOMMENDATIONS

The principal conclusion from this study is that once the reactor-grid support structure has been removed there is very little radioactivity remaining at DORF. Unfortunately the levels are definitely above background, but only by factors of several hundred, and the radioactivity is mainly distributed throughout concrete walls and floors. Deep excavations will not be necessary. However, this is of little consequence if one still has to remove several inch-thick layers from large areas. This is the situation in the exposure room. In fact, the exposure-room decontamination is by far the major problem and several possible methods of attack come to mind.

(1) Excavate and remove the three 5000-gallon waste-water holding tanks, cut off part of the tops and use them as shipping containers for the radioactive debris from DORF. For example, the wood has suffered radiation damage and dry rot so that it crumbles rather easily. It is a big volume (1200 ft^3) but relatively light in weight so it can easily be tossed or shoveled into the tanks and they could then be closure welded for shipment. There will also be much dust, dirt, paper and small concrete chips of radioactive waste, all of which could be put into the tanks.

(2) Mechanically cut, DO NOT CUT WITH A TORCH, the aluminum because of the radioactive "tar-paper" liner which could easily catch on fire and produce contaminated smoke. However by reference to the excavation-of-concrete details in this report, the places where the aluminum liner will be radioactive are easily identified. It does not appear that the liner will produce a problem in other than these areas.

(3) Thought should be given to the possibility of transferring some of the lead to AFRRI or APRF because its radioactivity is really not a serious hazard and these facilities need it for shielding in neutron fields. This could save a few dollars on transportation and disposal costs.

(4) Survey activities are going to be a problem because there just isn't much activity to survey right now. For example, depending on what is going to be done with the exposure room, it may not be necessary to excavate concrete from the rear wall of the room. In any event, thought should be given to how much the survey reading "from the rear wall only", before excavation, must be decreased by material removal to provide an "acceptable" survey level. In view of the expense to breakup and ship concrete, it is prudent to be practical about sealing up or burying very small, but detectable, amounts of radioactivity.

(5) Almost all of the materials exhibit one or two predominant and characteristic photopeaks. Therefore, survey activities could be determined by a sodium-iodide scintillation detector. It is suggested that a portable detector with a 3/4-inch-thick cylindrical lead shield around the sides would be practical. Calibration could be accomplished in a crude, but adequate, manner by measuring the response of a variety of sources simultaneously positioned over a square-meter plane area behind about 1/4-inch thick aluminum. This approximates the following situation. The dose rate to tissue in rads per hour in an infinite medium, of density ρ, uniformly contaminated by a gamma emitter, of energy E (MeV), is

$$2.12\ EC/\rho \qquad (3)$$

where C is in microcuries per cm^3. At the surface, the dose rate is about one half of this and for air a one-centimeter-from-the-surface survey is an adequate representation of the surface rate. By then surveying the "calibration setup at one meter" and correcting for $1/R^2$ to one centimeter, one can estimate the rads per hour efficiency of the scintillation detector. A variety of sources, repositioned should be used and the results averaged.

MEDEC-RS 12 December 1967

MEMORANDUM FOR RECORD

SUBJECT: Authorized Uses of DORF Reactor

1. Ref telecon between undersigned and Mr. Dube, Div, Materials Licensing, USAEC, Code 119-7491, this date.

2. In response to questions from Harry Diamond Laboratory concerning the proposed use of DORF to produce radioisotopes for medical purposes,' the undersigned, after a thorough search of the Code of Federal Regulations and other available documents, initiated a call to the office of Mr. Cecil Buchanan of the Atomic Energy Commission. Mr. Buchanan was not available, however Mr. Dube of that office discussed the question. Although Mr. Dube did not initially have a full answer to the question as to the authority to produce radioactive material, he obtained the needed information from Mr. Edward R. Fleury, Division of Reactor Licensing, AEC.

3. In the opinion of the Atomic Energy Commission, as expressed by Mr. Fleury and Mr. Dube, the authority to produce radioactive material for such purposes at the Walter Reed Research Reactor is contained in the provision of use of the Reactor for research and development. It is considered that a statement of that nature is broad enough to encompass any operations which are currently being conducted utilizing that Reactor.

4. Since DORF is an unlicensed reactor and comes under the provisions of Section 91b of the Atomic Energy Act, the regulation of the Atomic Energy Commission is quite indirect. In the initial planning stages it is necessary that the military show a military need for the reactor and thereby qualify it under Section 91b. Once this military need has been established, and so long as the reactor continues to be used for military purposes, the control of the reactor and the uses to which it will be put are strictly the purview of the military, in this case the Army.

5. If the reactor ceases to be used for military purposes or is sold or relinquished to some other source, then the facility must be licensed and it is necessary for the Army to make proper notification to the Atomic Energy Commission so that this may be done. The proposed project of activating gold is obviously a military related project. As such the control of DORF is strictly within the military so far as the AEC is concerned. This, of course, does not hold for the production of fissionable material.

6. Along with the discussion it was pointed out that certainly the internal safeguards which are established such as the Reactor Safeguards Committee, the Authorization Procedure, etc. must remain in effect for any project whether it is of military need or otherwise.

William F. Kendall

WILLIAM F. KENDALL
MAJ, MSC
Asst Health Physics Officer

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F.N. Wimenitz
Br 230, HDL

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

MAY 31 1967

Mr. W. B. Murphy
Chief, Safety Office
Office of the Chief of Engineers
Department of the Army
Washington, D. C. 20315

Dear Mr. Murphy:

We have reviewed the proposed revision of Army Regulation 385-80, Nuclear Reactor Systems Health and Safety, transmitted with your letter dated March 29, 1967, and have found that it is in accord with the AEC-DOD Memorandum of Understanding on the health and safety responsibilities for Section 91b reactors. We have also noted the statement in the proposed regulation that it does not abrogate the Army's responsibility to conform with the applicable AEC license requirements with respect to Army facilities that are licensed by the Commission.

It appears that the proposed regulation adequately defines the policies, responsibilities and procedures for the Department of the Army nuclear reactor health and safety program. We have noted that AEC's current concepts relating to the development of Safety Analysis Reports and Technical Specifications, as described in AEC's proposed amendment to 10 CFR Part 50, have been factored into the proposed Army regulation. In this connection, we suggest that, even for existing facilities which have submitted Final Safeguards Reports under paragraph 6.d of your proposed regulation, your plans include formation of Technical Specifications and their bases in accordance with current AEC concepts.

Sincerely yours,

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Chapter 1

SITE

LOCATION

The Diamond Ordnance Radiation Facility Research Reactor will be located within the metropolitan area of Washington, D.C., at the Forest Glen Annex of the Walter Reed Medical Center, which is five miles from the center of Washington, D.C., and approximately two miles south of Kensington, Maryland (see Fig. 1). The Forest Glen site is an area of approximately five acres of rolling, partially wooded and cleared areas, on which are located both research-laboratory facilities and hospital facilities for patients (see Fig. 2). The site is located in a commercial and residential area.

The location of the reactor will be near the southern border of the Forest Glen area (see Fig. 2) about 800 ft from the nearest research laboratories and about 500 ft north of Brookville Road, which bounds the property on the southeast. The site is surrounded by woods, except to the north, where a large open field separates it from the research laboratories. The reactor building will be encircled by an exclusion fence with a radius of approximately 200 ft (see Fig. 3). Access to the exclusion area will be controlled at the single entrance gate.

HYDROLOGY

Surface drainage at the reactor site is to the south and west into Rock Creek, which drains south through Rock Creek Park and into the Potomac River. The discharge of Rock Creek at Sherrill Drive in the District of Columbia, 7-1/2 miles upstream from its mouth, is as follows:

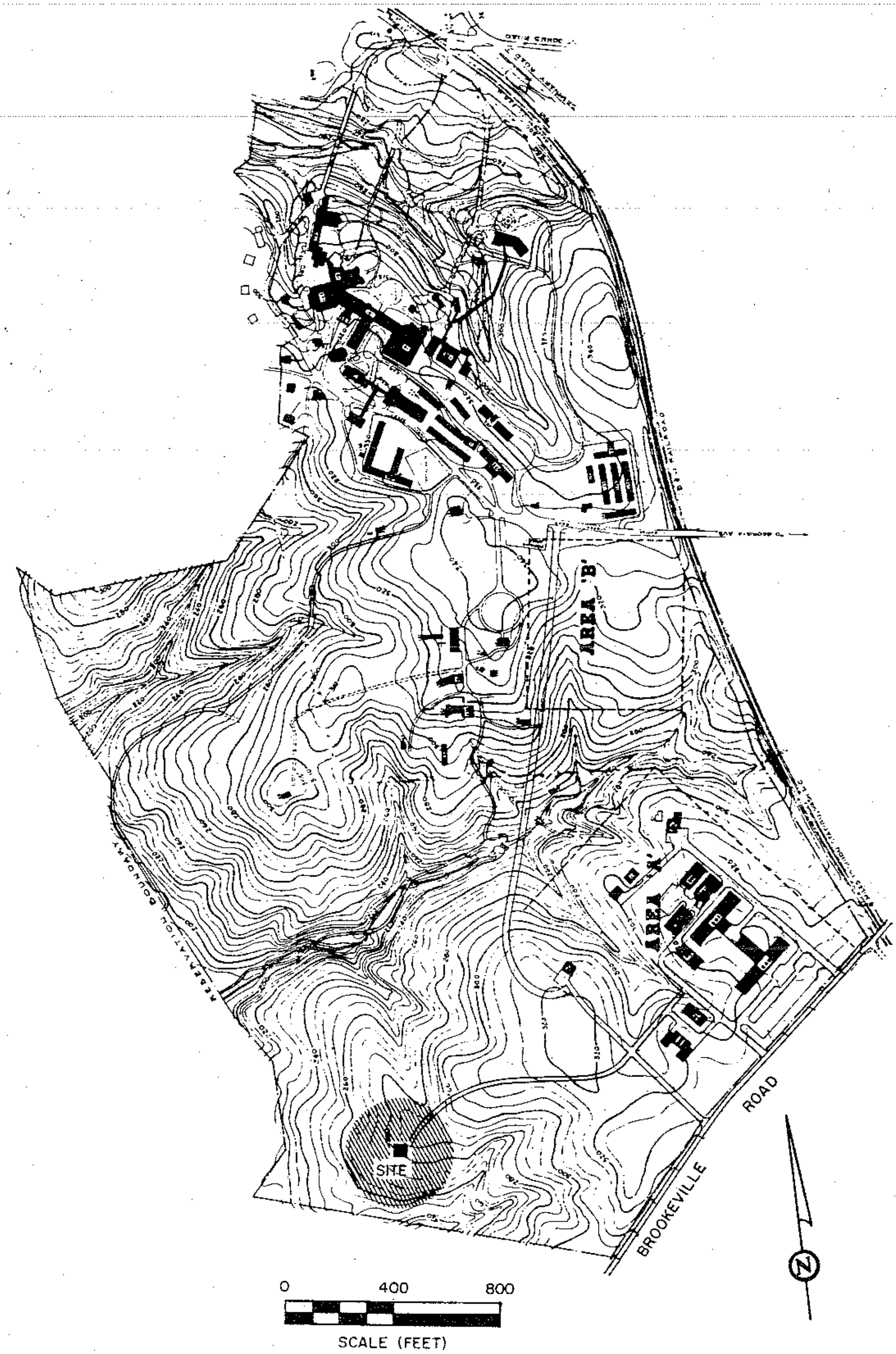


Fig. 2--Map of Forest Glen site for the DORF

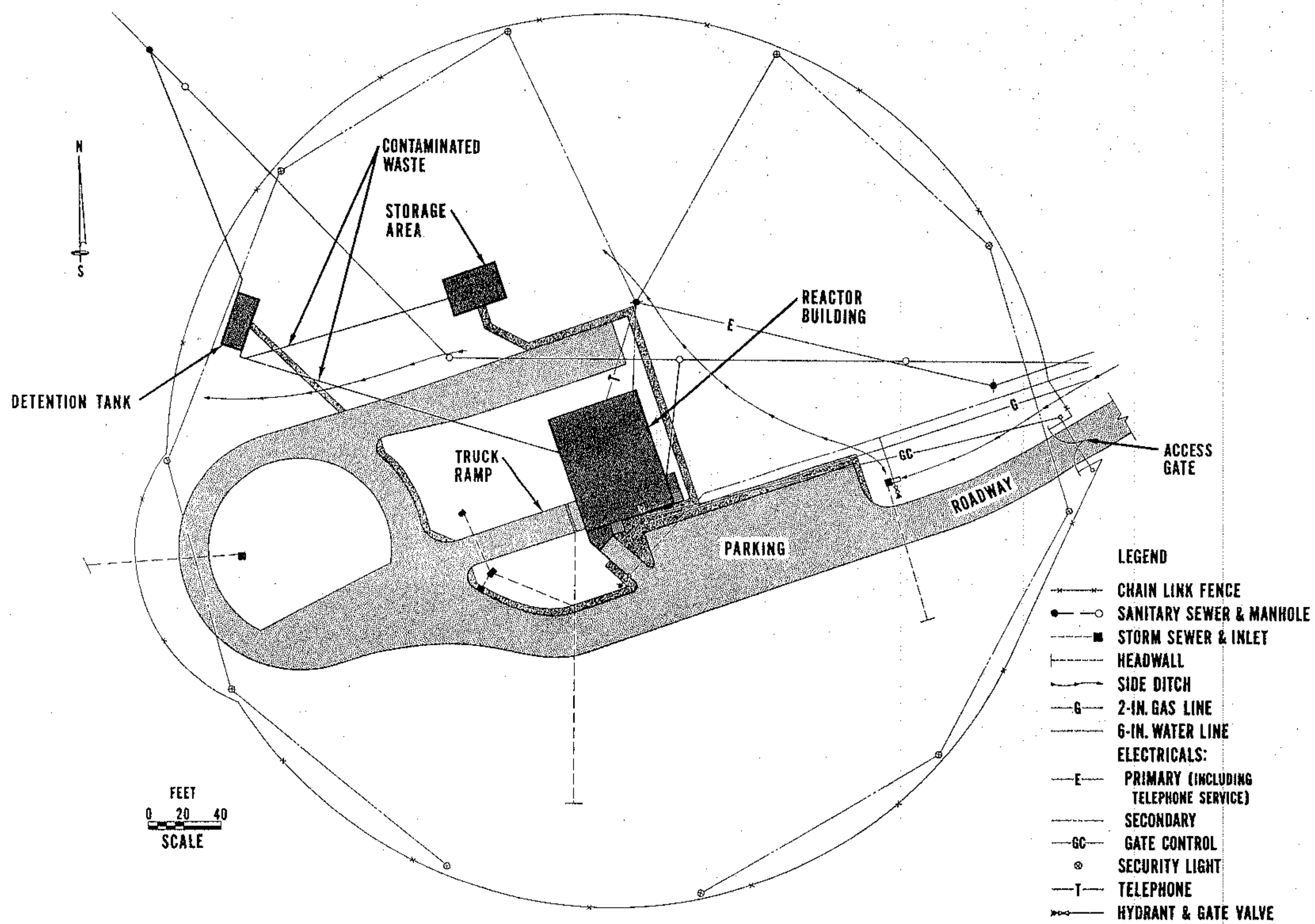


Fig. 3--The DORF site

28-yr average . . . . 55.8 ft^3/sec

1929-1957 maximum . . . . 7220 ft^3/sec, July 21, 1956

Calendar year 1956 average . . . . 48.4 ft^3/sec

Water year October, 1956,
through September, 1957:

Maximum . . . . 1210 ft^3/sec, April 5, 1957

Minimum . . . . 2 ft^3/sec, August 17, 18, 19, 23, and 25, 1960

The ground water at the proposed reactor site is locally recharged by precipitation and infiltration. The average yearly precipitation in the Montgomery County area is about 41 in., evenly distributed throughout the year. July is the wettest month (about 4.5 in. of rain), and October is the driest (about 2.8 in.). Most of the precipitation is lost to the atmosphere by evapotranspiration, and only a small fraction enters the zone of saturation. The zone of saturation, the zone below the water table, was not encountered in test borings at the reactor site, the deepest of which was 45 ft.

The sand, silt, and clay of the weathered rock, saprolite, has a low permeability, probably less than 1. Since the water table was not reached by the test borings, which were made during the peak of the average annual 10-ft variation in the water table found in the Washington vicinity, it is reasonable to assume that the depth of the water table will seldom be less than 45 ft.

It was not possible to determine the rate of ground-water movement. In general, however, the velocity of ground water through fine-grained sedimentary materials, such as fine sand, silt, and clay, is very low (on the order of a few feet per year).

WATER WELLS

No wells are known to exist in the immediate vicinity of the proposed reactor site. The nearest known supply well is 1-1/2 miles east of the site, and another is located about 2 miles north of the site. Both wells are considerably higher on the regional hydraulic gradient which rises to the northeast of the reactor site.

Rock Creek flows south past the reactor site, traverses the entire length of the District of Columbia through Rock Creek Park, a recreational area, and discharges into the Potomac River below the intakes for municipal water supplies.

EARTHQUAKES

The Maryland Piedmont, on which the site is located, is a region of comparative crustal stability. During the past 156 yr no severe quakes have occurred, though there have been a few quakes of low intensity. The probability of a destructive earthquake in the Montgomery County area is very slight. The probability of hazards arising from extra-site sources, such as explosions or floods, is greater than from seismic activity.

POPULATION

The DORF site will normally be occupied by 15 male personnel on a normal 8-hr day basis. The Forest Glen Annex of Walter Reed Army Medical Center has a total estimated population of 1041, located as follows:

Research and Development Area (Nearest to reactor site)	
Civilian employees	99
Military	39
Total	138
Convalescent Area	
Civilian employees	229
Military	374
Patients	250
Resident civilians	50
Total	903
Total estimated population	1041

The area outside the medical center towards metropolitan Washington is densely populated. Metropolitan Washington has an estimated 1.4 million people.

GEOLOGY

Topography

The Army Medical Center in Forest Glen is on the well-dissected hills bordering the east side of Rock Creek. The reactor site is on a small bench at an altitude of about 280 ft above mean sea level. Surface drainage flows from the site directly into Rock Creek and to the north and south into small unnamed tributaries of Rock Creek, and from these to the Potomac River near Georgetown and Theodore Roosevelt Island (see Fig. 1).

Subsurface Geology

The site is on a narrow belt of the Kensington-granite gneiss, which is a highly foliated, coarse granite intrusive in the Wissahickon schist complex and basic rocks. Petrographic mineral determinations are contained in Reports TDS-A526 and IDM-A526 of the United States Department of the Interior, Geological Survey, Geochemistry and Petrology Branch.

The relic structures of the bedrock just described extend into the weathered rock mantel, saprolite, which ranges in thickness at the site up to about 45 ft. Locally, boulders or veins of quartzite, essentially unaltered by weathering, remain in the saprolite. The saprolite is composed of micaceous silt, clay, and fine-to-medium grained quartz sand. The attitude of the planes of foliation orient the average permeability, so that ground water and other fluids that might be released to move through the ground travel more readily along the planes of foliation than across them. At the proposed reactor site, ground-water movement in the bedrock and in the saprolite, where it has retained relic structures of the bedrock, will be more rapid north and south, parallel to the foliation, than east and west across the planes of foliation. Fluids which may seep into the ground at the reactor site will most likely discharge at the surface in the gullies or hillsides to the north and south of the reactor site, in line with the direction of foliation, rather than directly down the surface slope toward Rock Creek. Thus, the west-northwest dip of the rocks at the site will retard any fluids released to the ground at the reactor site, provided that they do seep into the ground and do not run off on the surface directly into Rock Creek or into the small tributaries to the north and south. The ion-exchange capacity of the weathered metamorphic rocks in the vicinity of Washington, D.C., is usually less than 25 milliequivalents per 100 g (< 25 meq/100 g), which is low in comparison to purer clay formations, such as those of marine origin. Analyses of 12 rock samples from this site show exchange capacities of less than 10 meq/100 g. These analyses were made by the Geochemistry and Petrology Laboratory of the U.S. Geological Survey.

Test Drilling

Five bore holes and one auger hole were made to obtain samples for construction-engineering information and for the evaluation of environmental hazards. The deepest hole was bored to a depth of 45 ft. It was located exactly at the point where the reactor will operate, adjacent to the exposure room, and samples from this hole are representative of the earth materials that may be irradiated by reactor flux.

A second hole was bored outside the north wall of the exposure room, and a third was bored outside the south wall of the building, adjacent to the reactor pool. At the southwest corner of the building, a hole was bored to check the variation in the soil that might exist in the vicinity of the building. A bore hole and an auger boring were made at the proposed location of the waste-storage tanks and the line of waste discharge from the reactor building.

Chapter 2

FACILITY

GENERAL

The purpose of the facility is to study the effects of large, mixed doses of neutron and gamma radiation on electronic systems and related devices. To carry out these experiments, both a main exposure room and pool irradiation space are provided.

BUILDING

The building is designed to accommodate the reactor, reactor components, reactor shield structure, technical areas, personnel, and the equipment and instrumentation required to perform radiation-exposure testing of electrical and electronic components or systems and also metallurgical testing.

The floor plans and elevations of the three levels in the facility are shown in Figs. 4 and 5, respectively. The building, which is designed for a 25-yr life, is constructed of reinforced concrete, structural steel, and masonry. The building will accommodate an average of 15 male personnel on the basis of an 8-hr/day occupancy. Space within the building is divided into the following areas:

Exposure room	400 ft^2
Warm storage room	300 ft^2
Truck-access and sample-preparation area	300 ft^2
Mechanical-equipment room .	300 ft^2
Toilet and shower room	100 ft^2
Nuclear counting room	100 ft^2
Office	120 ft^2

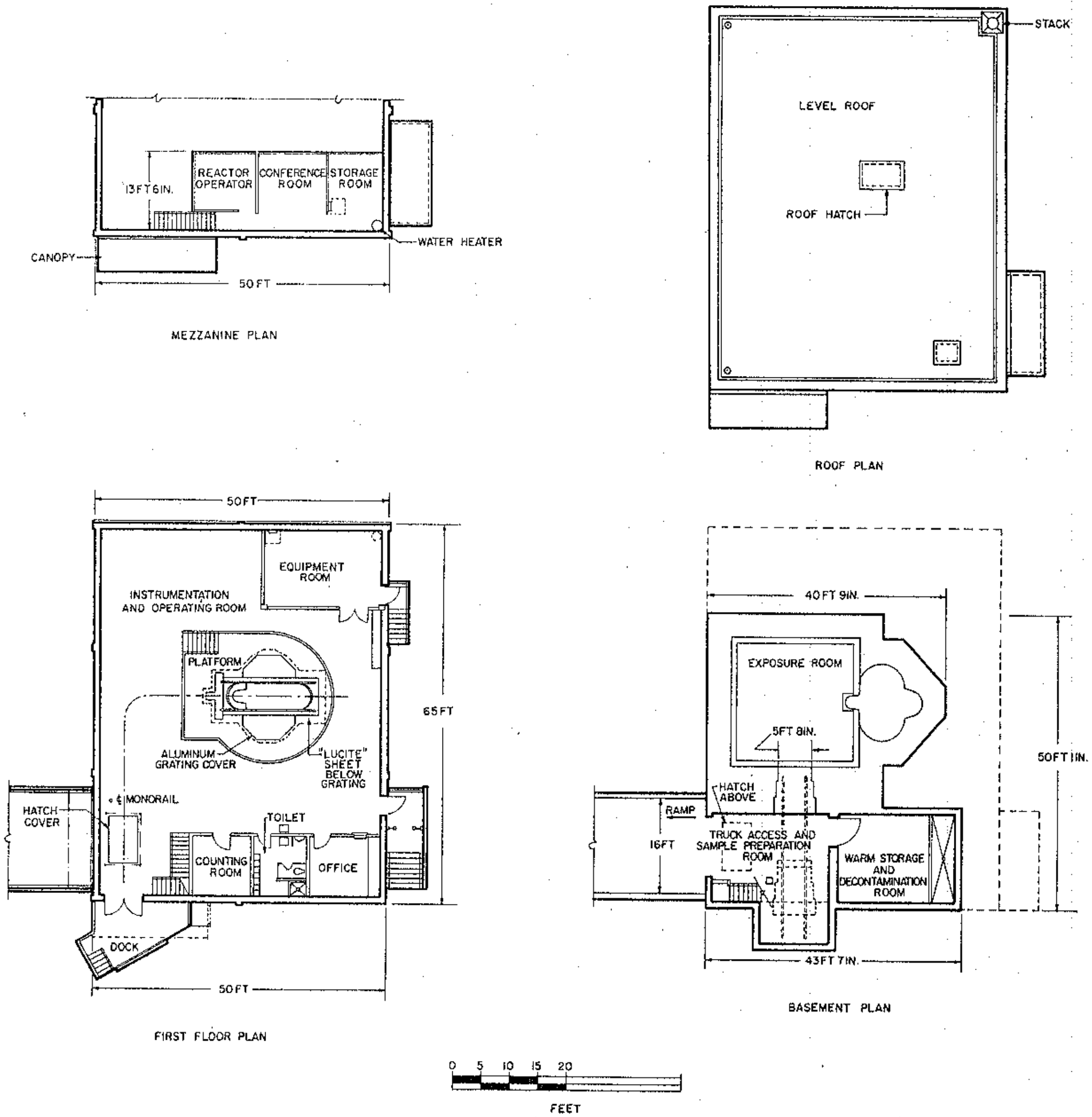


Fig. 4--Plan views of the DORF building

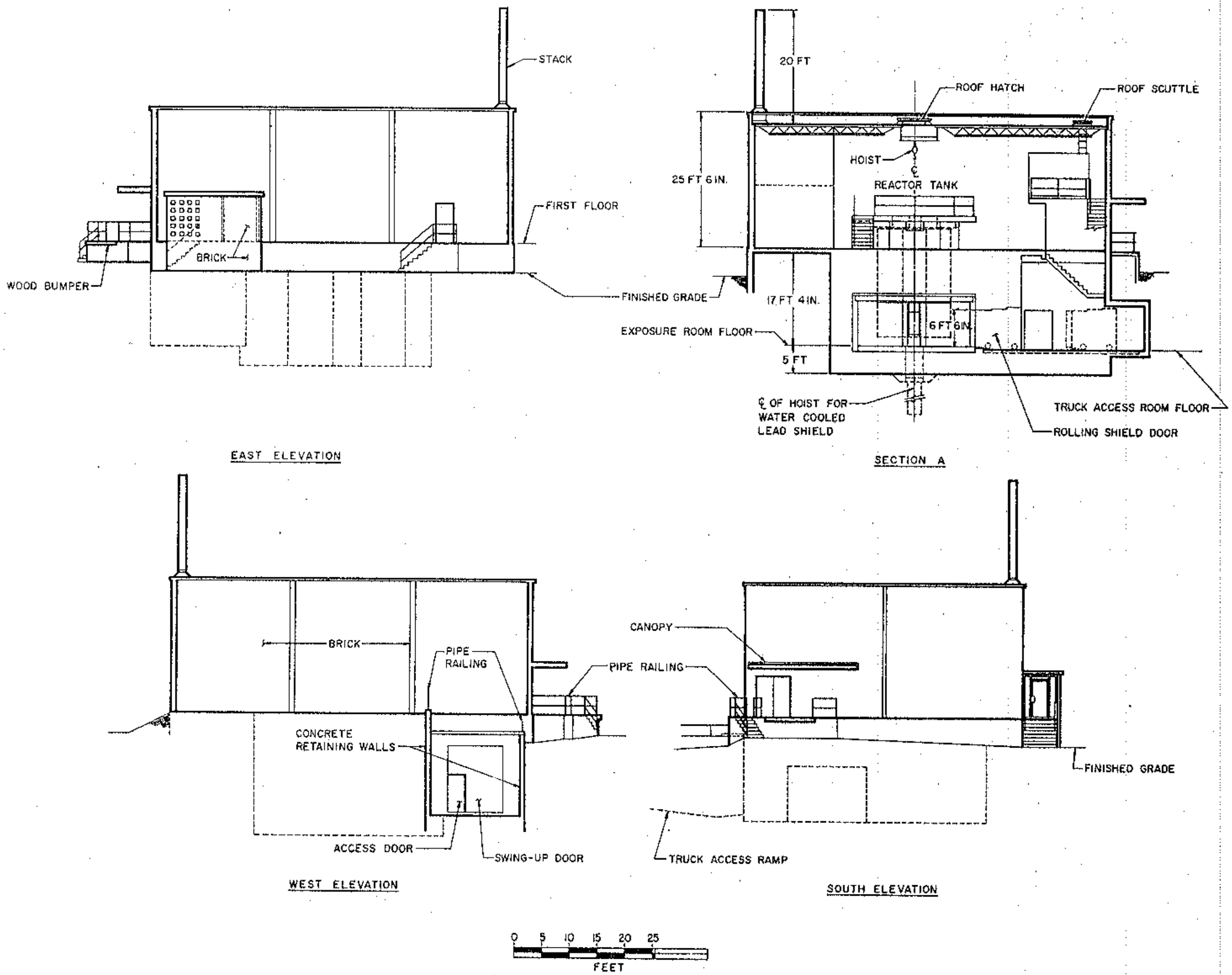


Fig. 5--Elevation and section views of the DORF building

Reactor operator station . . . 110 ft^2

Conference room . . . . 170 ft^2

Storage room . . . . 120 ft^2

The foundation and the exterior walls below the operating floor are constructed of reinforced concrete, with a waterproof membrane to prevent ground-water penetration into the building. The exterior walls above the level of the operating floor are constructed of 4-in. brick facing and an 8-in. cement-block backup. The cement blocks are sealed with an epoxy resin on the inner surface to reduce porosity.

The roof and ceiling are of perlite cast over metal decking with five layers of felt roofing on top.

AIR-CONFINEMENT CAPABILITIES

The building for the DORF-TRIGA reactor has been designed to operate at normal atmospheric pressure. The design also takes into consideration the intended uses of each area and provides good air confinement under adverse atmospheric conditions. Even under the unlikely possibility that a fuel-element cladding failure might occur concurrent with a sudden large drop in barometric pressure, the airborne radioactive fission products outside the building would not exceed the maximum permissible concentration. An analysis shows that the maximum dose which a person might receive in the area of maximum concentration outside the building would be 2 orders of magnitude below the maximum permissible dose for nonoccupational personnel established by 10 CFR 20.

The doorways and roof hatches in the building represent the only escape path for air except for the slight possibility that air might diffuse through several inches of concrete. All exterior doors and roof hatches will be well fitted and gasketed with rubber. As a result, the building will have no communication with the outside air when the doors and ventilation dampers are closed. A pressure-release system, described below, will control the differential pressure.

Building Ventilation Under Normal Conditions

The air-conditioning system exhausts all air from the reactor building through absolute filters and out the stack. This precaution is taken, even though no radioactive particulate matter will be in the air during normal operation.

The only possibility which might create airborne radioactive particulate matter would be a fuel-element cladding failure. Should such a failure occur, it is possible that small amounts of radioactive noble gases would be dispersed from the reactor pool into the reactor-room air, and these would decay into particulate matter.

Controlled Confinement of Air

Although the release of fission-product gases from a fuel-element cladding failure should in no way endanger the operating personnel or the public, the design of the building makes it possible to isolate and confine the air within the building. The system has been designed so that when the continuous air monitor indicates abnormal airborne contamination, an alarm automatically sounds and the positive-sealing dampers in the ventilation system automatically close, isolating the air in the building at ambient atmospheric conditions.

Two conditions which could cause the confined air to leak out of the building would be a sudden drop in barometric pressure or an excessive heating of the air within the building, causing it to expand. In order to control the release of air from the building under either of these rather abnormal meteorological phenomena, a special atmospheric relief duct is provided in the system.

The resistance to air flow provided by this relief duct is very much less than the resistance to the flow of air through the solid concrete walls or through the compressed rubber gaskets which seal the doors and hatches. Therefore, it can be reasonably assumed that the small volume of air that will be expelled from the facility under either of these conditions will be directed through the atmospheric relief duct and out of the building through the air stack. Since the circulation of outside air will continue through the emergency intake of the exhaust fan, the small volume of air expelled from the building will be considerably diluted as it is forced out the stack. By the time it reaches ground level, the concentration of Xe^{131} will be about 1/300 of that allowed by the Atomic Energy Commission.

Fission-gas Concentration From Stack Release

On the assumption that both events mentioned above have occurred and that a small fraction of the contaminated air from the reactor room has been released through the stack, a calculation of the resulting concentration of fission gas has been made. In order to determine the amount of fission-product gases which might be released at any one time, it was necessary to determine the effects of the two phenomena mentioned above.

Effect of Sudden Barometric Pressure Change on Room Air Leakage. A pressure differential between the inside and outside of the building can occur because of a sudden drop in barometric pressure. The barometric pressure changes in the Washington, D.C., area, as given in the records of the U.S. Weather Bureau dating from 1893, are as follows:

Maximum monthly variation, 1.87 in. Hg;
Maximum 24-hr variation (fall), 1.47 in. Hg;
Maximum 24-hr variation (rise), 1.22 in. Hg;
Maximum 12-hr variation (fall), 1 in. Hg.

When the barometric pressure drops, air will be expelled from the building. On the basis of the 1.47-in. drop in barometric pressure recorded above and the volume of reactor-room air, 71,350 ft^3, it has been calculated that 3,690 ft^3 of air will be expelled from the reactor room in 24 hr. This leakage amounts to 5.17% of the volume of air in the reactor room in 24 hr.

If it is assumed that the fission-product gases in the reactor room air resulting from a fuel-element cladding failure are xenon and krypton, with an average total activity in the room of 2.45 c, as calculated and discussed in Appendix IV, the average concentration of xenon and krypton in the reactor-room air will be 1.21×10^{-3} $\mu c/cm^3$. Since 5.17% of the activity will leak out with the expelled air, 0.127 c will escape in 24 hr, or 1.4 μc/sec.

On the conservative assumption that the average activity will not decrease from the natural decay of fission products during the 24-hr period, it has been calculated that the maximum concentration of the fission-product gases near the ground will occur at a distance of 1,940 ft from the reactor building. The maximum concentration in air at that point would be 8.0×10^{-10} $\mu c/cm^3$, which is 1/300 of the Xe^{133} maximum permissible concentration (10 CFR 20) for continuous exposure in unrestricted non-occupational areas.

Effect of Temperature Differential on Room Air Leakage. Another condition that could cause air to be exhausted from the reactor room, when the air in that room is isolated from the rest of the building because of a fission release from the reactor tank, would be high temperature outside the reactor building. Radiant heating from outside would raise the temperature of the isolated and confined reactor room air. Under these conditions, it has been calculated that an outside air temperature of 100°F might increase the temperature inside the reactor room 15°F before equilibrium was reached. These conditions are based on the assumption that the air-conditioning system is shut down early in the morning on a day on which the outside-air temperature

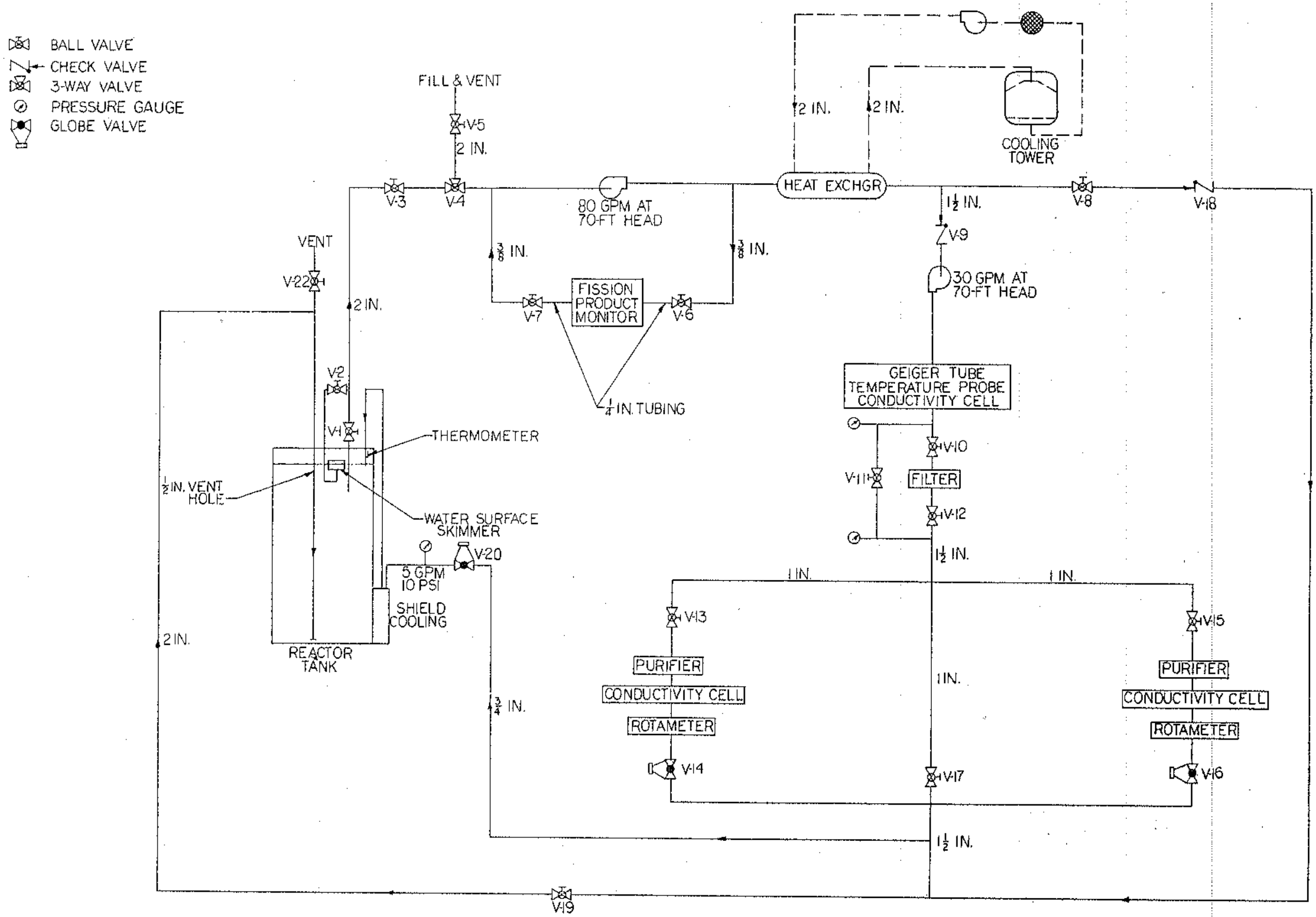


Fig. 7--Coolant-water purification system for the DORF

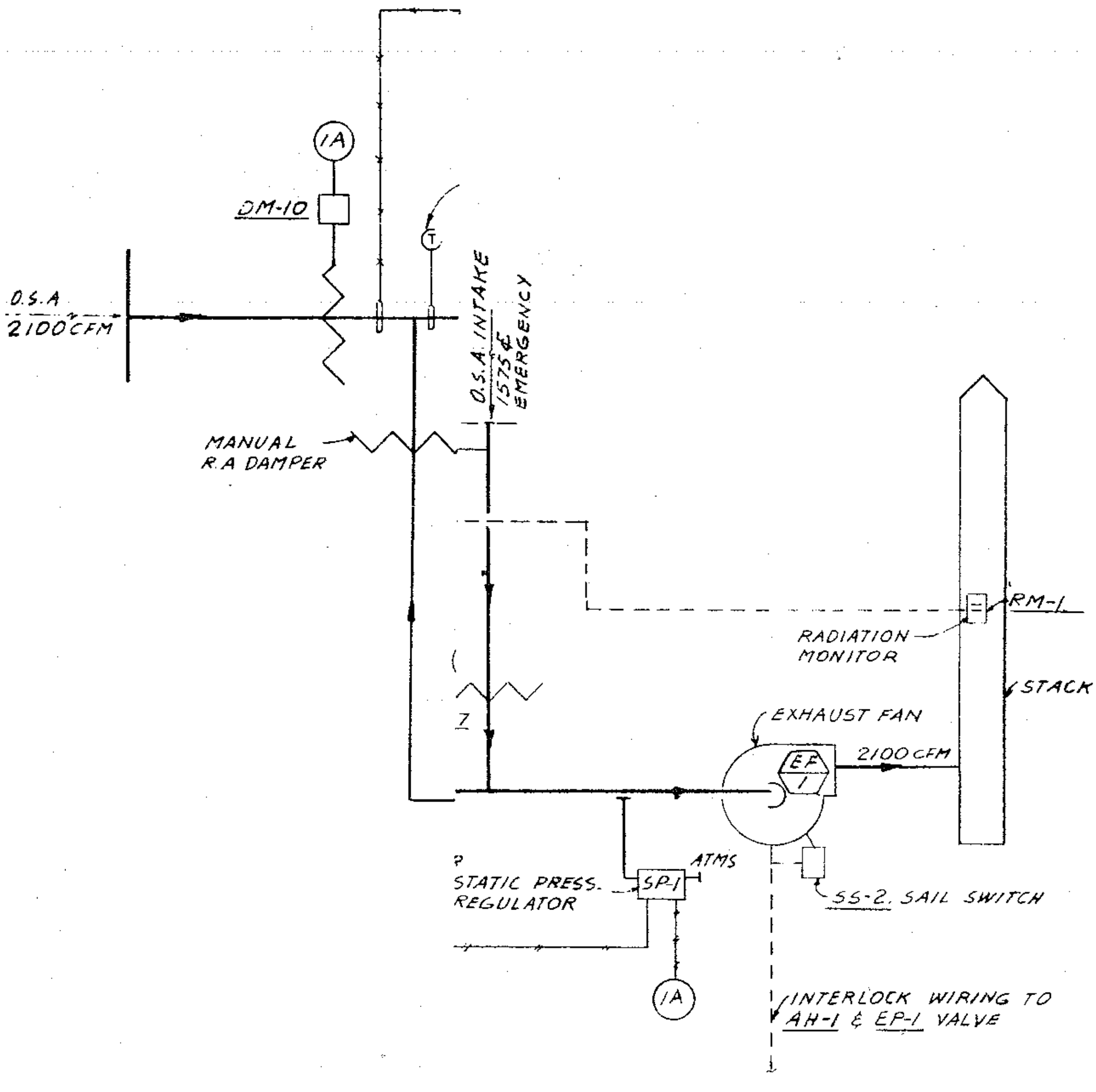


Fig. 6--Flow diagram of ventilation system for reactor building

reaches a maximum of 100°F. During the day, the air inside the building will increase in temperature from an assumed initial temperature of 70°F to 85°F by evening, when the temperature of the outside air would have decreased, creating an equilibrium. This condition would result in a leakage of 3.4% of the total reactor-room air volume in 24 hr.

It does not seem reasonable to assume that there could possibly be a coincidence of these two natural phenomena--namely, a sudden change in barometric pressure concurrent with a wide variation in inside and outside air temperatures.

Since the volume of air which would leak from the reactor building under the conditions created by a sudden change in barometric pressure is greater than the volume of air expelled by heating, the concentration in the air will be proportionately less in the latter case. Therefore, it can be assumed that there are no credible conditions which could endanger the health or safety of the public through the release of fission products from a fuel-element failure.

A calculation of the effect of a release of fission products in the building is presented in Appendix IV. The exposure (16.5 mr) which a person inadvertently remaining in the reactor room could receive following a fuel-element cladding failure is well below the maximum permissible daily exposure established by 10 CFR 20 for operating personnel.

VENTILATION SYSTEM

The ventilation system (Fig. 6) and air conditioning throughout the building will provide:

1. Dissipation of building heat generated by electrical equipment and lights;
2. Normal ventilation and cooling;

3. Positive filtered exhaust of all possibly contaminated air.

The entire building will be air-conditioned by a package type of system. Normal air recirculation will be used in the building, except in the exposure room and in the warm storage room. All exhaust air from the exposure room and the warm storage room will pass through an absolute filter of minimum 99.77% efficiency with a dioctyl phthalate (DOP) penetration of 0.05% for 0.3-μ-diameter particles, and then out through an exhaust stack that extends approximately 45 ft above the ground level. The flow of air will be controlled by maintaining a lower pressure within the exposure room and the warm storage room than in the remainder of the building. The remainder of the building will, in turn, have a slightly lower pressure than that found outside. Accidentally contaminated air is thus channeled through the filters.

A stack monitoring system activates alarms if undesirable quantities of radioactivity are being exhausted to the atmosphere. A smoke sensor is also provided, which activates alarms if smoke is detected in the exhaust system.

The building is heated by a gas-fired, 225,000-Btu/hr steam boiler and thermostatically controlled steam coils in the air-handling system.

WATER PURIFICATION AND COOLING SYSTEM

The coolant water for the reactor is purified and cooled in an external system (see Fig. 7), which consists of a mixed-bed demineralizer, heat exchanger, pump, and associated piping and valves. The system also includes a surface skimmer, a fission-product monitor, a fiber cartridge-type filter with pressure gauges, and a flow meter. The cooling capacity of the water system is 100 kw at water temperature of 90°F. The water conductivity is kept at about 2 μmho to minimize corrosion. The purification system removes radioactive ions or particles from the reactor water and helps to maintain optical clarity of the water.

The heat exchanger is a conventional shell-and-tube type. The shell and cover are made of carbon steel, and the tubes and all other parts in contact with the reactor water are made of stainless steel. The drop in water temperature across the heat exchanger is approximately $10^{\circ}F$, at a flow rate of 80 gpm. The water is circulated through the primary system by an aluminum centrifugal pump, which provides a head of approximately 70 ft. The secondary coolant enters at $70^{\circ}F$ and exits at approximately $80^{\circ}F$ during 100-kw operation. The design temperature for the shell side of the heat exchanger is $100^{\circ}F$; for the tube side it is $120^{\circ}F$. The design pressure for both sides of the exchanger is 75 psig.

The primary function of the demineralizer system is to maintain a water-conductivity level low enough to minimize fuel-element corrosion. The demineralizers are of a mixed-bed type that removes both positive and negative ions from the circulating water. The system includes two demineralizers, each of which has a flow capacity of 10 gpm and contains about 3 ft^3 of resin. The type of resin provided is a mixture of nuclear-grade Permutit A-H and Permutit S-1. The flow through each demineralizer is regulated by a flowmeter that is located on the downstream side.

A filter removes insoluble particulate matter from the reactor water system. It has a replaceable fiber cartridge, which is removed from the vessel and replaced when its pressure drop becomes excessive. Two filter cartridges l each rated at 10μ, remove all of the particles down to 1μ in size.

Two pressure gauges in the filter bypass line measure the pressure drop across the filter and indicate when the filter should be changed.

WARM-WASTE DISPOSAL SYSTEM

A warm-waste system which collects water from the floor drains in the warm-storage room, decontamination area, exposure room , lavatories and both showers, has a detention capacity of 15,000 gal. The system utilizes three 5,000-gal detention tanks, connected in parallel. Warm waste can be directed to any of the three detention tanks, and can later be discharged from any of the three tanks as the radioactivity drops within the acceptable range.

The warm-storage tanks will be sampled and analyzed for radioactivity, in order to ensure that the warm waste may be flushed into the sanitary sewer. Figure 8 shows the flow diagram of the waste-disposal system

DECONTAMINATION PROVISIONS

A special paint capable of withstanding a decontamination procedure is applied to the concrete walls, floor, and ceiling of the warm storage room. This paint is also applied to the concrete around the truck access area. The main operating floor and the mezzanine are covered with vinyl tile, and the interior walls are sealed with epoxy.

The exposure room is lined with 1 ft of structural-grade wood to reduce activation of the concrete. The floor in the exposure room is covered with a waterproof epoxy surface. The walls and ceiling are covered with a special decontamination paint.

Drains in the building are arranged so that all water or other fluids used for decontamination are channelled into the warm-waste disposal system. Should personnel be contaminated, an emergency shower is located in the toilet, and in the warm-storage room in the basement are located an emergency shower, eyewash, and sink.

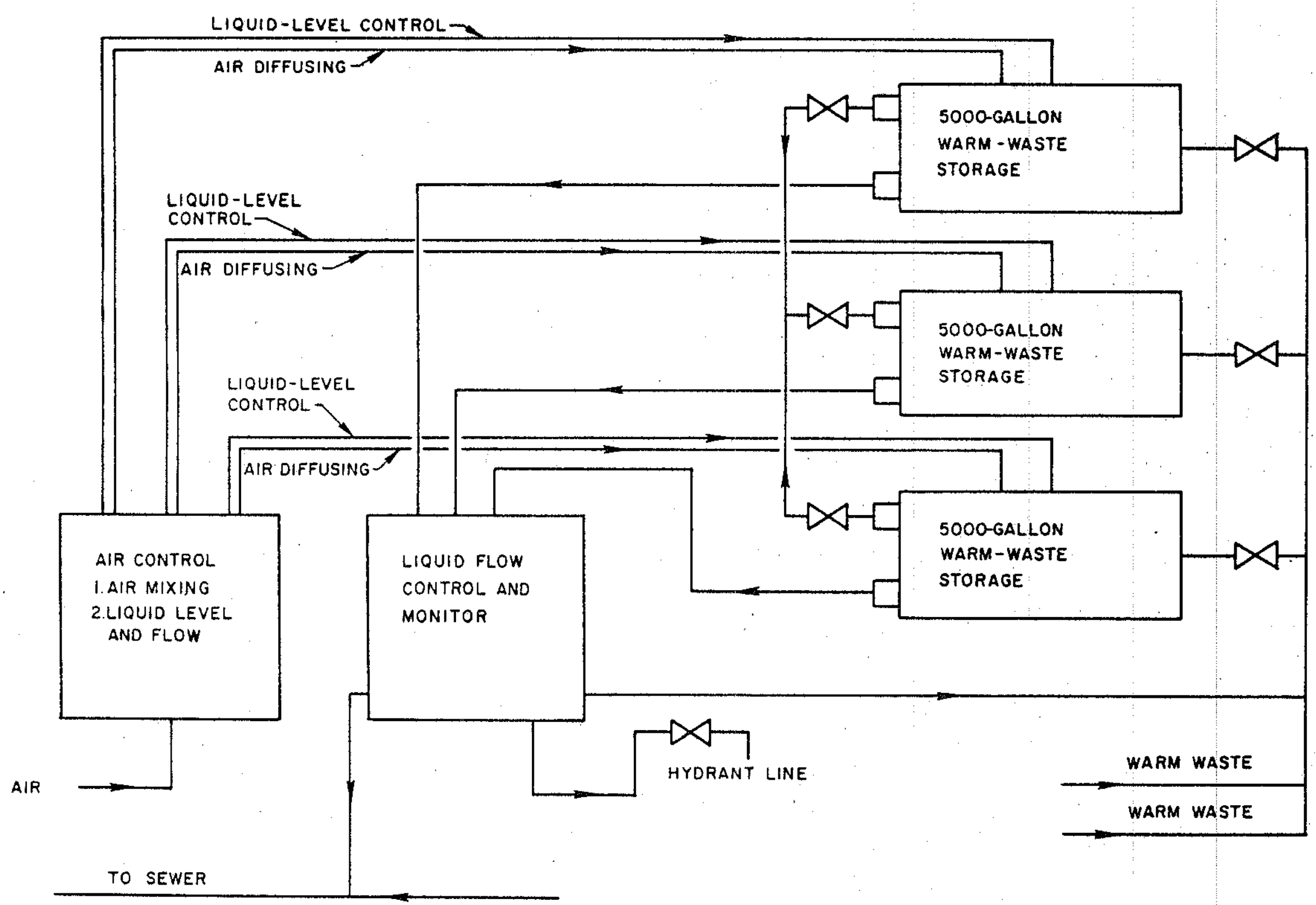


Fig. 8--Warm waste-disposal system for the DORF

UTILITIES

Potable water is piped to the building site through a 2-in. water line, which is approximately 450 ft long. It is estimated that the maximum normal water usage will be 1,200 gpd. An 8-in. sanitary-sewer service runs within 80 ft of the building. However, in order to meet sewer-line inverts and to accommodate the warm-waste detention system, the sanitary sewer from the building intersects the 8-in. sewer line at a manhole approximately 300 ft northwest of the reactor building.

Electric power will be purchased from the Potomac Electric Power Company at 120/208 v. Power is carried from Brookville Road on 13.2-kv overhead service lines to a power-company-owned substation located beside the reactor building. A total connected load of 195 kw indicates that an average estimated demand load of 110 kw will be required at the building.

Natural gas will be piped from the gas main approximately 450 ft through a 20-in. line. It is estimated that during the heating period 225 ft^3/hr will be required.

A 6-ft high chain-link fence with three rows of barbed wire will completely enclose the reactor building and will provide a 200-ft clear space on all sides of the building.

Access to the reactor building is through a 16-ft wide roadway that intersects the driveway from Brookville Road. A parking lot will accommodate nine cars and three 40-ft long trailers with tractors. Truck access to the building is at the basement level and at the operating-floor level.

SOLID-WASTE DISPOSAL

Disposal of solid waste will be handled through arrangement with Walter Reed Army Medical Center.

Chapter 3

REACTOR

DESIGN CRITERIA

The reactor design (see Figs. 9, 10, and 11) presented herein employs the TRIGA reactor fuel elements to provide a facility where studies may be undertaken of the effects of large pulses of neutrons and gamma rays on electronic and other devices of interest to the Diamond Ordnance Fuze Laboratory. The reactor core will have a minimum reflector so that large numbers of neutrons may be emitted by the core into a shielded exposure room. The reactor will have a 2-in.-thick water reflector, covering 180° of the periphery, through which will extend a small, 3 in. by 10 in., air-filled aluminum thimble when the reactor is adjacent to the exposure room. The thimble will be used for the irradiation of very small objects. The core will have the capability of being pulsed by the insertion of up to 2.2% $\delta k/k$ excess reactivity with no hazard to the reactor or the operating crew. Detailed operating characteristics are given in Chapter 5.

Easy access to the core and adequate cooling and shielding will be provided by the large pool of water in which the reactor-core assembly is submerged. Cooling will be provided so that the reactor may be operated at 100-kw steady state and at 250 kw, not to exceed 1 Mw-hr per day, to allow for extended irradiations. To make maximum use of the reactor, the facility will be arranged so that exposure of samples may occur either in the dry exposure room or within the reactor pool. To facilitate this utilization, the reactor will be transported by a carriage from one exposure position to the other within the pool in approximately 6 min. Shielding will be provided so that access to the exposure room may be made while experiments are being conducted in the pool.

Operation and maintenance requirements will be minimized by

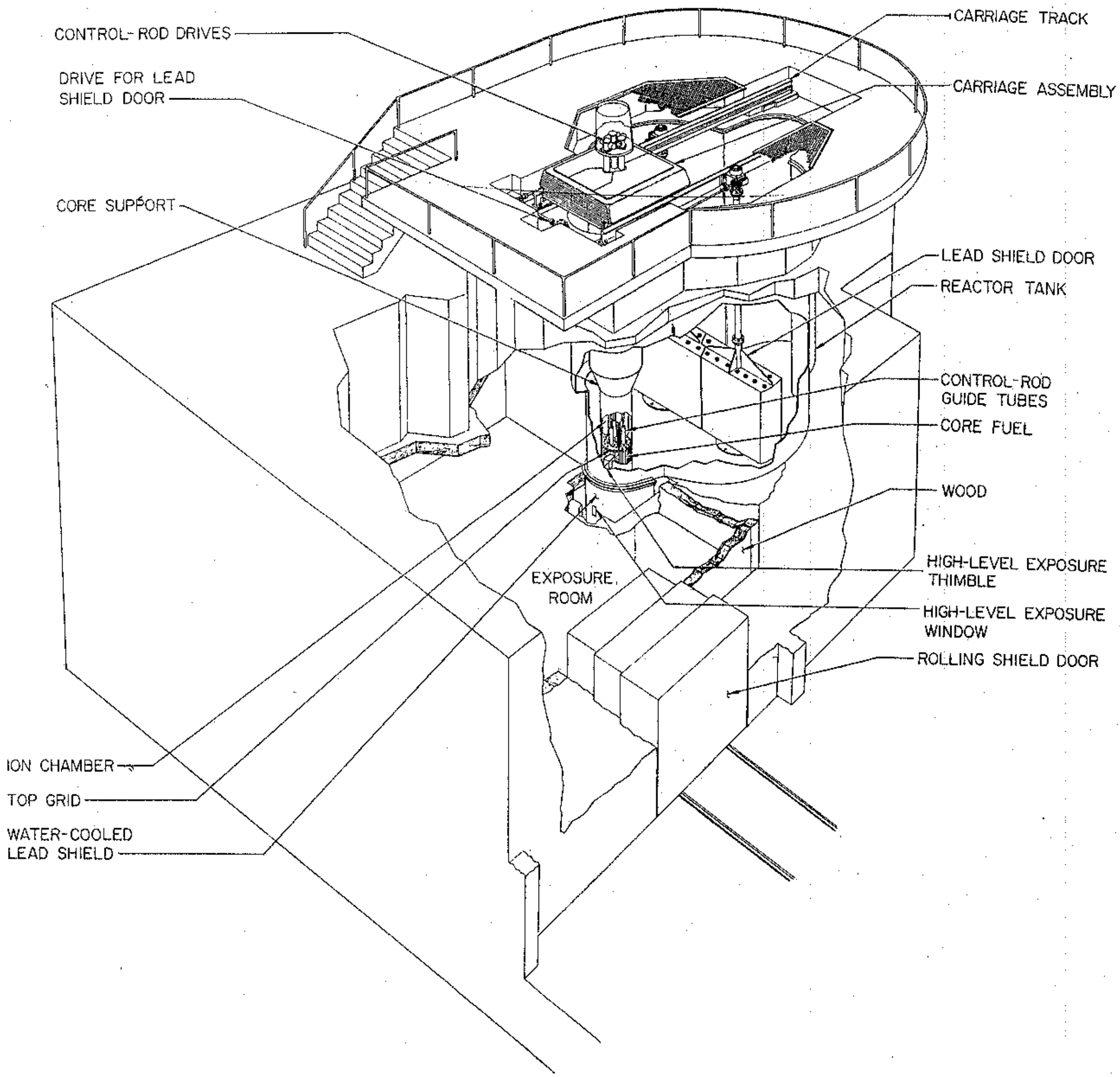


Fig. 9--Perspective view of the reactor

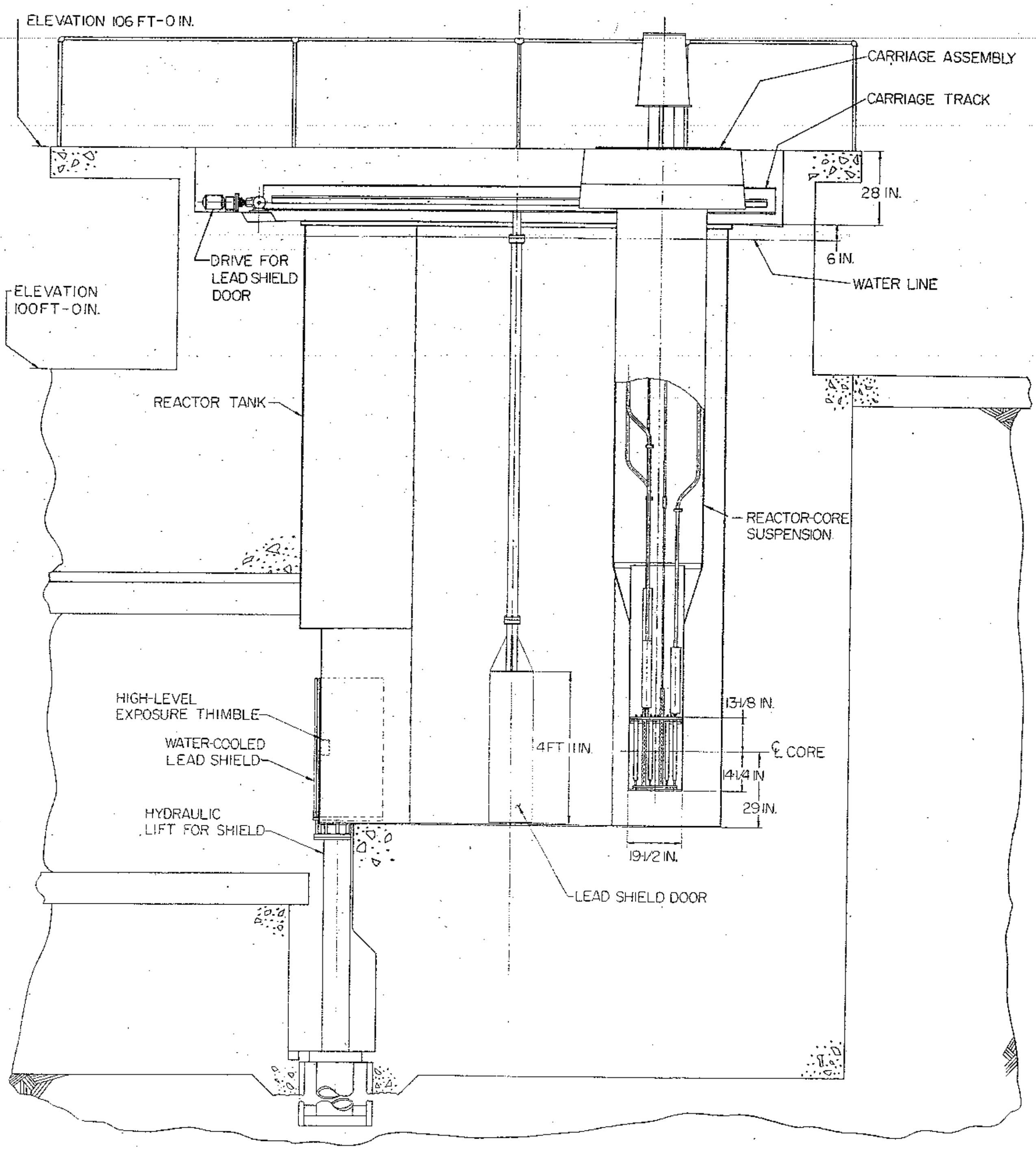


Fig. 10--Sectional elevation of the reactor

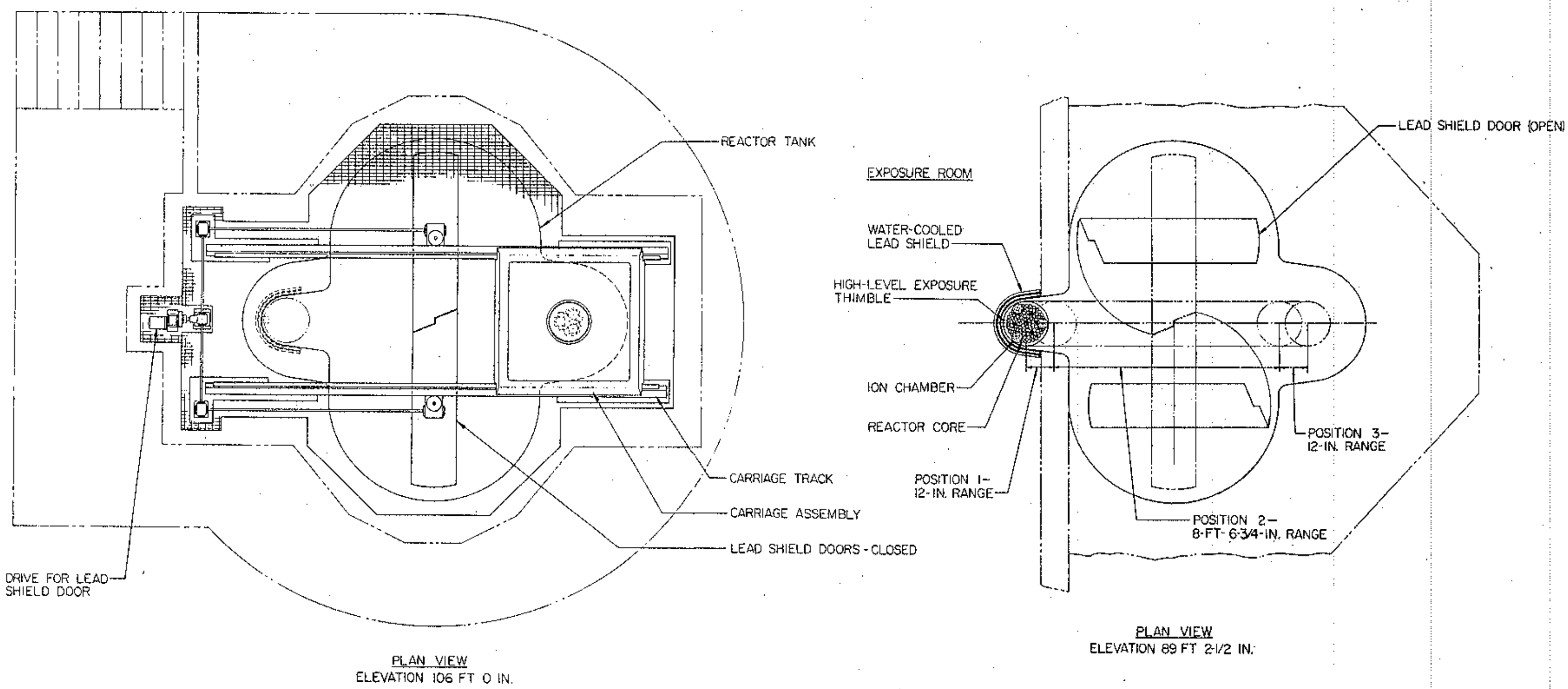


Fig. 11--Plan view of the reactor

employing a simplified control system. To assure safe operation, the reactor will be provided with a reflector of water sufficient to ensure that no change in reactivity larger than 0.5% $\delta k/k$ occurs as a result of moving the reactor from one position to another or of the addition of neutron and gamma-ray filters adjacent to the core. Adequate shielding will be provided for the reactor core and the exposure room so that operating personnel in areas external to the shielding will not receive doses of radiation exceeding one-tenth of that specified by the U.S. Atomic Energy Commission during 100-kw steady-state operation. The neutron-to-gamma-ray ratio will be variable by a factor of 10 by the addition of lead around the core as a gamma-ray filter or by water as a neutron filter.

When the reactor is moved into the exposure-room position, the water reflector is diminished to a 2 in. thickness around half of the core. In addition to this, there is a 1/4-in.-thick aluminum thimble, which is fabricated as part of the reactor tank, that penetrates through the 2-in. water reflector and into the F-ring of the reactor core. In the region where the thimble penetrates, two fuel elements are removed to accommodate the penetration. The closest approach of the thimble to a fuel element is 0.7 cm.

A 2-in.-thick water-cooled lead shield is provided, which is mounted external to the tank on a hydraulically operated piston. This lead shield has an opening in it which allows access to the thimble. The opening can be filled with a removable lead plug. Operation of the movable lead shield is interlocked so that the shield cannot be moved while the reactor is in operation.

The exposure room will be accessible through an opening through the shield containing a movable concrete shielding plug. Electrical interconnections will be accommodated through the roof of the exposure room by means of spiral conduits. These conduits are surrounded by lead to ensure the integrity of the shield.

The communication systems for the reactor and associated exposure room will include both audio and visual systems. A closed-circuit television system will be employed to monitor personnel and experiments in the exposure room and warm-storage area. An audio system will be employed to facilitate maintenance and coordination within the facility.

GENERAL ARRANGEMENT OF REACTOR

The reactor core forms a compact cylinder and consists of a lattice of approximately 85 cylindrical fuel-moderator elements, 4 control rods, and 1 neutron-source holder contained between the top and bottom aluminum grid plates and surrounded by an aluminum shroud which supports the grid plates. This assembly is located at the bottom of an aluminum reactor tank that is approximately 14 ft in diameter and 19 ft 6 in. high and holds 12,000 gallons of water. The core assembly is suspended by a support structure from a motor-driven carriage which is at the top of the tank and is capable of traversing the tank.

Control-rod drive mechanisms are located on the carriage and are connected to the control rods in the core. The reactor is controlled by two safety rods, a regulating rod, and a shim-safety-transient rod. Instrumentation is provided to monitor, indicate, and record the neutron flux. Three modes of operation are possible: Mode I--steady-state operation, with manual or servo control to 100 kw; Mode II--power square wave to 250 kw, maximum; Mode III--flashing operation to 2200 Mw.

In addition to the reactor-control instrumentation, an interlock system is provided to prevent reactor operation unless prescribed safety conditions have been met.

The reactor core is cooled by natural convection and the pool water is purified and cooled in an external system which consists principally of a water-to-water heat exchanger, a mixed-bed demineralizer, a pump and associated piping, valves, and flow-indicating devices. The secondary water to the heat exchanger is circulated through a filter and cooling tower.

In addition to the shielding provided by the water in the reactor tank, two 18-in.-thick shielding doors are located in the tank to shield the exposure room. These are mounted on bearings on the floor of the tank and can be rotated to allow passage of the core from one operating position to another when prescribed safety conditions have been met.

FUEL-MODERATOR ELEMENTS

A fuel-moderator-element assembly is shown in Fig. 12. Including the top and bottom aluminum end-fixtures, the fuel-moderator element is 28.44 in. long. The fuel part of each element, which is 1.42 in. in diameter by 14 in. in length, consists of an alloy of uranium-zirconium hydride containing 8 wt-% uranium enriched to 20% in U^{235}. The hydrogen-to-zirconium atomic ratio is approximately 1.0. A thin aluminum wafer at each end of the active fuel section contains a burnable poison. By this means an appropriate amount of burnable poison is incorporated in each fuel element at the time of fabrication to minimize the loss-of-reactivity effect due to fission-product poisoning and fuel burnup. Four-inch sections of graphite in the fuel can above and below the fuel region serve as top and bottom reflectors for the core. The fuel elements are clad with 0.030-in.-thick aluminum, and all closures are made by heliarc welding. An aluminum end fixture is fixed to each end of the can for positioning and handling. Each standard fuel element contains approximately 37 g of U^{235}. Partially loaded fuel elements containing lesser amounts of U^{235} are used, as necessary, to make available the exact amount of excess reactivity required in the reactor.

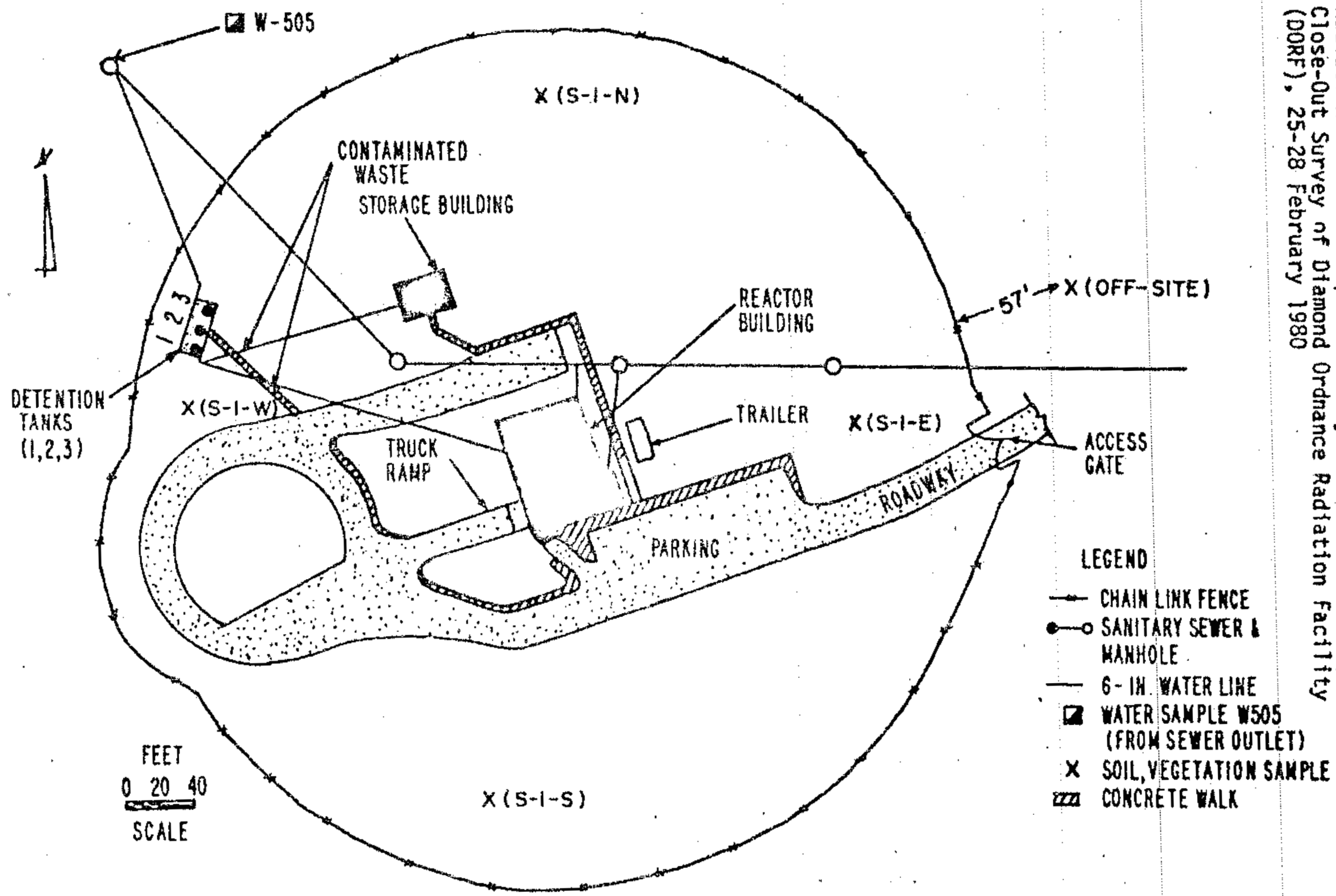
GRID PLATES

The fuel elements are spaced at the top and bottom by two 0.75-in.-thick aluminum grid plates. The grid plates have a total of 91 holes, or fuel-element spaces, 85 of which are available for fuel-moderator elements and the remainder for the 4 control rods. The bottom grid plate, which

HSE-RH/WP
SUBJECT: Radiation Protection Special Study No. 28-43-0982-80, Close-Out Survey of Diamond Ordnance Radiation Facility (DORF), 25-28 February 1980

Incl 3

2



THE DORF SITE
SOIL SAMPLES AND WATER SAMPLE (W-505)

Figure 1

Report# GA-5786

8. PRESSURE PULSE IN REACTOR PIT

Several sets of pressure measurements have been made in the water in and around the reactor core.

As the magnitude of reactivity insertion was increased from the starting point of $3.00, a slight acoustic output was observed shortly after a pulse of $3.60. At this time, a series of pressure measurements in and about the core was started along with vibration measurements to attempt to determine the source and its magnitude. The experiments run were extensive and in some cases gave inconclusive results. Of these, the pressure measurements in the pit water have been most troublesome, primarily because of the very small magnitude of the pressure waves. Most satisfactory of the measurements in terms of reproducibility of results has been the measurement of floor vibration by means of geophones. These measurements are related, however, to the observed acoustic effects, but provide no numerical information about the source of the signals.

The experiments conducted, the results, and conclusions drawn are given below.

.1 IN-CORE PRESSURE MEASUREMENTS

Upon observing an audible acoustic signal following a $3.60 pulse, series of in-core pressure measurements was begun. The transducer rrangement used is described in Section 4.5. This device allowed measrements of pressure signals transmitted through the 18-inch-long, .4-inch-diameter transmission tube that was inserted in a C-ring posion. For each measurement, the tube was water-filled to provide good coustic coupling. With the device mounted in the C-ring position, the ater region adjacent to B-, C-, and D-ring fuel elements was immediately elow the transmission tube.

A series of pulses was performed starting at step insertion of 3.00 and going up to $3.50. The output of the pressure transducer was ecorded along with a reactor power signal and timing markers to provide means of placing any signals in time reference to the reactor pulse.

The results of these first tests were a series of recordings of tran-
icer output signals, each of which showed a recorder deflection correspond-
g to a pressure signal of 3- to 4-psi magnitude. The polarity of the signal,
owever, indicated a negative-going pressure followed by a positive pressure
approximately equal magnitude. Checks were made of the possibility that
is was simply a pressure signal arising from upward acceleration of the
iter-filled acoustic transmission tube coupled to the upper grid plate. This
is found to be the case. Impact of the transient rods provided sufficient
celeration to the core to give rise to the observed acceleration signals.

Further tests using this transducer assembly have failed to show
y measureable pressure originating in the core region of sufficient
ignitude to override or distort this acceleration signal. The results
these measurements have led to the conclusion that pressures originating
-core due to pulsing up to a $4.60 step insertion are small compared with
osi and therefore pose no problem.

A further test of the possibility of transient pressures within the
re was made by means of a strain-gauge-instrumented aluminum fuel
ment can set up to record lateral deflection of the tubular structure.
evious to use the assembly was calibrated by dead weight loading to deter-
a force calibration for the assembly. Measurements were made to test
ation response of the assembly by suspending it only part way down
ough the upper grid plate and subjecting it to reactor pulse irradiation.
response was observed. The assembly was then lowered fully into the
e so that it was positioned and supported exactly like any fuel element.
ses were performed for $3.00 insertions and the output recorded. No
nal was observed that could be attributed to a lateral differential pressure
large as 1 psi.

OUT-OF-CORE MEASUREMENTS

Results similar to those above have been obtained with pressure
isducers to measure transient pressure in the pit outside the core.
isurements have been made at two positions outside the core region. One
isurement was made with the transducer positioned to observe signals
ismitted through the rectangular opening in the side of the core support
oud; in the other position the transducer was placed below the lower grid
e and oriented to observe any pressures from the cooling holes in the
er grid plate.

In the case of the measurements made at core center line outside the
angular opening, measurements of pressure signals as a function of dis-
e from the core showed no sharp fall-off in magnitude with distance as
expected but in all cases did show an increasing pressure amplitude
ansducer-core separation was increased. Measurements of pres-
es from the bottom of the core for reactivity insertion up to $4.60 showed

ıo signals which could be attributed to pressures as large as 1 psi. Vari-
ous "noise" signals were observed arising from transient rod motion and
act; however, these did not change in magnitude with reactivity inser-
ɔr peak power reached. Since no pressure distance relationship could
ɔe established which suggested a pressure signal from the core, it was con-
:luded that here, as for the in-core measurements, the magnitude of pres-
ure signals originating in the core is small (< 1 psi external to the core).

.3 FLOOR VIBRATION MEASUREMENTS

In order to obtain data related to the observed acoustic output, geo-
hones were attached to the reactor room floor near the edge of the pit.
'he units used were standard Hall-Sears geophones with 4.5-cps natural
esonant frequency. Units of this type are quite sensitive to vibration be-
ause they are velocity-dependent in output and are of moving-coil design
ith a large number of turns.

In use the geophones were tied directly into the input of the high-
peed recorder. In the arrangement used, the sensitivity allowed observa-
on of actuation of the air solenoid in the transient rod system as well as
npact of the rod on the bridge mount. Simultaneous recording was also
ade of the power to provide a time reference for the signals.

With this system, good reproducible data were obtained that corre-
d well with the observed acoustic output.

EFFECT OF INTRODUCING AN AIR VOID IN REACTOR PIT

Data obtained with the floor-mounted geophones showed a rather
arp (~ 15-cps frequency) vertical motion of the floor structure beginning
0 to 400 msec after the reactor pulse, followed by another similar signal
0 to 400 msec after the first. Often there would be a third and fourth
ɔration. The problem was suggestive of the result that occurs when
o wave trains of slightly different frequency interact.

A series of experimental runs was made with the large air-filled
adiation void in the pit about 18 in. from the reactor core. Pulsing under
se conditions gave a completely different floor vibration pattern. The
ries of spikes was eliminated, and a smooth vibration pattern of greatly
luced amplitude was recorded.

It would seem clear from these data that the wave train interference
lanation is reasonable and that the acoustic and vibrational phenomena
served are primarily a result of mechanical properties of the pit and
ɔr structure. The vibrations have presented no operational problems

er 4900 pulses up to $3.00. No measurable effects on core or components een found after over 600 pulses at $4.00 and above.

5 EFFECT OF ISOLATING REACTOR FROM FLOOR

To verify that the observed vibration was being transmitted through e water, the reactor carriage was blocked up on rubber mounts. With the rriage isolated from the floor, the pattern of floor vibrations was unchanged, dicating that the coupling to the reactor pit is by way of the water rather than way of the core support structure.

ERG-ED

24 March 1970

Mr. Turovlin/mo/46467

Mr. Frank Wimenitz
U. S. Army Material Command
Harry Diamond Laboratories
Washington, D. C. 20438

Dear Mr. Wimenitz:

References: Letter from General Atomic to HDL dated May 10, 1967.
Excerpt from General Atomic Report GA 5786.

Upon your informal telecon request, the Engineering Division, U.S. Army Engineer Reactors Group evaluated the problem of collapse of the protruding tank structure of your TRIGA reactor. The primary area of concern was the hydraulic pressure pulse upon this structure, due to the energy pulsing of the reactor.

The results of the evaluation have already been discussed with the HDL operating group, and this is to record the evaluation. The problem was evaluated in three aspects.

1. Reviewing the calculation done for the AFRRI TRIGA reactor; which was manufactured by the same vendor and is very similar to HDL TRIGA. A copy of the stress calculations done by the Naval Facilities Engineering Command, Chesapeake Division for the Armed Forces Radiobiological Research was provided to USAERG by your operating group.

2. Telecon with Gulf General Atomic, (GGA) Mr. Ralph Peters, reviewing experimental data developed by GGA. Also discussed with Mr. Peters was the referenced letter and the excerpt from GA 5786, copies of which were supplied to USAERG by HDL operating group.

3. Hand calculation using a simplified model and assuming uniaxial stress.

ERG-ED
Mr. Frank Wisenitz

24 March 1970

The conclusion after this review is that the energy pulsing of the reactor does not develop, by at least an order of magnitude, a large enough hydraulic pressure pulse to approach the collapse strength of the protruding tank structure. The possibility of severe mechanical damage with subsequent collapse does exist however if the protruding structure were struck a very heavy blow by a sharp edged heavy weight such as a cask.

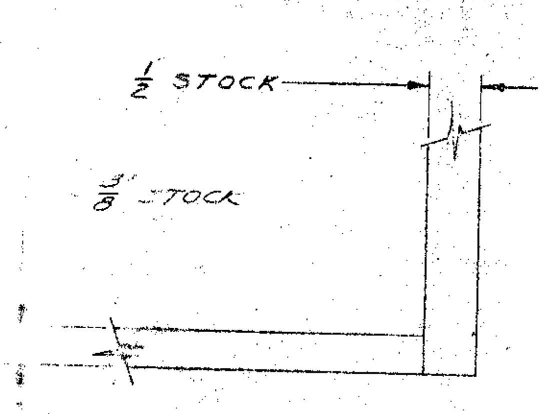
Sincerely yours,

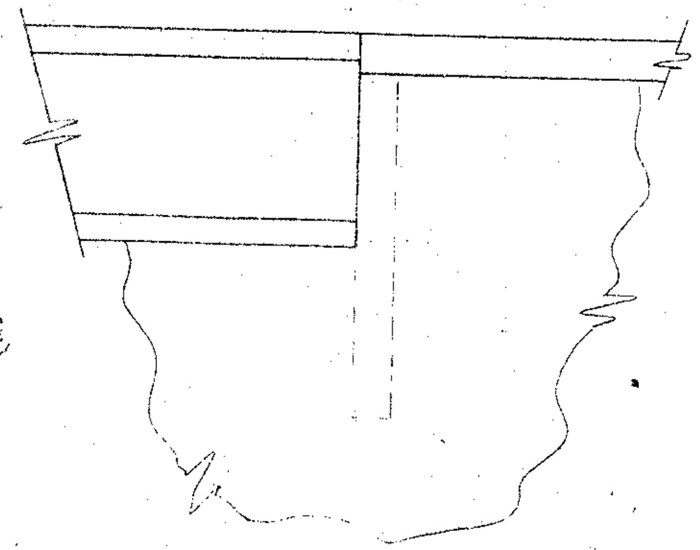
MICHAEL A. STOLLMEYER

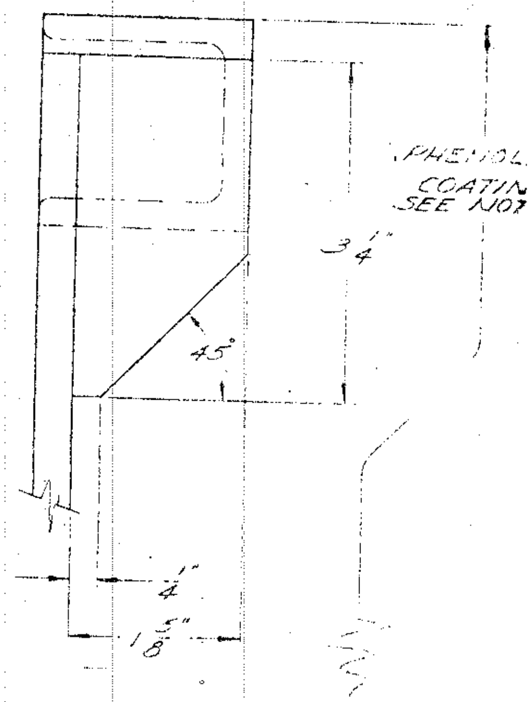
MICHAEL A. STOLLMEYER
Chief
Engineering Division

cc: corres
reading
stnh
PS Br Rdg
PS Br (Turovlin)
ED Files (PS Br)

FULL SCALE







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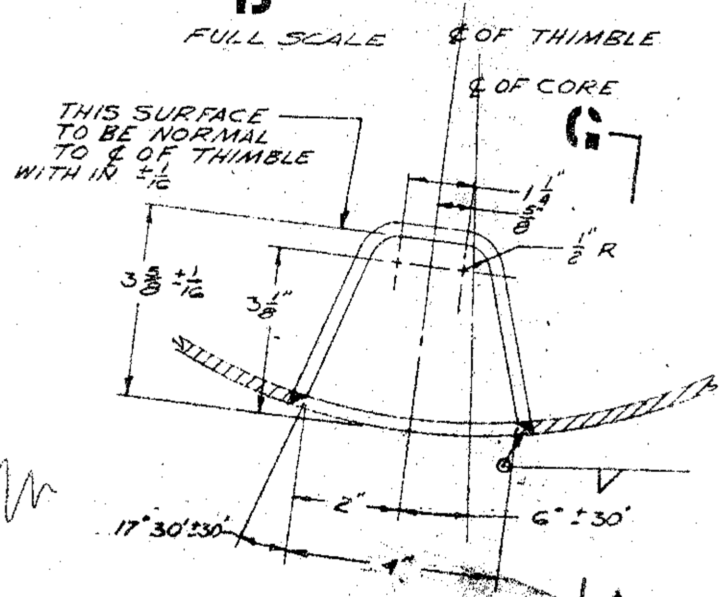
FULL SCALE



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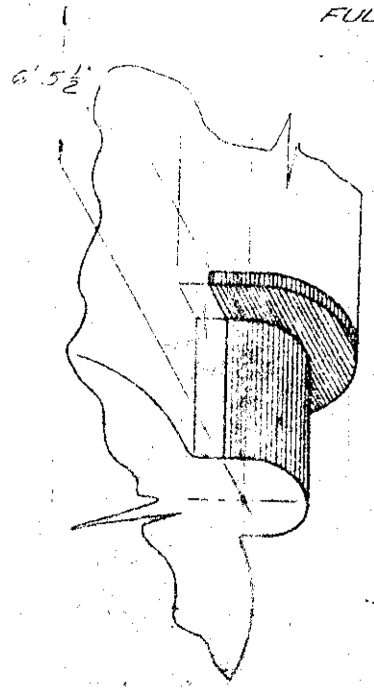
FULL SCALE



F **G**

HALF SCALE

NOTE

1. MAKE PER GENERAL ATOMIC MANUFACTURING SPECIFICATION GATR-M-6.
2. OUTSIDE COATING - BY OTHERS - PER GENERAL ATOMIC SPECIFICATION GATR-P-1.

3 SURFACE MARKED "X" TO BE PARALLEL WITH ℄ OF TANK WITHIN ±1/8".

4 SCRIBE ℄ 2"-3" LONG ON TOP AND BOTTOM OF TANK SIDES, AND ON TOP FLANGE - 4 SIDES.

GENERAL ATOMIC
DIVISION OF GENERAL DYNAMICS CORPORATION

P. O. BOX 608, SAN DIEGO, CALIFORNIA 92112

May 10, 1967

Mr. Frank Wimenitz
U. S. Army Materiel Command
Harry Diamond Laboratories
Washington, D. C. 20438

Dear Mr. Wimenitz:

This letter discusses the basis for General Atomic's recommendation that the HDL tank nose should be reinforced at the exposure room when this can be done conveniently, such as during modifications to the facility which require draining of the tank.

Following the original design and installation of the HDL and AFRRI reactor tank, another similar reactor tank was designed and fabricated. Independent reviews of this design were made by the stress analysis group at General Atomic and by the structural engineering group at Holmes & Narver, the A & E firm responsible for the detailed tank and shield design.

Like the HDL and AFRRI tanks, this tank is loaded in compression vertically in the area of the exposure room and the same section is subjected to a large horizontal load. Considering the reactor tank wall as a series of vertical strips or columns, the two loads act to complement each other. A sharp blow in the area of the 4-inch bend radius, where the tank wall is curved to form the exposure room nose section, could result in a progressively greater deformation until the tank wall failed by buckling. As a result of these reviews, additional bracing was added to the tank.

Although the HDL and AFRRI reactor tanks are loaded in a manner which might give rise to an unstable condition, there should be no cause for concern during normal operation. However, it is conceivable that the combination of the existing tank loading plus an additional horizontal load occasioned by a severe bump inside the tank (such as a fuel handling cask hitting the tank side near the exposure room) might produce a condition wherein the tank nose is elastically unstable.

As a means of providing additional safety, the three tanks designed and fabricated since the HDL and AFRRI tanks have incorporated additional bracing in the exposure room. In the case of the HDL and AFRRI reactors, a different method of strengthening the tank would be used. This would consist of welding a preformed aluminum plate around the critical 4-inch radius bend area. The strengthening plate would be about 3/4-inch thick, extending approximately 8 inches beyond either side of the bend from near the tank bottom to near the top of the exposure room nose section.

General Atomic strongly recommends that these reinforcings be installed at the time the reactor is upgraded to incorporate higher pulsing capability. We will be pleased to discuss this further if you desire.

Sincerely yours,

Albert P. Graff

Albert P. Graff, Manager
TRIGA Reactor Program

P = pressure causing buckling

$$p = \frac{E\left(\frac{t}{r}\right)}{1+\frac{1}{2}\left(\frac{\pi r}{nl}\right)^2}\left\{\frac{1}{n^2\left[1+\left(\frac{nl}{\pi r}\right)^2\right]^2} + \frac{n^2 t^2}{12 r^2 (1-\nu^2)}\left[1+\left(\frac{\pi r}{nl}\right)^2\right]^2\right.$$

$E = 10{,}000{,}000$

$t = 0.25''$

$r = 4.00''$

$n = 2$

$l = 78''$

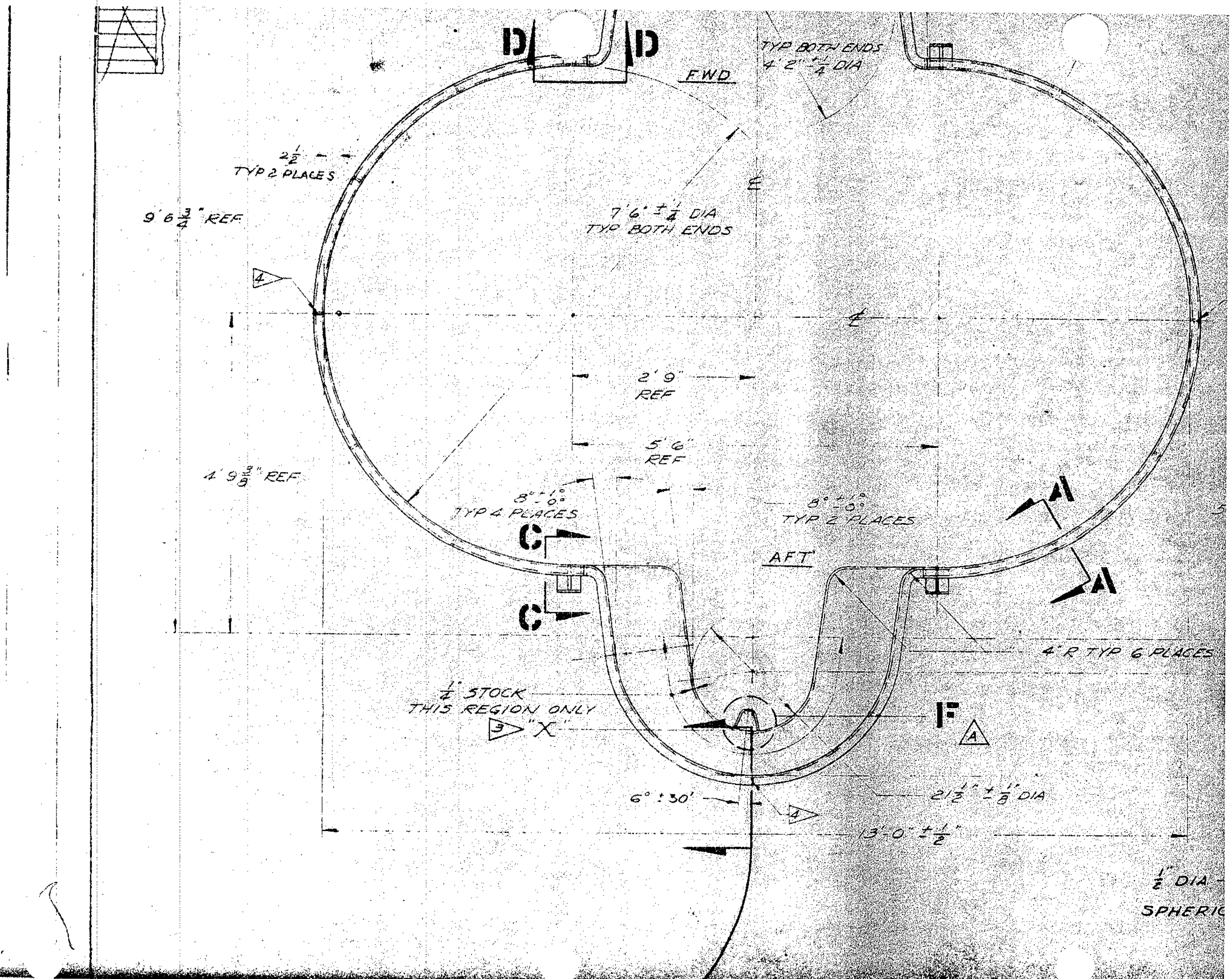
$\nu = 0.33$

n = number of lobes formed by buckling

$p = 928$ psi when $n = 2$

$= 2056$ psi when $n = 3$

$= 3800$ psi when $n = 4$



D D
TYP BOTH ENDS
4'2" ±1/4 DIA
FWD
2 1/2
TYP 2 PLACES
9'6 3/4" REF
7'6" ±1/4 DIA
TYP BOTH ENDS
4
2'9"
REF
5'6"
REF
4'9 3/8" REF
8° +1° -0°
TYP 4 PLACES
8° +1° -0°
TYP 2 PLACES
C
C
A
A
AFT
4" R TYP 6 PLACES
1/2" STOCK
THIS REGION ONLY
3 "X"
F
A
21 1/2" ± 1/8" DIA
6° ± 30'
4
13'-0" ± 1/2"
1/2" DIA
SPHERIC

ENGINEERING SERVICE REQUEST
NAVDOCKS 2038 (Rev. 2-61)

Submit in Quadruplicate

(98)

1. Copy No.

2. FROM

Commanding Officer, National Naval Medical Center, Bethesda, Maryland 20014

3. TO

Commanding Officer, Chesapeake Division, Naval Facilities Engineering Command

4. REFERENCES

5. ASDR OR SPECIAL PROJECT NO.

6. ENCLOSURES (Check)

☐ NAVCOMPT 140 ☐ NAVCOMPT 2038 ☐ NAVCOMPT 372 ☐ OTHER (Explain)

7. TYPE OF SERVICES REQUESTED

Engineering Study and Cost Estimate

8. DETAIL OF WORK

It is requested that an engineering study be performed to include analysis, recommendations and cost estimate for reinforcing the reactor tank in Building #42 - Armed Forces Radiobiology Research Institute.

The stress analysis of the reactor tank is also requested in this study.

1505-1 Reinforce Reactor Tank

9. FOR INFORMATION CONSULT (Name, title and phone)

J. Louis Gable

J. LOUIS GABLE, Gen'l.Eng. (Code 19 (51231

10. OFFICIAL REPRESENTATIVE (Signature and date)

H. A. FALK, JR., By Direction 9/27/6

RESERVED DPWO ENTRIES ONLY

11. DATE RECEIVED IN DPWO	12. SERVICES TO BE PERFORMED BY DPWO (Check)	13. DPWO PROJECT NO.
3 Oct 1967	☐ DESIGN ☐ OTHER (Explain)	NNMC-11-6

14a. SCOPE OF CONTRACTS

14b. NOY OR OTHER NO.

15. OTHER

DPWO INTERIM ENDORSEMENT

16a. ESTIMATED COMPL. DATE	16b. AUTHORIZED REPRESENTATIVE (Signature and date)
12-29-67	W. F. POTTER, Asst. to Acq. Coord. Officer 10-11-67

DPWO FINAL ENDORSEMENT TO ORIGINATOR

17a. ENCLOSURES FWD'D

☐ DWGS. & MAPS ☐ SPECS. ☐ REPORT ☐ OTHER:

17b. EST. COST (If applicable)	17c. AUTHORIZED REPRESENTATIVE (Signature and date)
	W. F. POTTER, Asst to Acq Coord Officer 1-5-68

FINAL ENDORSEMENT

Subject: Engineering Service Request, NNMC-41-67 of 11 October 1967

Encl: (1) Computations for Light Water Tank, Triga Mark F Reactor, Building #42, Armed Forces Radiobiology Research Institute

1. The subject tank was examined and found free of any signs which might indicate possible failure. There was some spalled concrete at the edge of the support pedestal which could easily be patched with an epoxy concrete.

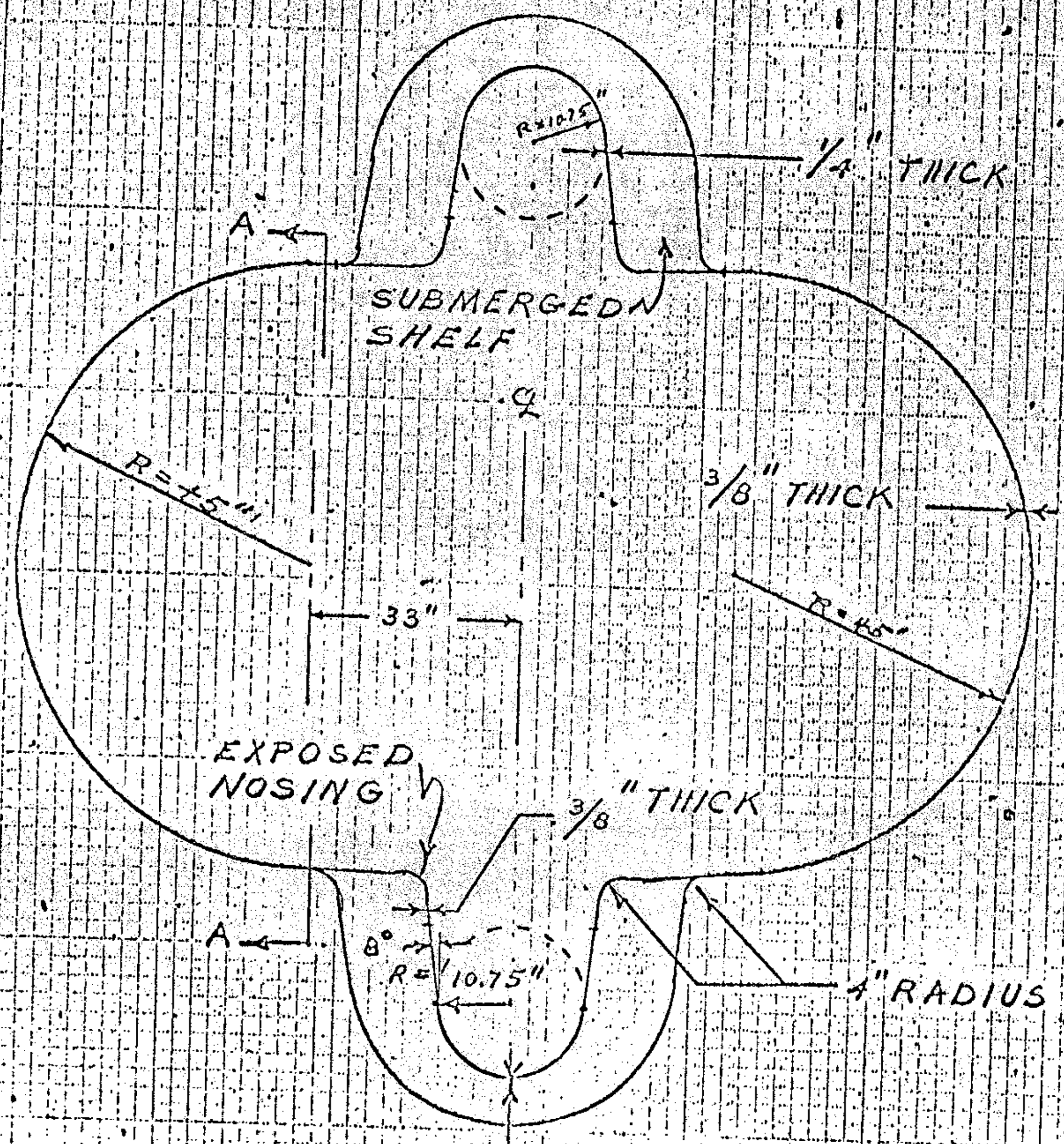
2. The main shell of the tank had a factor of safety of thirty-five against the yield strength of the material.

3. The exposed nosing was analyzed as a portion of a cylinder supported along two generators and along the two circular edges and subject to compression in the direction of the generators as well as lateral uniform pressure. The failure mode for this figure would be elastic buckling. The estimated ratio of existing pressure against buckling is on the order of at least twenty. This represents a substantial safety factor.

4. The possibility of a severe bump inside the tank (such as a fuel handling cask hitting the tank side near the exposure room) should be eliminated. This can be done by putting limit switches into the controls of the motorized trolley-hoist. This will eliminate the possibility of the fuel handling cask striking either the top or the sides of the submerged shelf.

5. The weight of the lead shield which rests on the submerged shelf was not included in the computations. It is recommended that this weight be carried by hanger rods supported on the concrete slab floor at the level of the top of the tank.

TRIGA MARK F REACTOR
LIGHT WATER TANK BUILDING #42 AFRRI
DIMENSIONS TAKEN FROM GENERAL DYNAMICS
GENERAL ATOMIC DIVISION DRAWING T3B100J110



TANK DEPTH = 19.5'
DEPTH TO shelf = 13'
SHELF TO BOTTOM = 6.5'

Sec. AA.

Compute Hoop stresses cylindrical section = S

$S_{max} = \frac{p\,r}{2\,t}$ $p = 0.433 \frac{lb}{in^2}$ ft. of ht.

$h = 19'$ = max. depth

$t = 3/8"$

$r = 45"$

$$S = \frac{(0.433)(19)(45)}{\left(\frac{3}{8}\right)}$$

$S = 986 \frac{lb}{in^2} \ll 35,000 \frac{lb}{in^2}$

Factor of safety $= \frac{35,000}{986} \approx 35$ vs yield str

Material Aluminum 6061 T6

Tension

Ult. Strength = 42 ksi

Yield strength = 35 ksi

Shear

Ult. Strength = 27 ksi

Yield strength = 20 ksi

The computation of the hoop stress for Sec. AA ignores the strength of the reinforced concrete walls within which the aluminum tank is contained.

Estimate of Factor of Safety for Exposed Nosing

Theoretical buckling pressure $= p' = 928$ psi

losses due to fabrication errors, stress concentrations, etc. $= 5$ (estimated maximum possib.

$$p'' = \frac{928 \text{ psi}}{5} \approx 185 \text{ psi}$$

p = pressure at bottom of tank

$$p = 0.433 \frac{\text{lb}}{\text{in}^2/\text{ft of ht.}} (19.5 \text{ ft.}) = 8.5 \text{ lb/in}^2$$

Estimated Factor of Safety Against Buckling Expressed as a Ratio Between Buckling Pressure and Existing Pressure

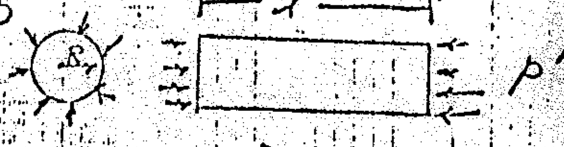
$$F.S. = \frac{185}{8.5} \approx 21$$

FORMULAS FOR STRESS & STRAIN

ROARK cylinder: Uniform pressure $= p'$

p' = pressure causing buckling

ASSUME $N = 2$



$$p' = \frac{E\left(\frac{t}{r}\right)}{1+\frac{1}{2}\left(\frac{\pi r}{nl}\right)^2}\left\{\frac{1}{n^2\left[1+\left(\frac{nl}{\pi r}\right)^2\right]^2} + \frac{n^2t^2}{12r^2(1-\nu^2)}\left[1+\left(\frac{\pi r}{nl}\right)^2\right]^2\right.$$

$E = 10{,}000{,}000$

$t = 0.25''$ $\quad t^2 = .0625$

$r = 4.00''$ $\quad r^2 = 16.0$

$n = 2$ $\quad n^2 = 4$ $\quad n$ = number of lobes formed by buckl

$l = 78''$

$\nu = 0.33$ $\quad \nu^2 = 0.1089$

$$p' = \frac{10(10)^6\left(\frac{.25}{4.0}\right)}{1+\frac{1}{2}\left[\frac{(3.14)(4.0)}{(2)(78.0)}\right]^2}\left\{\frac{1}{2^2\left[1+\left(\frac{(2)(78)}{(3.14)(4)}\right)^2\right]^2} + \left(\frac{(4)(.0625)}{(12)(16)(1-0.1\underset{.891}{}}\right.\right.$$

$$\left.\left[1+\left(\frac{(3.14)(4)}{(2)(78)}\right)^2\right]^2\right\}$$

$$p' = \frac{10(10)^6(.0625)}{1+\frac{1}{2}(.0805)^2}\left\{\frac{1}{4\left[1+(12.42)^2\right]^2} + (.001444)\left[\underset{1.00648}{1+.08^2}\right]^2\right.$$

$$p' = \frac{10(10)^6(.0625)}{1.00324}\left\{\frac{1}{\underset{(964{,}176)}{(4)(241{,}044.3)}} + (.001463)\left[1.0130\right]\right\}$$

$p' = 623{,}000\,\{.0000103 + .001482\}$

$= 623{,}000\,\{.001489\} = 928$ psi = buckling pr

$p' = 925$ psi when $n = 2$

Assuming $n=3$

$$p' = \frac{E\left(\frac{t}{r}\right)}{1+\frac{1}{2}\left(\frac{\pi r}{nl}\right)^2}\left\{\frac{1}{n^2\left[1+\left(\frac{nl}{\pi r}\right)^2\right]^2} + \frac{n^2t^2}{12r^2(1-\nu^2)}\left[1+\left(\frac{\pi r}{nl}\right)^2\right]^2\right\}$$

$$p' = \frac{10(10)^6\left(\frac{.25}{4}\right)}{1+\frac{1}{2}\left[\frac{(3.14)(4.0)}{(3)(70.0)}\right]^2}\left\{\frac{1}{3^2\left[1+\left(\frac{(3)(78)}{(3.14)(4)}\right)^2\right]^2} + \frac{9(.0625)}{(12)(16)(.891)}\left[1+\left(\frac{(3.14)(4)}{(3)(78)}\right)^2\right]^2\right\}$$

$$p' = \frac{(10)(10)^6(.0625)}{1+\frac{1}{2}[.00288]}\left\{\frac{1}{(9)\left[1+(18.63)^2\right]^2} + (.00328)\left[1+.00289\right]^2\right\}$$

$$p' = \frac{10(10)^6(.0625)}{1.00144}\left\{\frac{1}{981,000} + .00329\right\}$$

$$p' = (6.25)(10)10^4\left\{.00329\right\}$$

$$p' = \underline{20.56\ psi}$$

Assuming $n = 4$

$$p' = \frac{E\, t/r}{1 + \frac{1}{2}\left(\frac{\pi r}{n l}\right)^2} \left\{ \frac{1}{n^2\left[1 + \left(\frac{n l}{\pi r}\right)^2\right]^2} + \frac{n^2 t^2}{12 r^2 (1-\nu^2)} \left[1 + \left(\frac{\pi r}{n l}\right)^2\right]^2 \right\}$$

$$p' = \frac{10(10)^6(.0625)}{1 + \frac{1}{2}\left(\frac{(3.14)(4.0)}{(4)(78)}\right)^2} \left\{ \frac{1}{16\left[1 + \left(\frac{(4)(78)}{(3.14)(4)}\right)^2\right]^2} + \frac{(16)(.0625)}{(12)(16)(.891)} \left[1 + \left(\frac{(3.14)(}{(4)(7}\right.\right.\right.$$

$$p' = \frac{625{,}000}{1 + .04025^2} \left\{ \frac{1}{16\left[618.03\right]^2} + (.00607)\left[1 + .001620\right]^2 \right.$$

$$p' = \frac{625{,}000}{1 + .001620} \left\{ \frac{1}{6{,}110{,}784} + (.00607)(1.00324) \right\}$$

$$p' = (623{,}989)(\quad + .00609)$$

$$p' = 3{,}800 \text{ psi}$$

ARK – pg 315 (K) curved panel under uniform compression on curved edges.

$$s' = \frac{1}{6}\frac{E}{1-\nu^2}\left[\sqrt{12(1-\nu^2)\left(\frac{t}{r}\right)^2 + \left(\frac{\pi t}{b}\right)^4} + \left(\frac{\pi t}{b}\right)^2\right]$$

b = width of panel measured on arc $= \frac{\pi D}{4} = \frac{3.14\ 8}{4} = \pi 2 = 6.28$

E = 10,000,000 psi

t = 0.25"

r = 4.00 "

$\nu = 0.33 \qquad \nu^2 = 0.1089$

$$s' = \frac{1}{6}\frac{1\times10^7}{1-.33^2}\left[\sqrt{12(1-.1089)\left(\frac{.25}{4.}\right)^2 + \left(\frac{\pi .25}{6.28}\right)^4} + \left(\frac{\pi .25}{6.28}\right)^2\right]$$

$$S' = \frac{1}{6}\frac{1\times10^7}{.8911}\left[\sqrt{12(.8911)(.0039) + (2.43\times10^{-4})} + (1.5\times10^{-2})\right]$$

$$= \frac{1}{6}\frac{1\times10^7}{8.911\times10^{-1}}\left[\sqrt{12(8.9\times10^{-1})(3.9\times10^{-3}) + (2.43\times10^{-4})} + (1.5\times10^{-2})\right]$$

$$= \frac{1}{6}\frac{1\times10^7}{8.911\times10^{-1}}\left[\sqrt{\left(\cancel{416.9\times10^{-3}}\ 416\times10^{-4}\right) + (2.43\times10^{-4})} + (1.5\times10^{-2})\right]$$

$$= \frac{1}{6}\frac{1\times10^7}{8.911\times10^{-1}}\left[\underbrace{\sqrt{(418.43\times10^{-4})}}_{20.4\times10^{-2}} + (1.5\times10^{-2})\right]$$

$$= \frac{1}{6}\frac{1\times10^7}{8.911\times10^{-1}}\left[21.9\times10^{-2}\right] = \frac{1}{6}\frac{1\times10^7}{8.911\times10^{-1}}\ 21.9\times10^{-1} = \frac{21.9\times10^6}{53.4}$$

$$= 411\times10^6 = 411{,}000\ \text{psi}$$

20% 411,000 = 82 200 psi crit unit stress

HAZARDS SUMMARY REPORT
for
Diamond Ordnance Radiation Facility

Department of Army-Ordnance Corps

Diamond Ordnance Fuze Laboratories

Washington, D.C.

Prepared by

GENERAL DYNAMICS CORPORATION / GENERAL ATOMIC DIVISION

PREFACE

The Diamond Ordnance Radiation Facility (DORF) will utilize an advanced, pulsing type of TRIGA reactor that is to be used primarily for studying the effects of short pulses of very intense neutron and gamma radiations on electronic instruments.

The radiation facility, which is to be operated by the Diamond Ordnance Fuze Laboratories of the Department of the Army at the Forest Glen Annex of the Walter Reed Medical Center, is being designed to utilize a water-reflected version of the TRIGA reactor core. This version is an advanced TRIGA core with a minimum reflector so that large numbers of fast neutrons are available for irradiation studies.

The operating behavior of the DORF-TRIGA has been manifested in a research and development program, including the design, construction, and operation of the TRIGA Mark F Reactor (AEC Docket No. 50-163) located at the General Atomic John Jay Hopkins Laboratory for Pure and Applied Science at San Diego, California.

SUMMARY

The DORF-TRIGA reactor is to be located within the limits of the Forest Glen Annex of Walter Reed Medical Center, which is about one mile north of the city limits of Washington, D. C., in Montgomery County, Maryland. The reactor and attendant facilities will be housed in a building of reinforced concrete.

The DORF-TRIGA has been designed so that the reactor can

- Operate at a steady-state power of 100 kw.
- Operate at 250 kw 4 hr/day.
- Be pulsed with the insertion of up to 2.2% $\delta k/k$ excess reactivity. The pulse will have a duration of approximately 10 msec and an energy release of about 24 Mw-sec.
- Be moved within the reactor tank to permit its use in two separated locations.

The experiments of principal interest include studies of the radiation effects on electrical and electronic components or systems and on metallurgical and chemical specimens. All experiments will be reviewed by the DORF Reactor Safety and Planning Committee to ascertain their effect on the safety of reactor operation. These reviews will be conducted prior to the performance of any experiment.

The principal feature contributing to the safety of the DORF-TRIGA is the prompt negative temperature coefficient of the TRIGA fuel elements, which automatically limits the reactor power to a safe level in the event of a power excursion. Step insertions of reactivity as large as 3.1% $\delta k/k$ have been performed without hazard, and these reactivity insertions have confirmed the inherent stability of this type of reactor.

Even though the operation of other TRIGA reactors has demonstrated that the TRIGA is inherently safe, prototype experiments have been conducted in General Atomic's TRIGA Mark F facility. The results of these

experiments have been used as a basis for establishing the final operating parameters of the DORF-TRIGA. The prototype experiments in the TRIGA Mark F will continue at General Atomic for an extended period of time.

In support of the prototype experiments and to demonstrate the safety of this reactor system, a study of abnormal conditions which might occur has been included in this hazards summary. Furthermore, the particular design features of the entire facility which contribute to the reactor's safety are described, and the production of radioactive gases, variations in excess reactivity, improper fuel loading, malfunction of experiments, and loss-of-coolant accident have been investigated.

Rigorous administrative controls for reactor operation will minimize the possibility of even trivial releases of radioactivity.

To minimize air leakage, a slight negative pressure will be maintained within the reactor building, and access openings will be rubber-gasketed. Air from the building, including that in the exposure room, will be monitored and will be discharged through a system of filters and a stack that will extend about 45 ft above ground level.

Based on these analyses, which are supported by experimental data from tests on the Mark F prototype and operating history from other TRIGA reactors, and the necessary administrative control, it is concluded that the operation of the proposed DORF-TRIGA at the Forest Glen Annex of the Walter Reed Army Medical Center does not present an undue hazard to the health and safety of the DORF operating personnel or the general public.

REACTOR PARAMETERS

Fuel elements

Number	~85
Fuel-moderator material	Uranium–zirconium hydride
Uranium enrichment	20% U^{235}
Fuel-element dimensions (over-all)	1.48 in. diam. x 28.4 in. long
Cladding	0.030-in.-thick aluminum
Dimensions of active lattice	~17 in. diam. x 14 in. high
Poison	Burnable poisons

Reflector

Material:	
Radial	Water or water and lead
Axial	Water and graphite
Thickness:	
Radial	Variable
Axial	4 in. of graphite and water

Nuclear characteristics

δ - Barn (10^{-24} cm)

Fuel inventory	3.2 kg
Average thermal-neutron flux in core at 100 kw	1.1×10^{12} neutrons/cm^2-sec
Initial excess reactivity allowance	2.9% $\delta k/k$
Transient reactivity insertion	2.2% $\delta k/k$ (maximum)
Reactivity value in control system	7.2% $\delta k/k$ (minimum)
Prompt temperature coefficient of reactivity at ~300°C	$-(1.1 \pm 0.1) \times 10^{-4}$ $\delta k/k/°C$
Energy shutdown coefficient	$(2.6 \pm 0.2) \times 10^{-5}$ $watt^{-1}$ sec^{-2}
Void coefficient of reactivity in core at 20°C	$- 2 \times 10^{-3}$ (% $\delta k/k$)/(% void)
Prompt neutron lifetime	45 μsec
Average delayed neutron fraction	0.0073

Thermal characteristics

Power level	100 kw continuous, or 250 kw 4 hr/day
Maximum fuel temperature	530°C during transient operation
Shutdown temperature level	530°C
Cooling	Natural convection

Control

1 Boron carbide or cadmium transient rod

3 Boron carbide control rods

Drives:

Control rods	Rack and pinion
Transient rod	Air-operated piston and motor-driven ball-nut lead screw with a variable speed motor
Maximum reactivity addition rate for approach to criticality	0.04% $\delta k/k$/sec

Instrumentation

Log n and period channel
Power-level channel (scram circuit)
Servoamplifier for regulating rod control
Rod-position indicator
Water-radiation monitor
Fuel-element temperature (indicator, recorder, and scram circuit)

Neutron sources

Polonium-beryllium
Antimony-beryllium

Shielding

Radial	Ordinary concrete ~ 4 ft thick
Vertical	16 ft of water above core

CONTENTS

Appendix

Chapter 1

SITE

LOCATION

The Diamond Ordnance Radiation Facility Research Reactor will be located within the metropolitan area of Washington, D.C., at the Forest Glen Annex of the Walter Reed Medical Center, which is five miles from the center of Washington, D.C., and approximately two miles south of Kensington, Maryland (see Fig. 1). The Forest Glen site is an area of approximately five acres of rolling, partially wooded and cleared areas, on which are located both research-laboratory facilities and hospital facilities for patients (see Fig. 2). The site is located in a commercial and residential area.

The location of the reactor will be near the southern border of the Forest Glen area (see Fig. 2) about 800 ft from the nearest research laboratories and about 500 ft north of Brookville Road, which bounds the property on the southeast. The site is surrounded by woods, except to the north, where a large open field separates it from the research laboratories. The reactor building will be encircled by an exclusion fence with a radius of approximately 200 ft (see Fig. 3). Access to the exclusion area will be controlled at the single entrance gate.

HYDROLOGY

Surface drainage at the reactor site is to the south and west into Rock Creek, which drains south through Rock Creek Park and into the Potomac River. The discharge of Rock Creek at Sherrill Drive in the District of Columbia, 7-1/2 miles upstream from its mouth, is as follows:

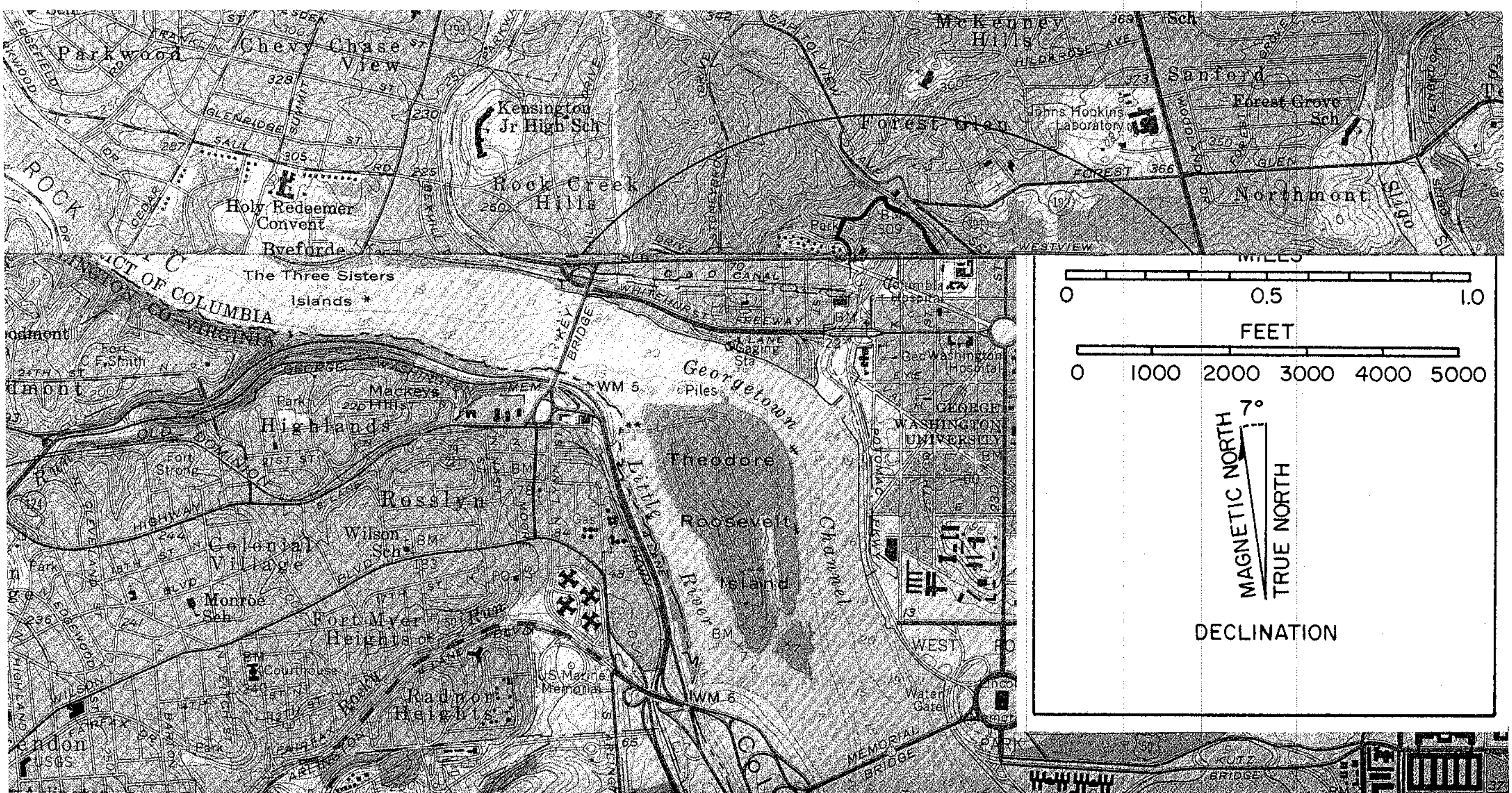


Fig. 1--Map of area surrounding the DOFL reactor facility

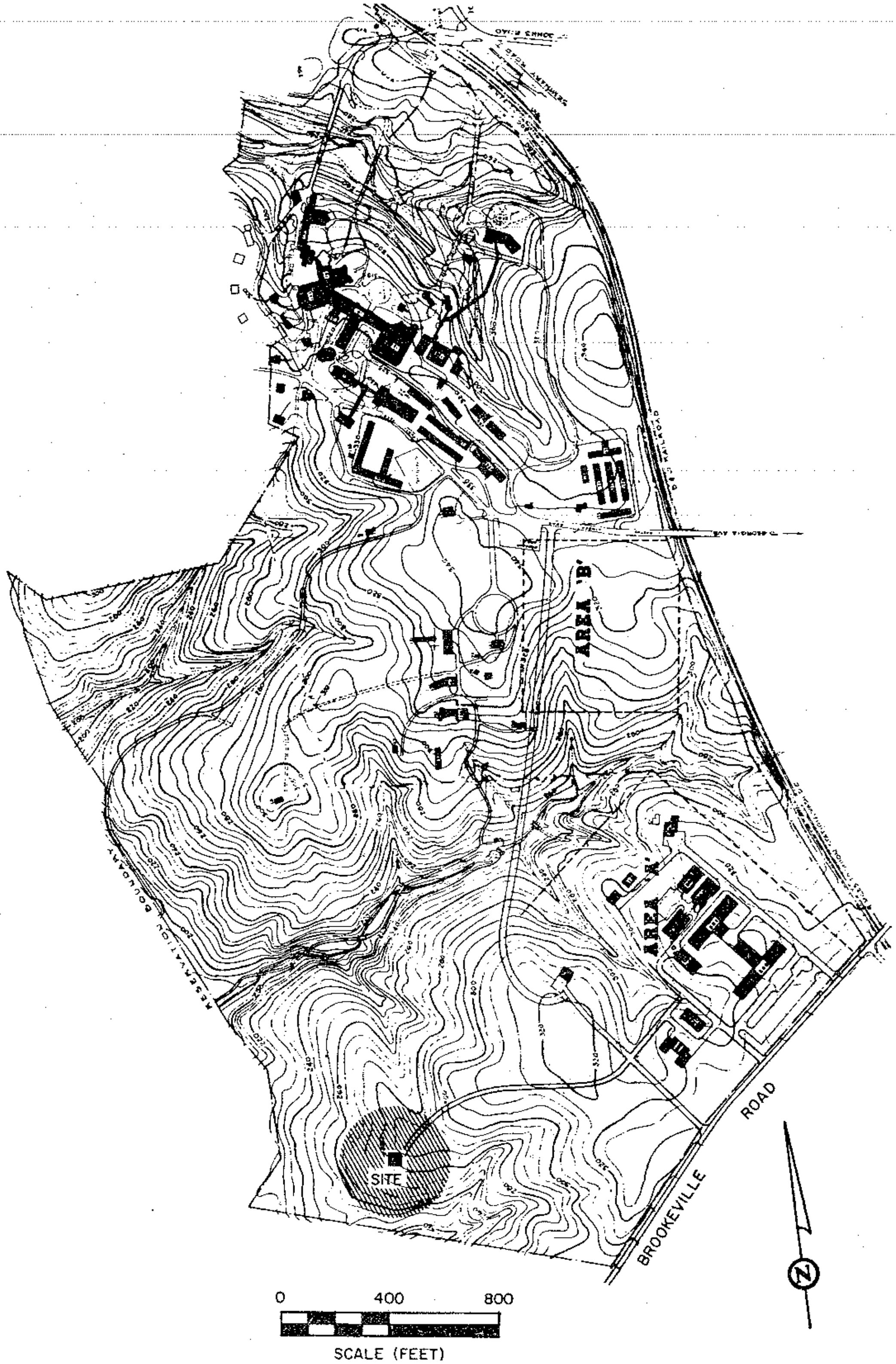


Fig. 2--Map of Forest Glen site for the DORF

28-yr average . . . . 55.8 ft^3/sec

1929-1957 maximum . . . . 7220 ft^3/sec, July 21, 1956

Calendar year 1956 average . . . . 48.4 ft^3/sec

Water year October, 1956, through September, 1957:

Maximum . . . . 1210 ft^3/sec, April 5, 1957

Minimum . . . . 2 ft^3/sec, August 17, 18, 19, 23, and 25, 1960

The ground water at the proposed reactor site is locally recharged by precipitation and infiltration. The average yearly precipitation in the Montgomery County area is about 41 in., evenly distributed throughout the year. July is the wettest month (about 4.5 in. of rain), and October is the driest (about 2.8 in.). Most of the precipitation is lost to the atmosphere by evapotranspiration, and only a small fraction enters the zone of saturation. The zone of saturation, the zone below the water table, was not encountered in test borings at the reactor site, the deepest of which was 45 ft.

The sand, silt, and clay of the weathered rock, saprolite, has a low permeability, probably less than 1. Since the water table was not reached by the test borings, which were made during the peak of the average annual 10-ft variation in the water table found in the Washington vicinity, it is reasonable to assume that the depth of the water table will seldom be less than 45 ft.

It was not possible to determine the rate of ground-water movement. In general, however, the velocity of ground water through fine-grained sedimentary materials, such as fine sand, silt, and clay, is very low (on the order of a few feet per year).

WATER WELLS

No wells are known to exist in the immediate vicinity of the proposed reactor site. The nearest known supply well is 1-1/2 miles east of the site, and another is located about 2 miles north of the site. Both wells are considerably higher on the regional hydraulic gradient which rises to the northeast of the reactor site.

Rock Creek flows south past the reactor site, traverses the entire length of the District of Columbia through Rock Creek Park, a recreational area, and discharges into the Potomac River below the intakes for municipal water supplies.

EARTHQUAKES

The Maryland Piedmont, on which the site is located, is a region of comparative crustal stability. During the past 156 yr no severe quakes have occurred, though there have been a few quakes of low intensity. The probability of a destructive earthquake in the Montgomery County area is very slight. The probability of hazards arising from extra-site sources, such as explosions or floods, is greater than from seismic activity.

POPULATION

The DORF site will normally be occupied by 15 male personnel on a normal 8-hr day basis. The Forest Glen Annex of Walter Reed Army Medical Center has a total estimated population of 1041, located as follows:

Research and Development Area (Nearest to reactor site)	
Civilian employees	99
Military	39
Total	138
Convalescent Area	
Civilian employees	229
Military	374
Patients	250
Resident civilians	50
Total	903
Total estimated population	1041

The area outside the medical center towards metropolitan Washington is densely populated. Metropolitan Washington has an estimated 1.4 million people.

GEOLOGY

Topography

The Army Medical Center in Forest Glen is on the well-dissected hills bordering the east side of Rock Creek. The reactor site is on a small bench at an altitude of about 280 ft above mean sea level. Surface drainage flows from the site directly into Rock Creek and to the north and south into small unnamed tributaries of Rock Creek, and from these to the Potomac River near Georgetown and Theodore Roosevelt Island (see Fig. 1).

Subsurface Geology

The site is on a narrow belt of the Kensington-granite gneiss, which is a highly foliated, coarse granite intrusive in the Wissahickon schist complex and basic rocks. Petrographic mineral determinations are contained in Reports TDS-A526 and IDM-A526 of the United States Department of the Interior, Geological Survey, Geochemistry and Petrology Branch.

The relic structures of the bedrock just described extend into the weathered rock mantel, saprolite, which ranges in thickness at the site up to about 45 ft. Locally, boulders or veins of quartzite, essentially unaltered by weathering, remain in the saprolite. The saprolite is composed of micaceous silt, clay, and fine-to-medium grained quartz sand. The attitude of the planes of foliation orient the average permeability, so that ground water and other fluids that might be released to move through the ground travel more readily along the planes of foliation than across them. At the proposed reactor site, ground-water movement in the bedrock and in the saprolite, where it has retained relic structures of the bedrock, will be more rapid north and south, parallel to the foliation, than east and west across the planes of foliation. Fluids which may seep into the ground at the reactor site will most likely discharge at the surface in the gullies or hillsides to the north and south of the reactor site, in line with the direction of foliation, rather than directly down the surface slope toward Rock Creek. Thus, the west-northwest dip of the rocks at the site will retard any fluids released to the ground at the reactor site, provided that they do seep into the ground and do not run off on the surface directly into Rock Creek or into the small tributaries to the north and south. The ion-exchange capacity of the weathered metamorphic rocks in the vicinity of Washington, D.C., is usually less than 25 milliequivalents per 100 g (< 25 meq/100 g), which is low in comparison to purer clay formations, such as those of marine origin. Analyses of 12 rock samples from this site show exchange capacities of less than 10 meq/100 g. These analyses were made by the Geochemistry and Petrology Laboratory of the U.S. Geological Survey.

Test Drilling

Five bore holes and one auger hole were made to obtain samples for construction-engineering information and for the evaluation of environmental hazards. The deepest hole was bored to a depth of 45 ft. It was located exactly at the point where the reactor will operate, adjacent to the exposure room, and samples from this hole are representative of the earth materials that may be irradiated by reactor flux.

A second hole was bored outside the north wall of the exposure room, and a third was bored outside the south wall of the building, adjacent to the reactor pool. At the southwest corner of the building, a hole was bored to check the variation in the soil that might exist in the vicinity of the building. A bore hole and an auger boring were made at the proposed location of the waste-storage tanks and the line of waste discharge from the reactor building.

Chapter 2

FACILITY

GENERAL

The purpose of the facility is to study the effects of large, mixed doses of neutron and gamma radiation on electronic systems and related devices. To carry out these experiments, both a main exposure room and pool irradiation space are provided.

BUILDING

The building is designed to accommodate the reactor, reactor components, reactor shield structure, technical areas, personnel, and the equipment and instrumentation required to perform radiation-exposure testing of electrical and electronic components or systems and also metallurgical testing.

The floor plans and elevations of the three levels in the facility are shown in Figs. 4 and 5, respectively. The building, which is designed for a 25-yr life, is constructed of reinforced concrete, structural steel, and masonry. The building will accommodate an average of 15 male personnel on the basis of an 8-hr/day occupancy. Space within the building is divided into the following areas:

Exposure room	400 ft^2
Warm storage room	300 ft^2
Truck-access and sample-preparation area	300 ft^2
Mechanical-equipment room .	300 ft^2
Toilet and shower room	100 ft^2
Nuclear counting room	100 ft^2
Office	120 ft^2

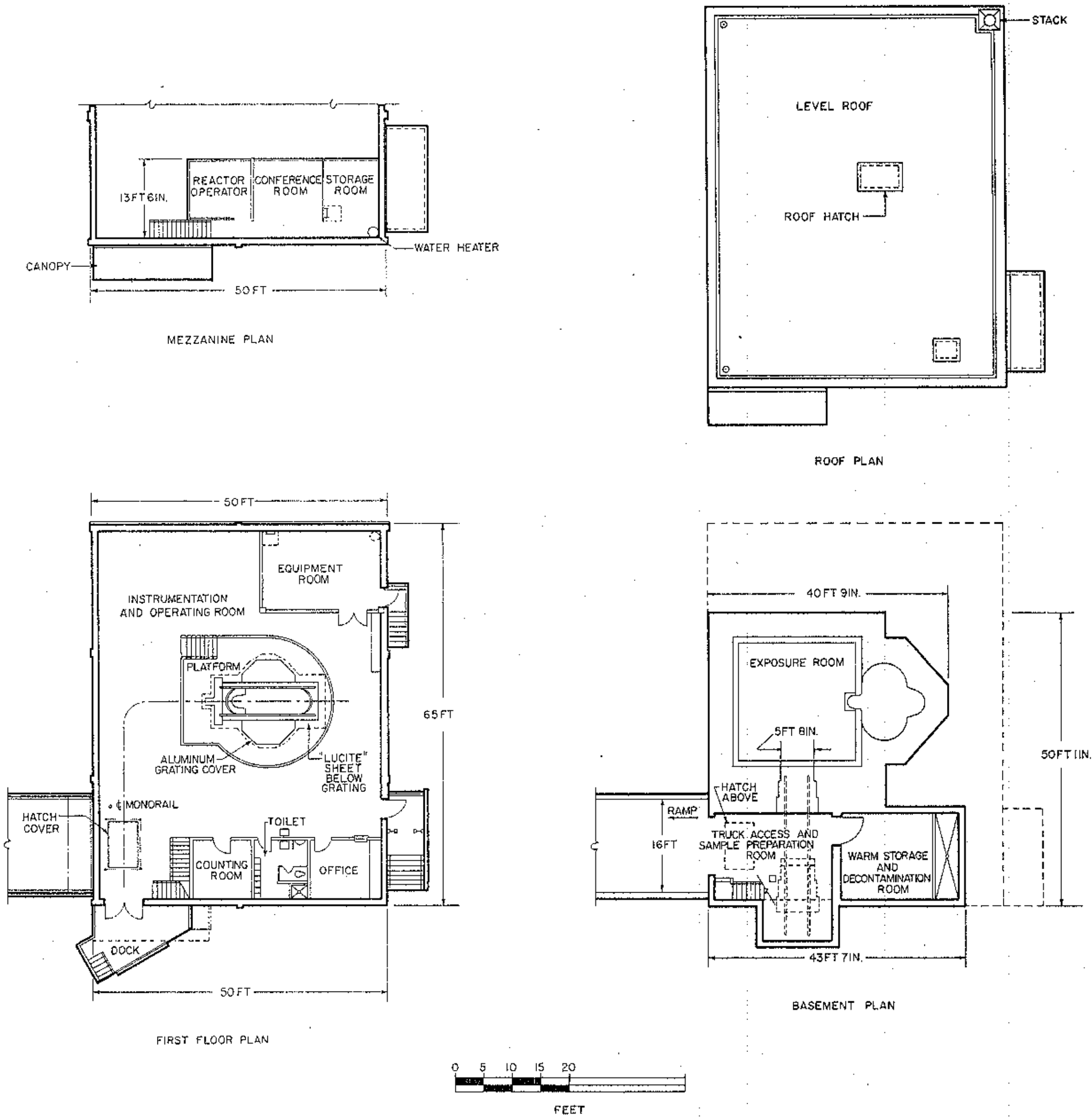


Fig. 4--Plan views of the DORF building

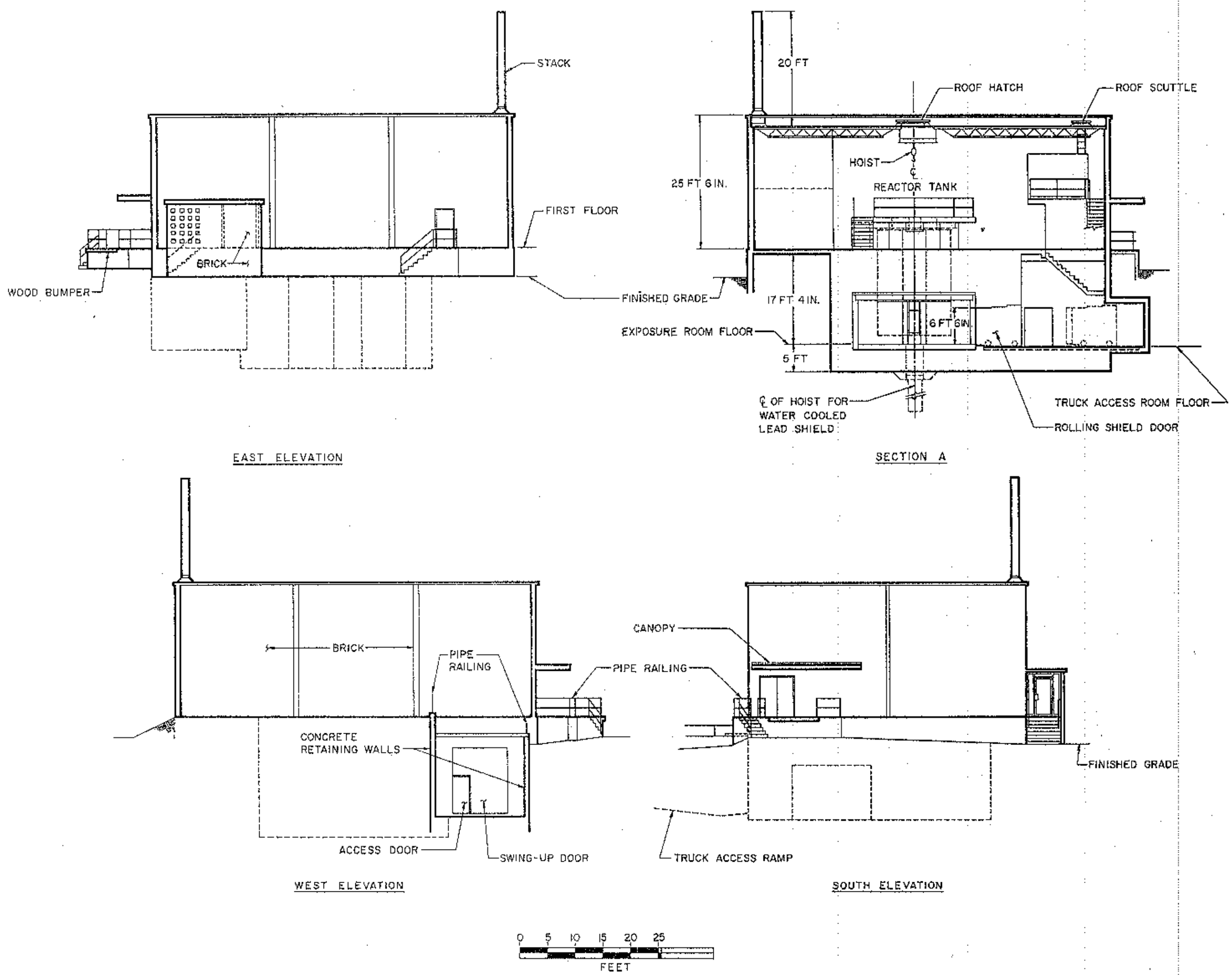


Fig. 5--Elevation and section views of the DORF building

Reactor operator station . . . 110 ft^2
Conference room . . . . 170 ft^2
Storage room . . . . 120 ft^2

The foundation and the exterior walls below the operating floor are constructed of reinforced concrete, with a waterproof membrane to prevent ground-water penetration into the building. The exterior walls above the level of the operating floor are constructed of 4-in. brick facing and an 8-in. cement-block backup. The cement blocks are sealed with an epoxy resin on the inner surface to reduce porosity.

The roof and ceiling are of perlite cast over metal decking with five layers of felt roofing on top.

AIR-CONFINEMENT CAPABILITIES

The building for the DORF-TRIGA reactor has been designed to operate at normal atmospheric pressure. The design also takes into consideration the intended uses of each area and provides good air confinement under adverse atmospheric conditions. Even under the unlikely possibility that a fuel-element cladding failure might occur concurrent with a sudden large drop in barometric pressure, the airborne radioactive fission products outside the building would not exceed the maximum permissible concentration. An analysis shows that the maximum dose which a person might receive in the area of maximum concentration outside the building would be 2 orders of magnitude below the maximum permissible dose for nonoccupational personnel established by 10 CFR 20.

The doorways and roof hatches in the building represent the only escape path for air except for the slight possibility that air might diffuse through several inches of concrete. All exterior doors and roof hatches will be well fitted and gasketed with rubber. As a result, the building will have no communication with the outside air when the doors and ventilation dampers are closed. A pressure-release system, described below, will control the differential pressure.

Building Ventilation Under Normal Conditions

The air-conditioning system exhausts all air from the reactor building through absolute filters and out the stack. This precaution is taken, even though no radioactive particulate matter will be in the air during normal operation.

The only possibility which might create airborne radioactive particulate matter would be a fuel-element cladding failure. Should such a failure occur, it is possible that small amounts of radioactive noble gases would be dispersed from the reactor pool into the reactor-room air, and these would decay into particulate matter.

Controlled Confinement of Air

Although the release of fission-product gases from a fuel-element cladding failure should in no way endanger the operating personnel or the public, the design of the building makes it possible to isolate and confine the air within the building. The system has been designed so that when the continuous air monitor indicates abnormal airborne contamination, an alarm automatically sounds and the positive-sealing dampers in the ventilation system automatically close, isolating the air in the building at ambient atmospheric conditions.

Two conditions which could cause the confined air to leak out of the building would be a sudden drop in barometric pressure or an excessive heating of the air within the building, causing it to expand. In order to control the release of air from thebuilding under either of these rather abnormal meteorological phenomena, a special atmospheric relief duct is provided in the system.

The resistance to air flow provided by this relief duct is very much less than the resistance to the flow of air through the solid concrete walls or through the compressed rubber gaskets which seal the doors and hatches. Therefore, it can be reasonably assumed that the small volume of air that will be expelled from the facility under either of these conditions will be directed through the atmospheric relief duct and out of the building through the air stack. Since the circulation of outside air will continue through the emergency intake of the exhaust fan, the small volume of air expelled from the building will be considerably diluted as it is forced out the stack. By the time it reaches ground level, the concentration of Xe^{131} will be about 1/300 of that allowed by the Atomic Energy Commission.

Fission-gas Concentration From Stack Release

On the assumption that both events mentioned above have occurred and that a small fraction of the contaminated air from the reactor room has been released through the stack, a calculation of the resulting concentration of fission gas has been made. In order to determine the amount of fission-product gases which might be released at any one time, it was necessary to determine the effects of the two phenomena mentioned above.

Effect of Sudden Barometric Pressure Change on Room Air Leakage. A pressure differential between the inside and outside of the building can occur because of a sudden drop in barometric pressure. The barometric pressure changes in the Washington, D.C., area, as given in the records of the U.S. Weather Bureau dating from 1893, are as follows:

Maximum monthly variation, 1.87 in. Hg;

Maximum 24-hr variation (fall), 1.47 in. Hg;

Maximum 24-hr variation (rise), 1.22 in. Hg;

Maximum 12-hr variation (fall), 1 in. Hg.

When the barometric pressure drops, air will be expelled from the building. On the basis of the 1.47-in. drop in barometric pressure recorded above and the volume of reactor-room air, 71,350 ft^3, it has been calculated that 3,690 ft^3 of air will be expelled from the reactor room in 24 hr. This leakage amounts to 5.17% of the volume of air in the reactor room in 24 hr.

If it is assumed that the fission-product gases in the reactor room air resulting from a fuel-element cladding failure are xenon and krypton, with an average total activity in the room of 2.45 c, as calculated and discussed in Appendix IV, the average concentration of xenon and krypton in the reactor-room air will be 1.21×10^{-3} $\mu c/cm^3$. Since 5.17% of the activity will leak out with the expelled air, 0.127 c will escape in 24 hr, or 1.4 $\mu c/sec$.

On the conservative assumption that the average activity will not decrease from the natural decay of fission products during the 24-hr period, it has been calculated that the maximum concentration of the fission-product gases near the ground will occur at a distance of 1,940 ft from the reactor building. The maximum concentration in air at that point would be 8.0×10^{-10} $\mu c/cm^3$, which is 1/300 of the Xe^{133} maximum permissible concentration (10 CFR 20) for continuous exposure in unrestricted non-occupational areas.

Effect of Temperature Differential on Room Air Leakage. Another condition that could cause air to be exhausted from the reactor room, when the air in that room is isolated from the rest of the building because of a fission release from the reactor tank, would be high temperature outside the reactor building. Radiant heating from outside would raise the temperature of the isolated and confined reactor room air. Under these conditions, it has been calculated that an outside air temperature of 100°F might increase the temperature inside the reactor room 15°F before equilibrium was reached. These conditions are based on the assumption that the air-conditioning system is shut down early in the morning on a day on which the outside-air temperature reaches a maximum of 100°F. During the day, the air inside the building will increase in temperature from an assumed initial temperature of 70°F to 85°F by evening, when the temperature of the outside air would have decreased, creating an equilibrium. This condition would result in a leakage of 3.4% of the total reactor-room air volume in 24 hr.

It does not seem reasonable to assume that there could possibly be a coincidence of these two natural phenomena--namely, a sudden change in barometric pressure concurrent with a wide variation in inside and outside air temperatures.

Since the volume of air which would leak from the reactor building under the conditions created by a sudden change in barometric pressure is greater than the volume of air expelled by heating, the concentration in the air will be proportionately less in the latter case. Therefore, it can be assumed that there are no credible conditions which could endanger the health or safety of the public through the release of fission products from a fuel-element failure.

A calculation of the effect of a release of fission products in the building is presented in Appendix IV. The exposure (16.5 mr) which a person inadvertently remaining in the reactor room could receive following a fuel-element cladding failure is well below the maximum permissible daily exposure established by 10 CFR 20 for operating personnel.

VENTILATION SYSTEM

The ventilation system (Fig. 6) and air conditioning throughout the building will provide:

1. Dissipation of building heat generated by electrical equipment and lights;
2. Normal ventilation and cooling;

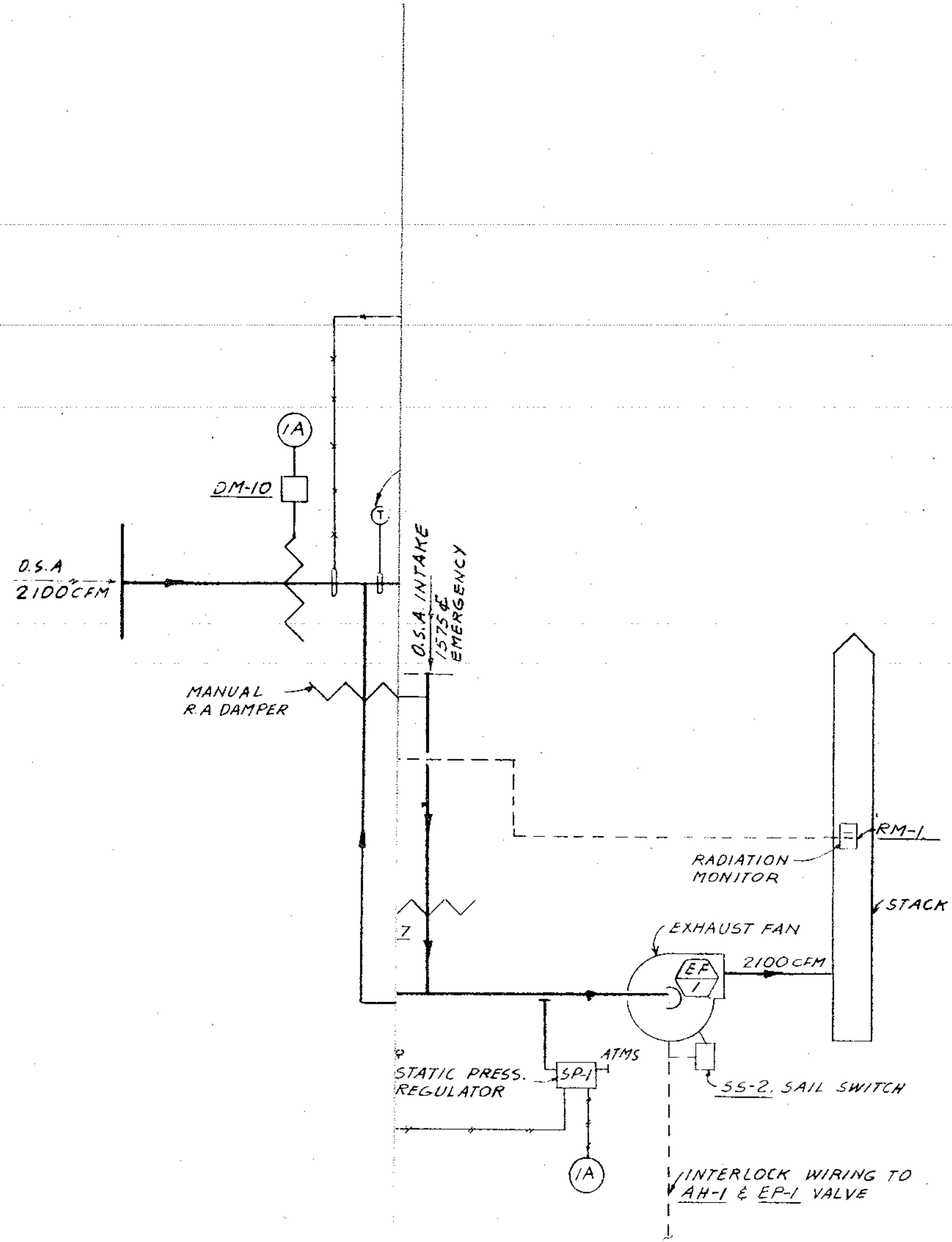


Fig. 6--Flow diagram of ventilation system for reactor building

3. Positive filtered exhaust of all possibly contaminated air.

The entire building will be air-conditioned by a package type of system. Normal air recirculation will be used in the building, except in the exposure room and in the warm storage room. All exhaust air from the exposure room and the warm storage room will pass through an absolute filter of minimum 99.77% efficiency with a dioctyl phthalate (DOP) penetration of 0.05% for 0.3-μ-diameter particles, and then out through an exhaust stack that extends approximately 45 ft above the ground level. The flow of air will be controlled by maintaining a lower pressure within the exposure room and the warm storage room than in the remainder of the building. The remainder of the building will, in turn, have a slightly lower pressure than that found outside. Accidentally contaminated air is thus channeled through the filters.

A stack monitoring system activates alarms if undesirable quantities of radioactivity are being exhausted to the atmosphere. A smoke sensor is also provided, which activates alarms if smoke is detected in the exhaust system.

The building is heated by a gas-fired, 225,000-Btu/hr steam boiler and thermostatically controlled steam coils in the air-handling system.

WATER PURIFICATION AND COOLING SYSTEM

The coolant water for the reactor is purified and cooled in an external system (see Fig. 7), which consists of a mixed-bed demineralizer, heat exchanger, pump, and associated piping and valves. The system also includes a surface skimmer, a fission-product monitor, a fiber cartridge-type filter with pressure gauges, and a flow meter. The cooling capacity of the water system is 100 kw at water temperature of 90°F. The water conductivity is kept at about 2 μmho to minimize corrosion. The purification system removes radioactive ions or particles from the

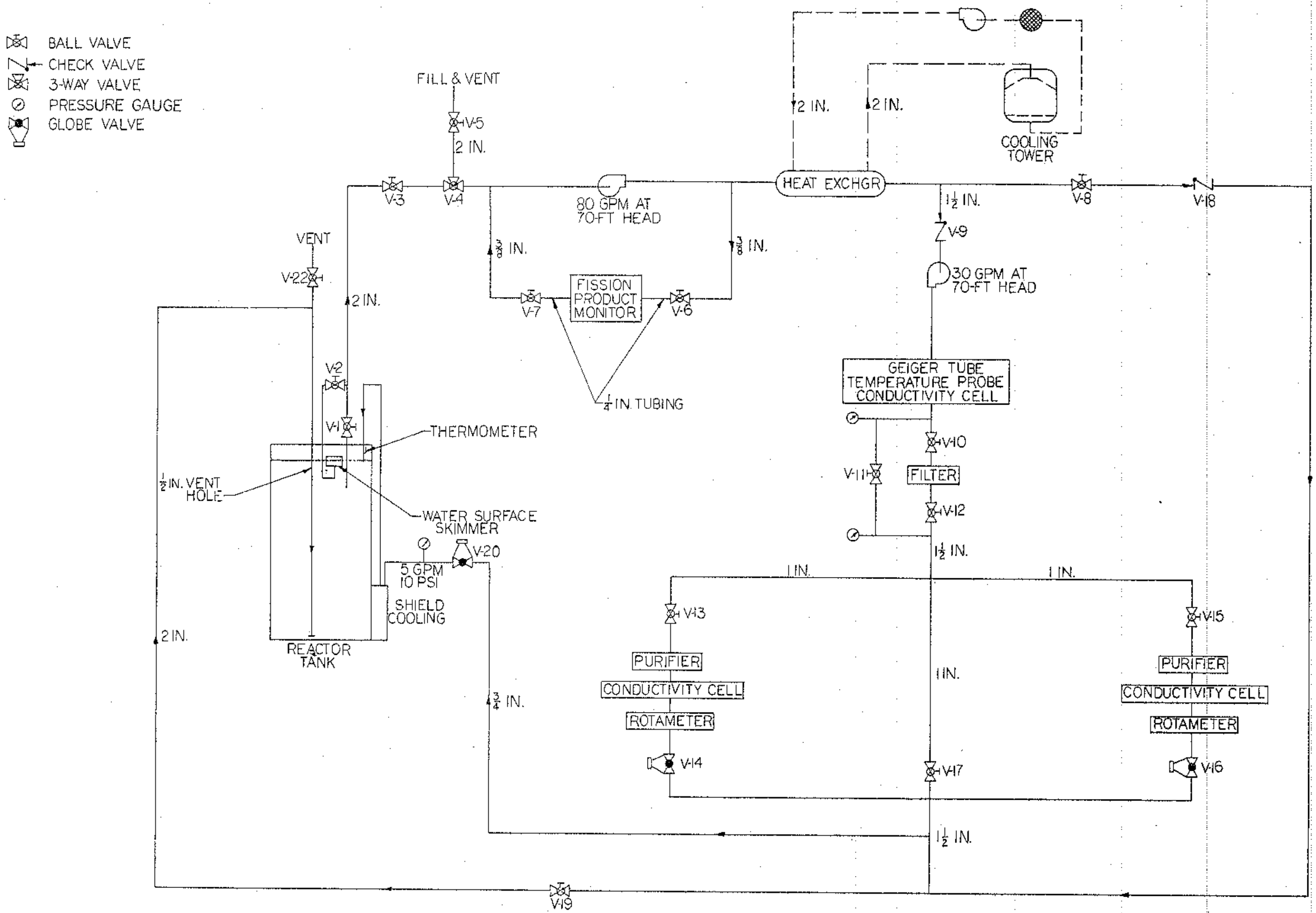


Fig. 7--Coolant-water purification system for the DORF

reactor water and helps to maintain optical clarity of the water.

The heat exchanger is a conventional shell-and-tube type. The shell and cover are made of carbon steel, and the tubes and all other parts in contact with the reactor water are made of stainless steel. The drop in water temperature across the heat exchanger is approximately $10^{\circ}F$, at a flow rate of 80 gpm. The water is circulated through the primary system by an aluminum centrifugal pump, which provides a head of approximately 70 ft. The secondary coolant enters at $70^{\circ}F$ and exits at approximately $80^{\circ}F$ during 100-kw operation. The design temperature for the shell side of the heat exchanger is $100^{\circ}F$; for the tube side it is $120^{\circ}F$. The design pressure for both sides of the exchanger is 75 psig.

The primary function of the demineralizer system is to maintain a water-conductivity level low enough to minimize fuel-element corrosion. The demineralizers are of a mixed-bed type that removes both positive and negative ions from the circulating water. The system includes two demineralizers, each of which has a flow capacity of 10 gpm and contains about 3 ft^3 of resin. The type of resin provided is a mixture of nuclear-grade Permutit A-H and Permutit S-1. The flow through each demineralizer is regulated by a flowmeter that is located on the downstream side.

A filter removes insoluble particulate matter from the reactor water system. It has a replaceable fiber cartridge, which is removed from the vessel and replaced when its pressure drop becomes excessive. Two filter cartridges 1 each rated at 10μ, remove all of the particles down to 1μ in size.

Two pressure gauges in the filter bypass line measure the pressure drop across the filter and indicate when the filter should be changed.

WARM-WASTE DISPOSAL SYSTEM

A warm-waste system which collects water from the floor drains in the warm-storage room, decontamination area, exposure room, lavatories and both showers, has a detention capacity of 15,000 gal. The system utilizes three 5,000-gal detention tanks, connected in parallel. Warm waste can be directed to any of the three detention tanks, and can later be discharged from any of the three tanks as the radioactivity drops within the acceptable range.

The warm-storage tanks will be sampled and analyzed for radioactivity, in order to ensure that the warm waste may be flushed into the sanitary sewer. Figure 8 shows the flow diagram of the waste-disposal system

DECONTAMINATION PROVISIONS

A special paint capable of withstanding a decontamination procedure is applied to the concrete walls, floor, and ceiling of the warm storage room. This paint is also applied to the concrete around the truck access area. The main operating floor and the mezzanine are covered with vinyl tile, and the interior walls are sealed with epoxy.

The exposure room is lined with 1 ft of structural-grade wood to reduce activation of the concrete. The floor in the exposure room is covered with a waterproof epoxy surface. The walls and ceiling are covered with a special decontamination paint.

Drains in the building are arranged so that all water or other fluids used for decontamination are channelled into the warm-waste disposal system. Should personnel be contaminated, an emergency shower is located in the toilet, and in the warm-storage room in the basement are located an emergency shower, eyewash, and sink.

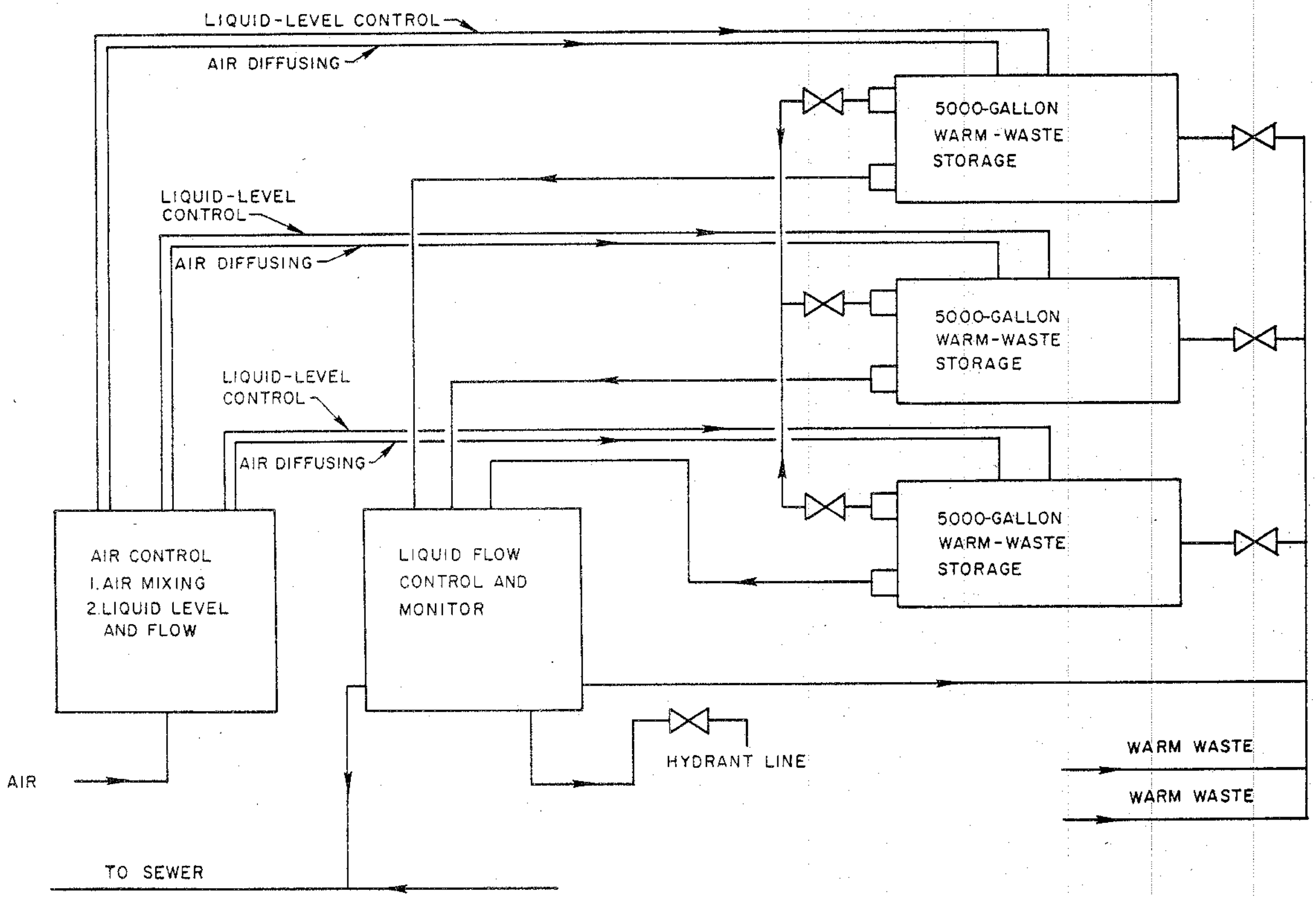


Fig. 8--Warm waste-disposal system for the DORF

UTILITIES

Potable water is piped to the building site through a 2-in. water line, which is approximately 450 ft long. It is estimated that the maximum normal water usage will be 1,200 gpd. An 8-in. sanitary-sewer service runs within 80 ft of the building. However, in order to meet sewer-line inverts and to accommodate the warm-waste detention system, the sanitary sewer from the building intersects the 8-in. sewer line at a manhole approximately 300 ft northwest of the reactor building.

Electric power will be purchased from the Potomac Electric Power Company at 120/208 v. Power is carried from Brookville Road on 13.2-kv overhead service lines to a power-company-owned substation located beside the reactor building. A total connected load of 195 kw indicates that an average estimated demand load of 110 kw will be required at the building.

Natural gas will be piped from the gas main approximately 450 ft through a 20-in. line. It is estimated that during the heating period 225 ft^3/hr will be required.

A 6-ft high chain-link fence with three rows of barbed wire will completely enclose the reactor building and will provide a 200-ft clear space on all sides of the building.

Access to the reactor building is through a 16-ft wide roadway that intersects the driveway from Brookville Road. A parking lot will accommodate nine cars and three 40-ft long trailers with tractors. Truck access to the building is at the basement level and at the operating-floor level.

SOLID-WASTE DISPOSAL

Disposal of solid waste will be handled through arrangement with Walter Reed Army Medical Center.

Chapter 3

REACTOR

DESIGN CRITERIA

The reactor design (see Figs. 9, 10, and 11) presented herein employs the TRIGA reactor fuel elements to provide a facility where studies may be undertaken of the effects of large pulses of neutrons and gamma rays on electronic and other devices of interest to the Diamond Ordnance Fuze Laboratory. The reactor core will have a minimum reflector so that large numbers of neutrons may be emitted by the core into a shielded exposure room. The reactor will have a 2-in.-thick water reflector, covering 180° of the periphery, through which will extend a small, 3 in. by 10 in., air-filled aluminum thimble when the reactor is adjacent to the exposure room. The thimble will be used for the irradiation of very small objects. The core will have the capability of being pulsed by the insertion of up to 2.2% $\delta k/k$ excess reactivity with no hazard to the reactor or the operating crew. Detailed operating characteristics are given in Chapter 5.

Easy access to the core and adequate cooling and shielding will be provided by the large pool of water in which the reactor-core assembly is submerged. Cooling will be provided so that the reactor may be operated at 100-kw steady state and at 250 kw, not to exceed 1 Mw-hr per day, to allow for extended irradiations. To make maximum use of the reactor, the facility will be arranged so that exposure of samples may occur either in the dry exposure room or within the reactor pool. To facilitate this utilization, the reactor will be transported by a carriage from one exposure position to the other within the pool in approximately 6 min. Shielding will be provided so that access to the exposure room may be made while experiments are being conducted in the pool.

Operation and maintenance requirements will be minimized by

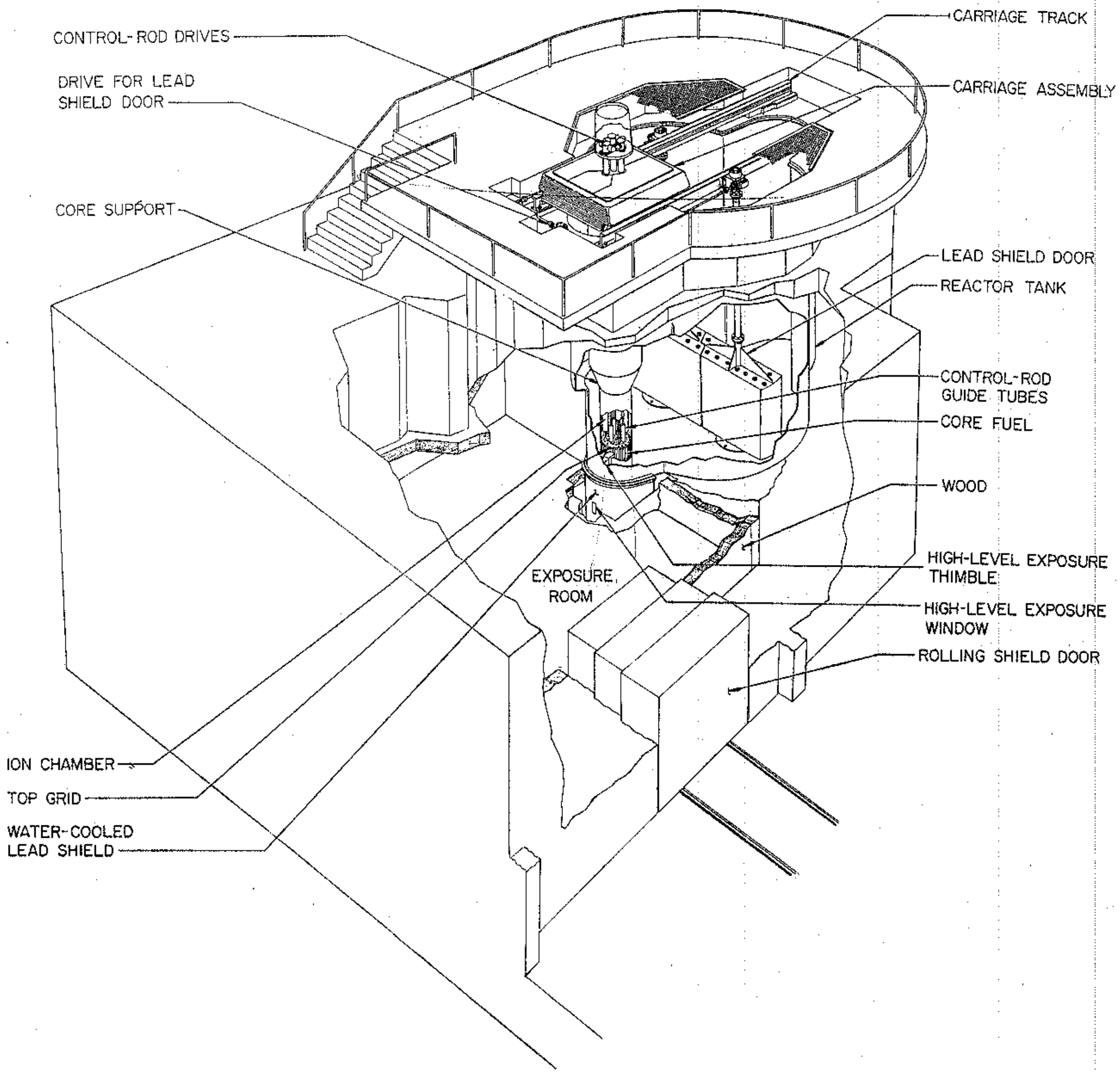


Fig. 9--Perspective view of the reactor

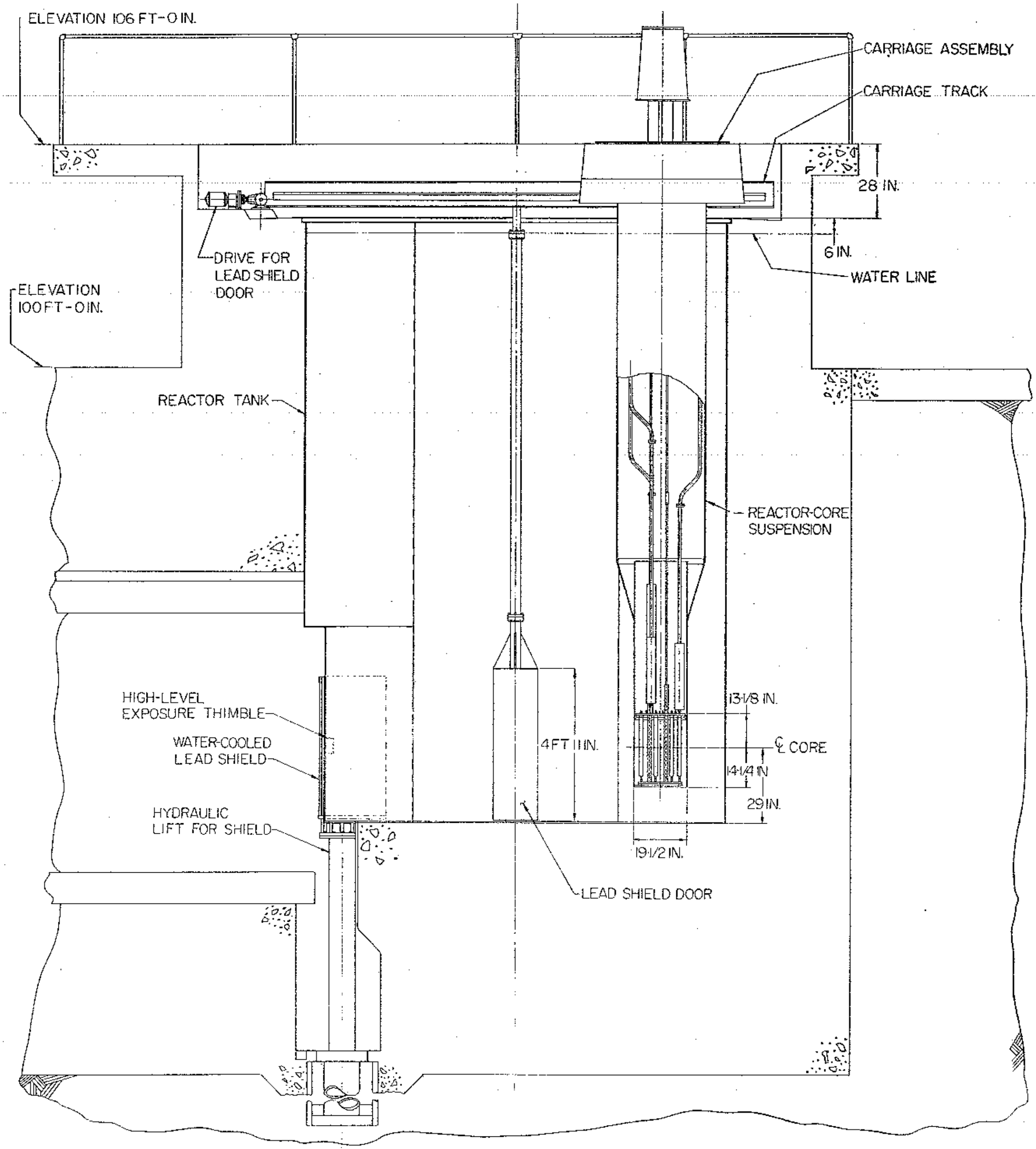


Fig. 10--Sectional elevation of the reactor

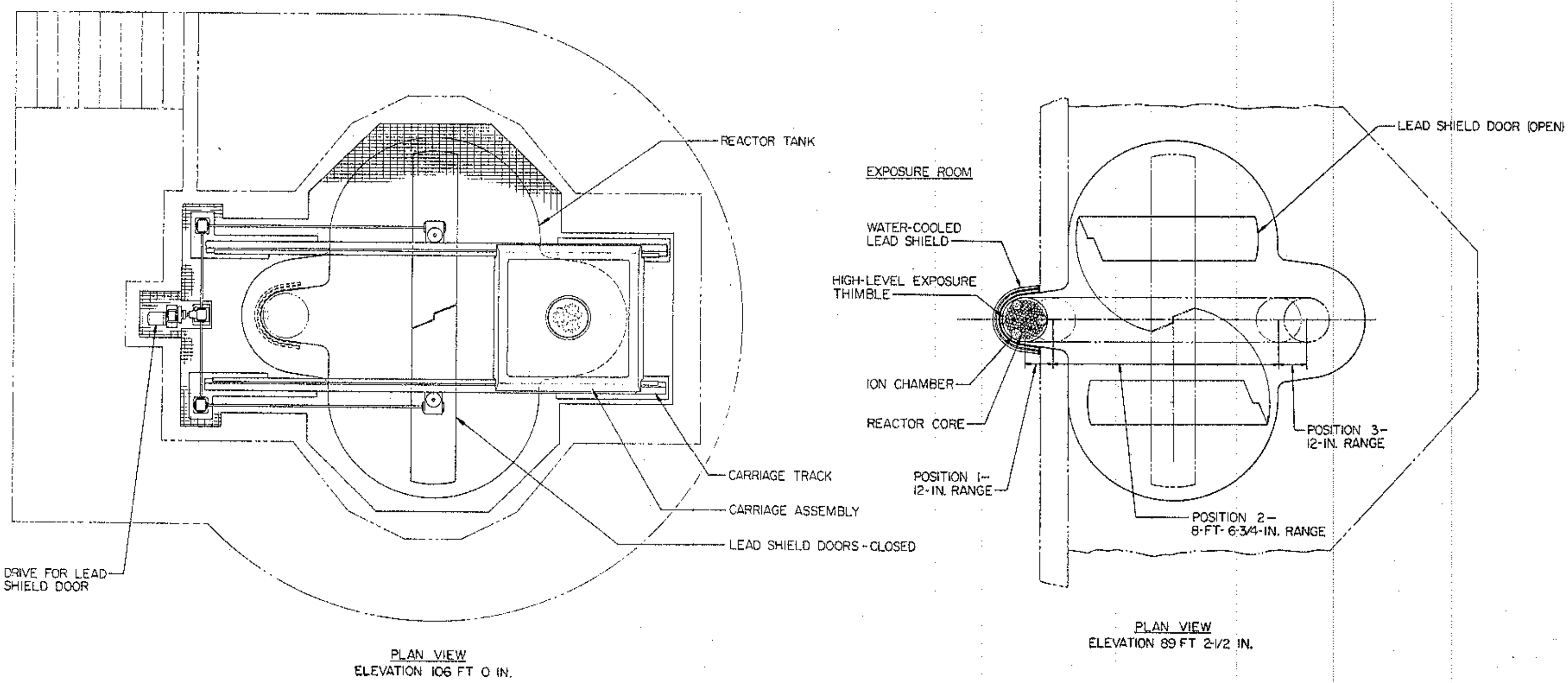


Fig. 11--Plan view of the reactor

employing a simplified control system. To assure safe operation, the reactor will be provided with a reflector of water sufficient to ensure that no change in reactivity larger than 0.5% $\delta k/k$ occurs as a result of moving the reactor from one position to another or of the addition of neutron and gamma-ray filters adjacent to the core. Adequate shielding will be provided for the reactor core and the exposure room so that operating personnel in areas external to the shielding will not receive doses of radiation exceeding one-tenth of that specified by the U.S. Atomic Energy Commission during 100-kw steady-state operation. The neutron-to-gamma-ray ratio will be variable by a factor of 10 by the addition of lead around the core as a gamma-ray filter or by water as a neutron filter.

When the reactor is moved into the exposure-room position, the water reflector is diminished to a 2 in. thickness around half of the core. In addition to this, there is a 1/4-in.-thick aluminum thimble, which is fabricated as part of the reactor tank, that penetrates through the 2-in. water reflector and into the F-ring of the reactor core. In the region where the thimble penetrates, two fuel elements are removed to accommodate the penetration. The closest approach of the thimble to a fuel element is 0.7 cm.

A 2-in.-thick water-cooled lead shield is provided, which is mounted external to the tank on a hydraulically operated piston. This lead shield has an opening in it which allows access to the thimble. The opening can be filled with a removable lead plug. Operation of the movable lead shield is interlocked so that the shield cannot be moved while the reactor is in operation.

The exposure room will be accessible through an opening through the shield containing a movable concrete shielding plug. Electrical interconnections will be accommodated through the roof of the exposure room by means of spiral conduits. These conduits are surrounded by lead to ensure the integrity of the shield.

The communication systems for the reactor and associated exposure room will include both audio and visual systems. A closed-circuit television system will be employed to monitor personnel and experiments in the exposure room and warm-storage area. An audio system will be employed to facilitate maintenance and coordination within the facility.

GENERAL ARRANGEMENT OF REACTOR

The reactor core forms a compact cylinder and consists of a lattice of approximately 85 cylindrical fuel-moderator elements, 4 control rods, and 1 neutron-source holder contained between the top and bottom aluminum grid plates and surrounded by an aluminum shroud which supports the grid plates. This assembly is located at the bottom of an aluminum reactor tank that is approximately 14 ft in diameter and 19 ft 6 in. high and holds 12,000 gallons of water. The core assembly is suspended by a support structure from a motor-driven carriage which is at the top of the tank and is capable of traversing the tank.

Control-rod drive mechanisms are located on the carriage and are connected to the control rods in the core. The reactor is controlled by two safety rods, a regulating rod, and a shim-safety-transient rod. Instrumentation is provided to monitor, indicate, and record the neutron flux. Three modes of operation are possible: Mode I--steady-state operation, with manual or servo control to 100 kw; Mode II--power square wave to 250 kw, maximum; Mode III--flashing operation to 2200 Mw.

In addition to the reactor-control instrumentation, an interlock system is provided to prevent reactor operation unless prescribed safety conditions have been met.

The reactor core is cooled by natural convection and the pool water is purified and cooled in an external system which consists principally of a water-to-water heat exchanger, a mixed-bed demineralizer, a pump and associated piping, valves, and flow-indicating devices. The secondary water to the heat exchanger is circulated through a filter and cooling tower.

In addition to the shielding provided by the water in the reactor tank, two 18-in.-thick shielding doors are located in the tank to shield the exposure room. These are mounted on bearings on the floor of the tank and can be rotated to allow passage of the core from one operating position to another when prescribed safety conditions have been met.

FUEL-MODERATOR ELEMENTS

A fuel-moderator-element assembly is shown in Fig. 12. Including the top and bottom aluminum end-fixtures, the fuel-moderator element is 28.44 in. long. The fuel part of each element, which is 1.42 in. in diameter by 14 in. in length, consists of an alloy of uranium-zirconium hydride containing 8 wt-% uranium enriched to 20% in U^{235}. The hydrogen-to-zirconium atomic ratio is approximately 1.0. A thin aluminum wafer at each end of the active fuel section contains a burnable poison. By this means an appropriate amount of burnable poison is incorporated in each fuel element at the time of fabrication to minimize the loss-of-reactivity effect due to fission-product poisoning and fuel burnup. Four-inch sections of graphite in the fuel can above and below the fuel region serve as top and bottom reflectors for the core. The fuel elements are clad with 0.030-in.-thick aluminum, and all closures are made by heliarc welding. An aluminum end fixture is fixed to each end of the can for positioning and handling. Each standard fuel element contains approximately 37 g of U^{235}. Partially loaded fuel elements containing lesser amounts of U^{235} are used, as necessary, to make available the exact amount of excess reactivity required in the reactor.

GRID PLATES

The fuel elements are spaced at the top and bottom by two 0.75-in.-thick aluminum grid plates. The grid plates have a total of 91 holes, or fuel-element spaces, 85 of which are available for fuel-moderator elements and the remainder for the 4 control rods. The bottom grid plate, which

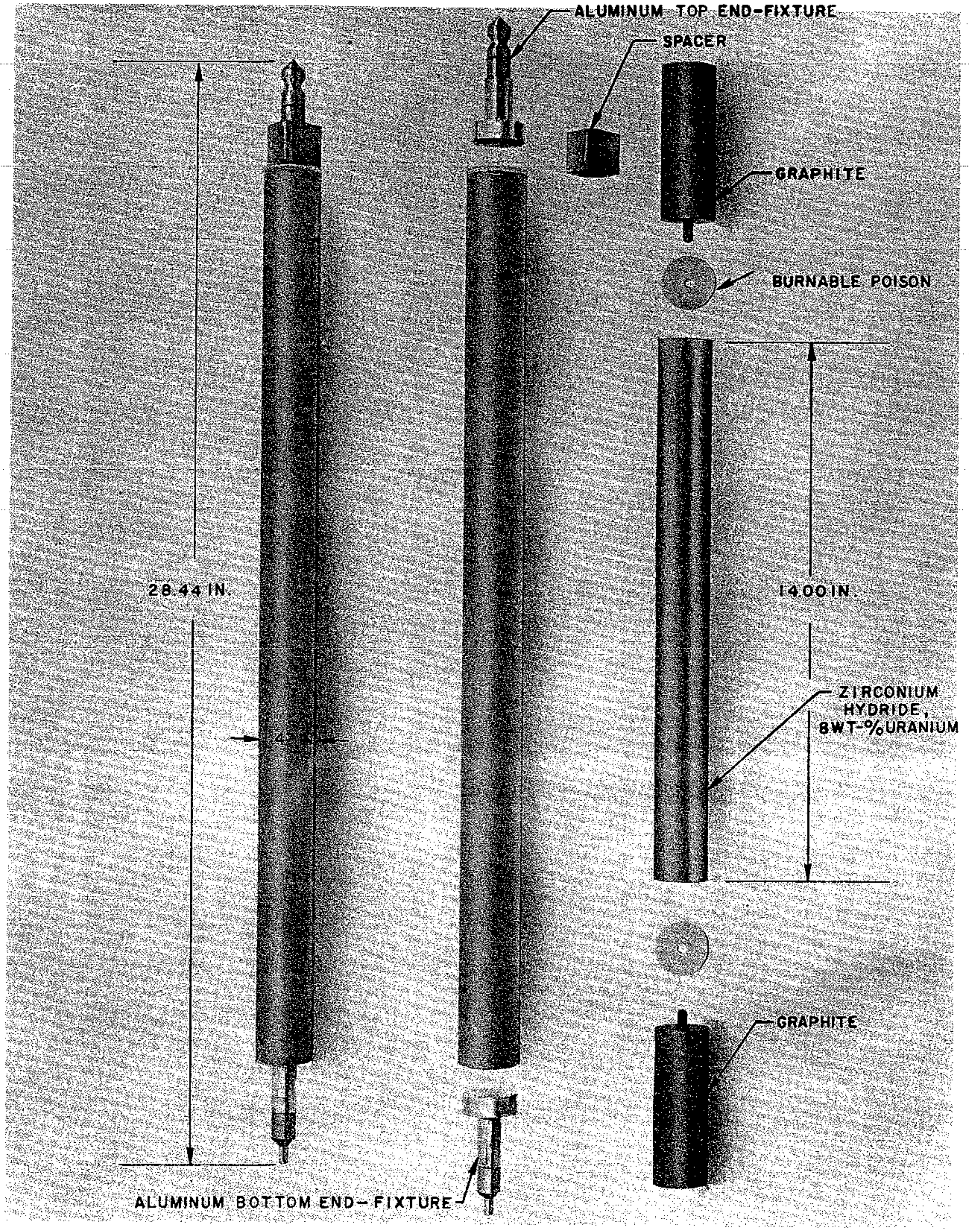


Fig. 12--Fuel-moderator-element assembly

supports the weight of the fuel elements, has holes to receive the lower end-fixtures. The 1.5-in.-diameter holes in the top grid plate space the fuel elements and allow withdrawal of the elements from the core. Triangular spacers on the top end-fixtures allow the cooling water to pass through the top grid plate when the fuel elements are in position. Both grid plates are supported by the core shroud.

CONTROL RODS

Reactor control is provided by four boron-carbide control rods which operate in perforated aluminum guide tubes that are held in place by the top and bottom grid plates. The shim-safety-transient rod has a total worth of approximately 2.7% $\delta k/k$, the two safety rods have approximately 1.5% $\delta k/k$ each, and the regulating rod has approximately 1.5% $\delta k/k$. The control rods are 1.25 in. OD and 20 in. long; the maximum travel of a rod from the full "in" position to the full "out" position is 15 in.

The aluminum control-rod guide tubes fit into standard fuel-element locations and rest on the bottom grid plate by means of a shoulder. The tubes are normally held in position by a locking mechanism that is below the bottom grid plate and is accessible when the control rods are removed.

Safety and Regulating Control Rods

The control-rod drive mechanisms (see Figs. 13 and 14), located on the carriage at the top of the reactor pool structure, consist of a motor and reduction gear which drive a rack and pinion and also drive a potentiometer for position indication. All rod drives have a tube extending to the surface of the water where the dashpot is located. The dashpot-and-rod assembly is connected to the rack through an electromagnet and armature. In the event of a power failure or a scram signal, all of the magnets are de-energized and the safety and regulating rods

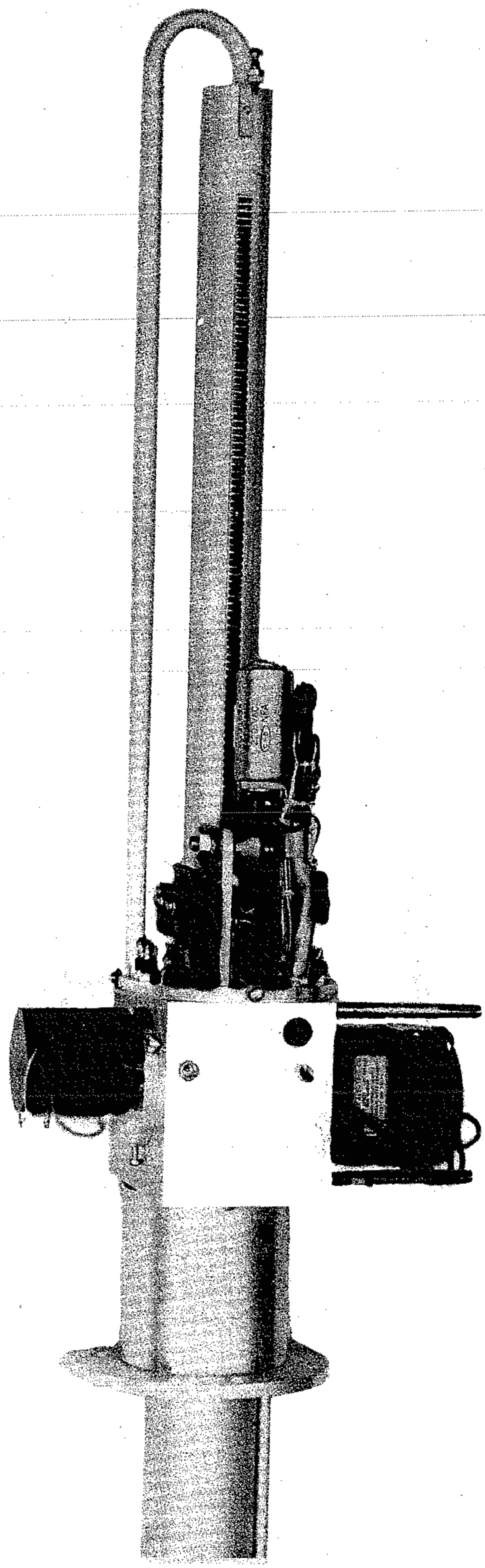


Fig. 13--Control-rod drive mechanism

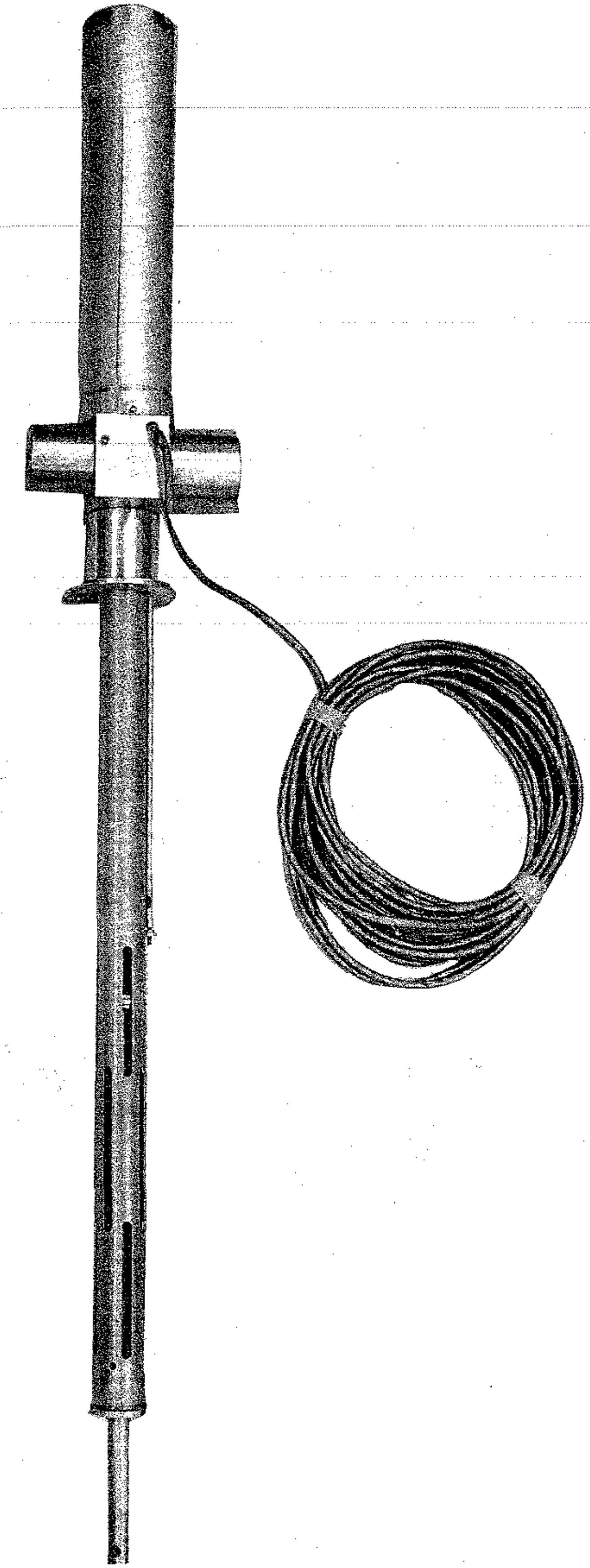


Fig. 14--Control-rod drive assembly

fall into the core. The standard drive is nonsynchronous, single phase, and instantly reversible. Electrical dynamic and static braking on the motor is used for fast stop. Limit switches mounted on the drive assembly indicate the up and down position of the magnet, the down position of the rod, and magnet contact.

The maximum rod withdrawal rate for the safety and regulating-rod drive mechanisms is 12 in./min, which gives a maximum rate of reactivity insertion of 0.04% $\delta k/k$ per second. Continuous rod-position indicators are provided on the regulating rod.

Transient-rod Operation for Pulse and Square Wave

The transient-rod drive (see Fig. 15) used for Mode II and Mode III operation is both pneumatic and electromechanical in operation. The transient rod is fastened to a piston, the stroke of which is determined by the location of an associated cylinder. The positioning of this cylinder is controlled by an electric motor and ball-nut drive. In the de-energized position, the control rod is completely within the reactor core and held there by gravity. Application of pressurized air to the cylinder causes a rapid withdrawal of the rod to a distance determined by prior positioning of the cylinder. The amount of rod that may be withdrawn varies from zero to the rod length required for the maximum permissible step-reactivity insertion, which at no time will ever have a possibility of inserting more than 2.2% $\delta k/k$ in a step insertion of reactivity. Both ends of the pneumatically driven portion of the stroke on this rod are limited by built-in metal-to-metal stops and the control rod will be designed with a reactivity no greater than 2.2% $\delta k/k$ in the length equivalent to that portion of the rod which can be pneumatically driven. In reaching the fully withdrawn position of the cylinder, the rod will be partially removed from the core by metal-to-metal contact and without air pressure applied, so that the amount of reactivity which can be accidentally inserted is limited to 2.2% $\delta k/k$ even though this rod has a total reactivity of 2.7% $\delta k/k$.

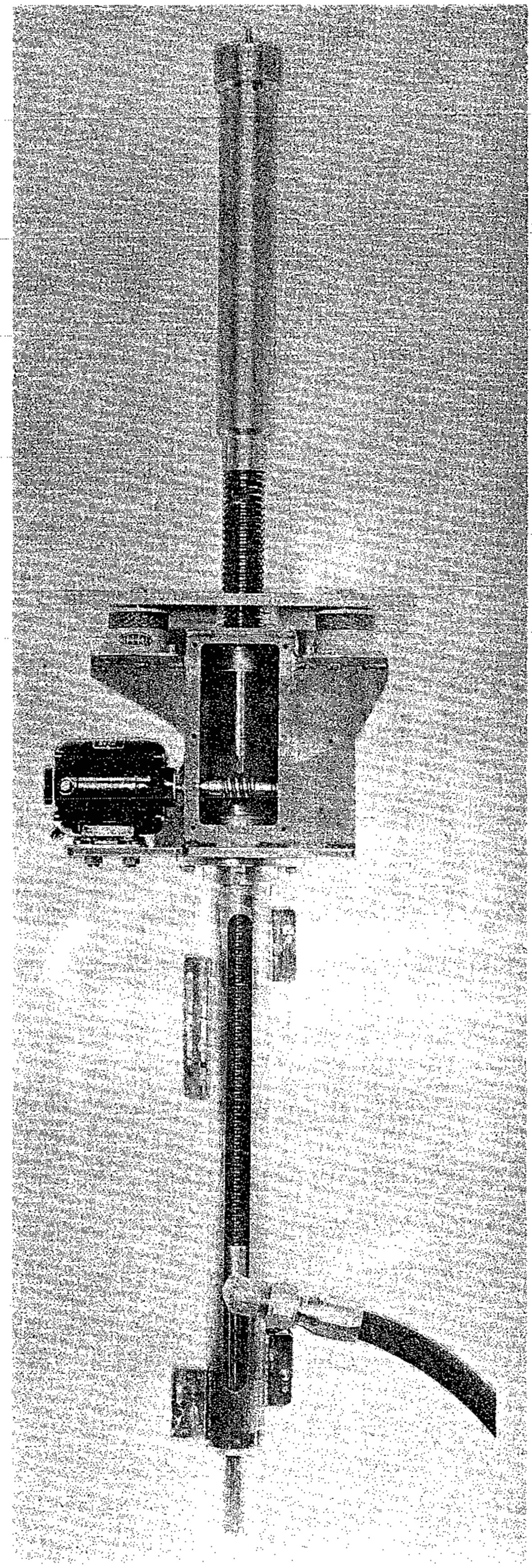


Fig. 15--Pneumatic electromechanical transient rod-drive mechanism

For pulse-type operation (Mode III) the amount of rod worth to be withdrawn is predetermined and, based on a previous calibration, is converted to a preset position of the cylinder. With the air pressure off and the cylinder at this predetermined height, the rod is ready for pulsing operation. The application of 75 psig of air will cause the rod to be withdrawn in approximately 0.1 sec; the exact time will depend on the length of rod travel required. Removal of the air pressure will allow the rod to fall back into the reactor due to gravity.

Preset maximum power operation (Mode II) is quite similar to pulse operation of the rod but has the added feature of closed-loop control of the power level. A short period is established by rapid pneumatic withdrawal of a rod having a maximum worth of 0.66% $\delta k/k$. At the same time that the rod is withdrawn, a closed-loop electronic servo control system is established which continues to regulate the rod position to maintain the preset power level. This electronic control system primarily withdraws the cylinder and the control rod to compensate for the changing negative temperature coefficient of the reactor core.

When the prescribed irradiation time has elapsed, the reactor is shut downautomatically by a preset timer or manually by the reactor operator.

A prototype model of this rod drive is undergoing extensive testing by General Atomic in the TRIGA Mark F.

CORE SHROUD AND SUPPORT STRUCTURES

The reactor core structure consists of a cylindrical shroud, support ring, and a top and bottom grid plate. The shroud is a right circular cylinder, 27 in. high and 19-1/8 in. ID, made of 3/16-in.-thick aluminum. Reactor-coolant-flow openings are located near the top and bottom of the shroud.

The reactor core is supported within the reactor tank such that the lower grid plate is 13 in. above the bottom of the tank. The core support structure, consisting of a 5/16-in.-thick aluminum cylinder approximately 18.9 in. in diameter and 16.8 ft long, connects at its bottom to the core support ring and at its top is bolted to the carriage, thus allowing the core to be translated to various in-line positions within the tank. One 16-in.-wide slot extends the full height of the support structure to allow easy access to the core region.

CARRIAGE

The carriage (Fig. 16) supports the reactor core assembly, the control and pulse rod drives, and an instrumentation harness. The carriage structure is approximately 44 in. long by 30 in. high and supports the rod drives on a mounting plate elevated approximately 3-1/2 ft above the carriage base. This allows clearance above the water surface and thus permits direct-line access to the core components and fuel elements. Four wheels support the carriage on two guide rails which span the top of the tank and are secured to the concrete structure. A two-speed electric motor and a reduction-gear drive enables the core to be moved back and forth between the two extreme positions in the reactor tank, and the driving force is transmitted through a pinion to a rack attached to one of the two rails. The engagement of the wheels is designed to restrain any vertical or lateral displacement of the carriage.

The motor drive is equipped with a position indication system. Remote control and position indication of the carriage are provided at the operating console. The control of the carriage is dependent on the interlock system, which allows movement only when certain conditions are met.

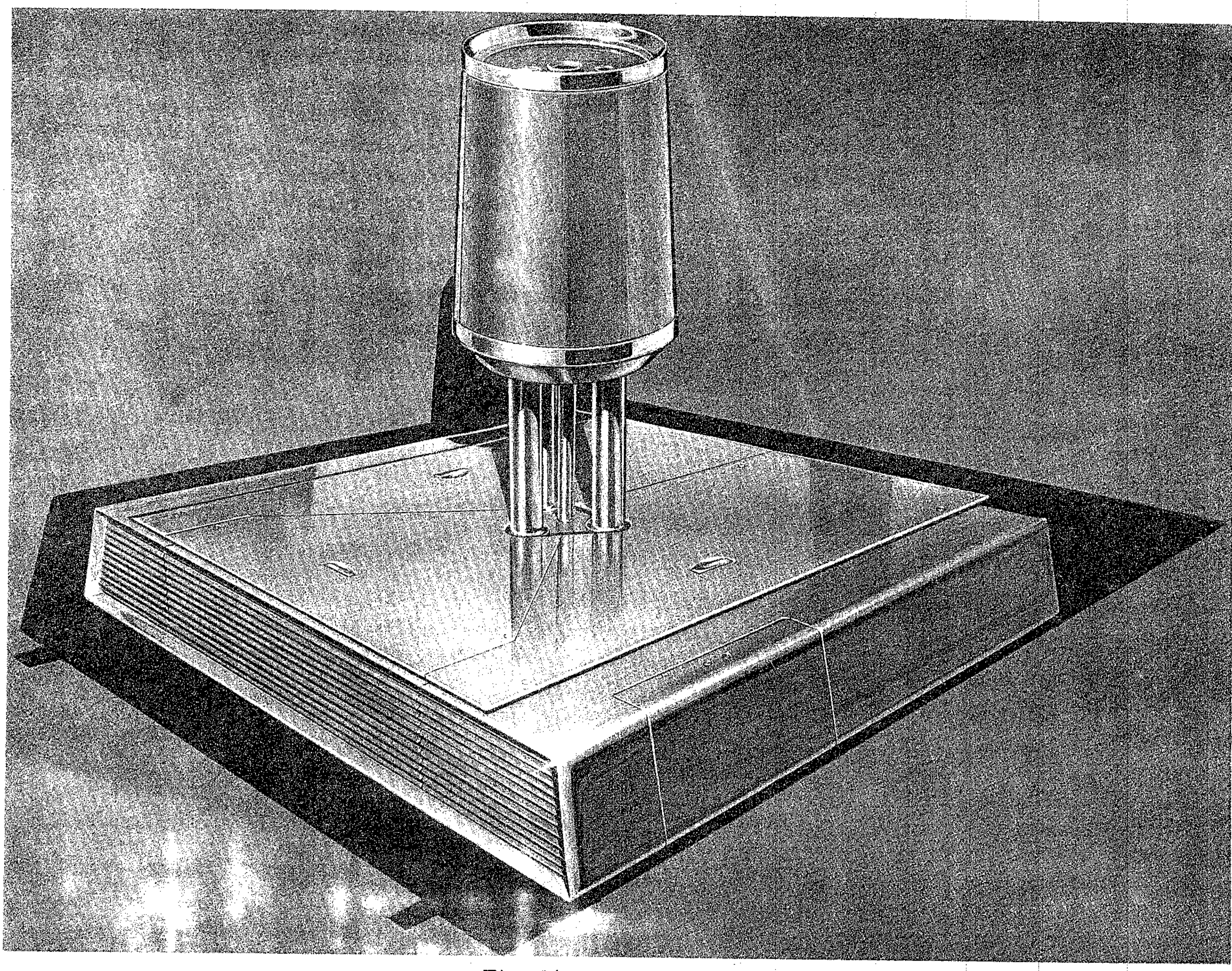


Fig. 16--Reactor carriage

INSTRUMENTATION

Reactor Control

The desk-type control console for the reactor, which is similar to that shown in Fig. 17, has provision for the measurements and control required for the reactor operation. This console is located in the Reactor Control Area on the mezzanine floor of the facility.

Three principal modes of reactor operation are possible:

Mode I (manual) or Mode IA (servo control)--Steady-state operation (100 kw maximum);

Mode II (program control)--Power square wave operation (250 kw maximum);

Mode III (manual control)--Pulsing operation (2200 Mw maximum).

Mode I is employed for manual reactor startup, change of power level, and steady-state operation to 100 kw. Mode IA provides automatic power-level control at all power levels above 1 w by means of a servo amplifier operating the regulating rod.

As indicated in the block diagrams shown in Figs. 18 and 19, power-level information is available for all modes of operation. Switches A, B, C, and D are mechanically coupled and operate from one control knob. With this control in the mode I position, power level is measured from source level to 100 kw by means of a compensated ion chamber which feeds the multirange linear power amplifier which in turn feeds the linear recorder. From a few tenths of a watt to 1 Mw, power is also indicated on the log power channel. Period indication from ∞ to 1 sec is also available from this channel over the same power range.

The linear power channel provides a source interlock signal, which prevents rod withdrawal unless the source strength is adequate. Both a polonium-beryllium and an antimony-beryllium photoneutron source will ensure adequate neutron signal for startup.

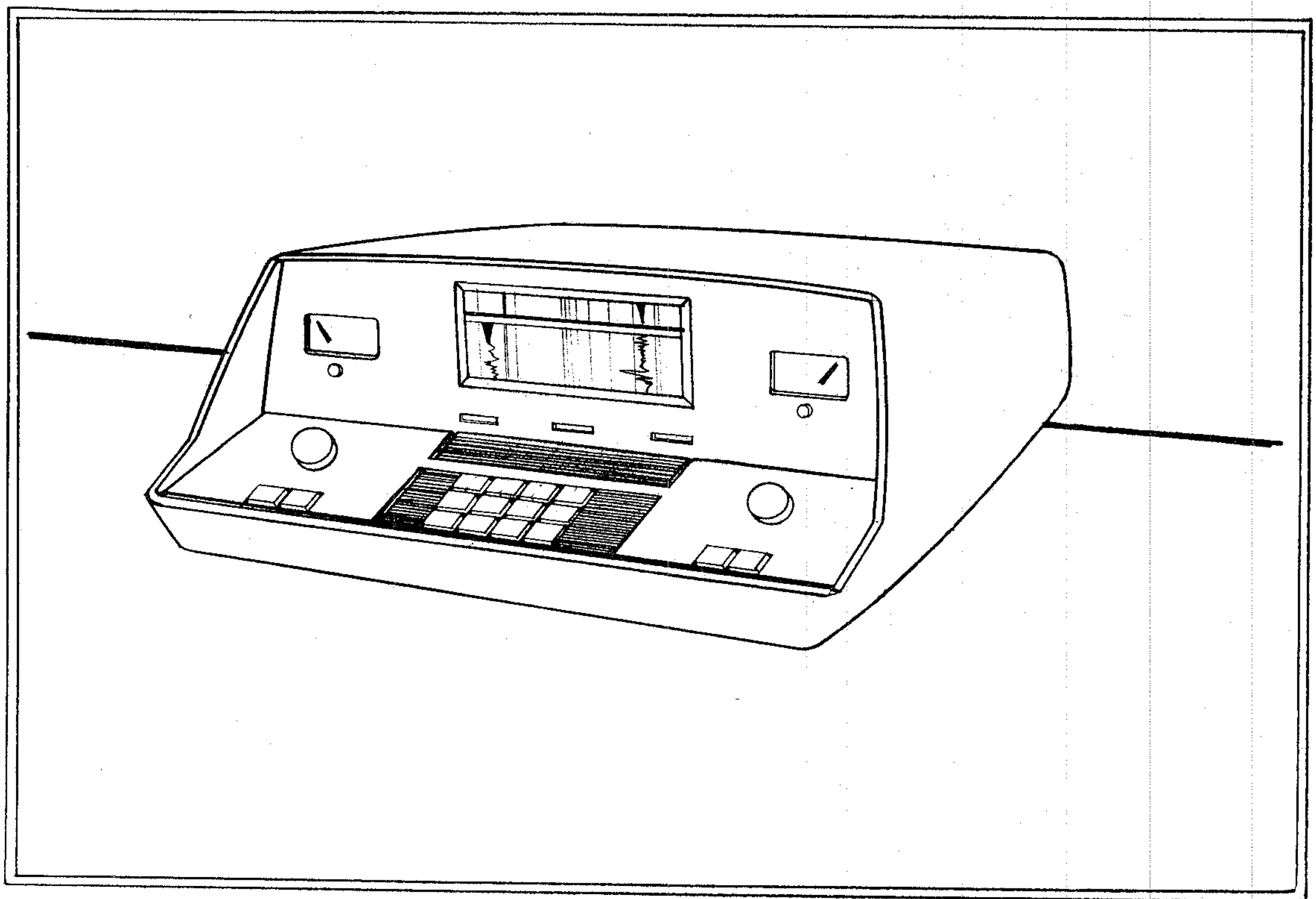


Fig. 17--Reactor control console

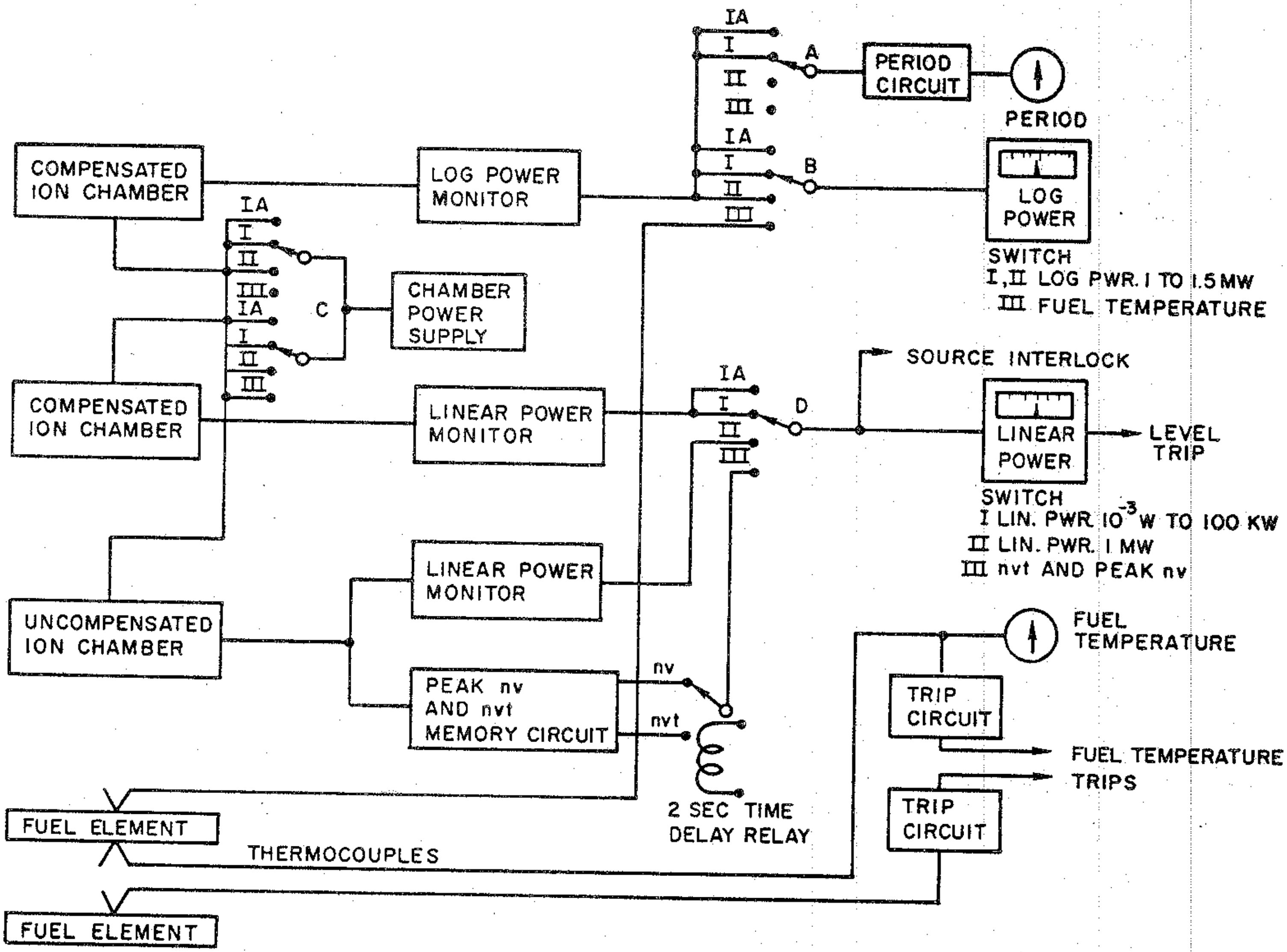


Fig. 18--Diagram of the reactor control circuit

Mode II operation is for producing a rapid rise (in a few seconds) from a low power level to a high power level that is adjustable up to 250 kw, and is automatically stabilized at this preset power level until the reactor is shut down, either manually or by a timer. A control knob sets the servo demand power at the correct value for a given square wave while simultaneously programming the rod-drive mechanism to produce the rapid rise to the desired power level. Power-level measurement is provided by both the log and linear channels for this operating mode. The linear recorder, however, receives its signal from an uncompensated ion chamber that is located in a less sensitive position.

Mode III is for producing high power pulses of short duration by means of the transient rod. As indicated in Fig. 18 all of the low-level neutron chambers are disconnected. The uncompensated ion chamber will give pulse power level indication up to 2200 Mw. The chamber current feeds an integrator, peak-reading, and memory circuit which will (1) measure the peak power of the pulse, (2) measure the integrated neutron flux or nvt, (3) store these data in a memory circuit, and (4) read out each of the values of (1) and (2) sequentially on the linear recorder within a few seconds after the transient is over.

During transient operation (Mode III), fuel-element temperature is recorded, and for all modes of operation, fuel-element temperature is indicated on a meter.

The temperature shutdown system will include two shutdown circuits receiving independent signals from two thermocoupled fuel elements. Four thermocoupled fuel elements will be provided, each of which will contain three thermocouples. The four elements will be located in the central portion of the core. A fuel-element temperature of 530° C will cause a shutdown of the reactor.

In addition to the shutdown system described above, and to assist the reactor operator in assuring that the reactor will not achieve power levels inconsistent with experimental requirements, a linear power level shutdown circuit is also provided. This linear power level shutdown system is so designed that the reactor power is prevented from exceeding 110% of any preset power level selected by a range switch.

Since the DORF reactor will be routinely operated on periods as short as 3 msec during mode III operation, no period shutdown circuit can be technically justified. Mode II operation also involves the step insertion of reactivity, followed by the electromechanically driven insertion of additional reactivity to maintain power level at any desired preset value.

The reactor shutdown is provided by three independently acting shutdown circuits under all modes of operation.

In addition to the above described instrumentation, push-button controls with up-down limit lights for all rods and position indicators for the regulating rod and for the shim-safety-transient rod are provided. Push-button controls and limit lights for the lead door and the core carriage are also provided. The core carriage has a position indicator for intermediate positions. All position indicators are accurate to 0.2% of full travel. Also included on the control console are a key-actuated operation switch, a coolant-on switch, exposure-room-door open-closed indicator lights, and water-gamma-level and temperature alarms. The safety interlock system, described in detail under "Reactor Safety Interlock System," furnishes several inputs to the control console which interlock the console operational controls in such a way that they are active only under proper conditions and sequences.

Additionally, the source interlock signal is provided by the linear power level channel. In the event that the safety interlock devices, the scram and STOP buttons, power failure, or excessive fuel temperature prevent operation, annunciator lights on the console will indicate the source of the fault.

Additional instrumentation is provided for monitoring gamma activity in the water, water temperature, and water conductivity. Slave monitors from these indicators are located in the reactor control area. Remote alarms for excessive water temperature and excessive gamma-ray activity in the water are provided at the operating console, as noted above.

Safety Interlock

An interlock system is provided in order to ensure the safety of personnel within the exposure area and to protect the reactor core and supporting structures from damage by the lead doors. A detailed discussion of this system is given under "Reactor Safety Interlock System."

Auxiliary Systems

Auxiliary communication systems are provided for the facility. The communication systems include television monitoring, audio intercommunication, concrete door position indication, and "reactor on" indicator lights.

The television monitoring system provides for remote monitoring of activities in the exposure room and in other areas in the facility. It includes two tripod cameras, one master control monitor located in the reactor control area, and two portable slave monitors. The cameras are mobile and are provided with remote on/off switch and pan, tilt, and zoom lens controls located at the master control monitor. Supplemental lighting fixtures are provided with each camera.

Intercommunication is provided between the exposure, access, and reactor rooms and in the reactor operator's area.

The concrete-plug-door "IN" and "OUT" indicator lights are located at the reactor control console. Reactor position is indicated near the concrete plug door.

Fifteen experimental instrumentation conduits, ranging in size from 2 in. to 5 in. in diameter, are provided through the shielding above the exposure room so that minimum-length cables can be employed to experimental equipment in the exposure room. Each conduit follows a helical path through the concrete to prevent radiation streaming.

REACTOR TANK AND SHIELDING DOORS

The reactor pool tank is approximately 14 ft in diameter by 9-1/2 ft high. An outline of the tank is shown on Figs. 9, 10, 11.

A lead reflector, having the form of a half-cylinder, is mounted external to the tank on a hydraulic hoist; this reflector may be positioned adjacent to the reactor core or lowered to below the bottom plane of the core. The lead reflector is water-cooled, the water being supplied via an aluminum header to cooling passages approximately 1/8-in. thick by 6 in. wide. These cooling passages are developed in pairs, one for inlet flow and the adjoining one for outlet flow into an outlet header.

Two rotating lead-filled shielding doors are located in the reactor pool to divide the pool into two specific areas. The doors are made of 3/8-in. aluminum plate with 1/4 in. structural members welded into watertight containers. They are approximately 18 in. thick, 4 ft high, and 4 ft wide. Each door is filled with approximately 10,000 lb of chemical lead shot. Gaps between the doors are stepped to prevent radiation streaming through the door.

Each door is supported on a low-friction thrust bearing mounted at the bottom of the tank. These two bearings are designed for maximum life and minimum maintenance during operation in the reactor water. The doors are operated by a single fractional-horsepower motor and reduction drive located in a small pit at the reactor top. Power is transmitted through two drive shafts which are mounted into the side of the carriage tracks. Each door is connected to its drive shaft by a vertical shaft extending up from the top of the door to a bearing mounted on the carriage-track support structure. Gears transmit the driving power from the horizontal shafts on the doors. Operating controls for the doors are located at the control console and operation is restricted through the interlock system--movement can not take place until certain conditions in the over-all reactor operation have been met. Door position is indicated by "all open" or "all closed" lights at the reactor operating console.

WATER PURIFICATION AND COOLING SYSTEM

The reactor water system (see Fig. 7) consists of a surface skimmer, water-monitoring equipment, a fiber cartridge-type filter with pressure gauges, mixed-bed demineralizers, flow meter, pump, heat exchanger, and associated piping and valving. This system has several functions. The water system provides for cooling the reactor and has a capacity of 100 kw of continuous heat removal while maintaining the reactor water temperature at $90^{\circ}F$. The demineralizers maintain the specific conductivity of the reactor water at about 2 micromhos, minimizing corrosion effects, remove radioactive material from the reactor water, and help to maintain optical clarity of the water.

FISSION-PRODUCT MONITOR

A low-level fission-product water monitor, upstream of the heat exchanger and the demineralizer, detects any damage to the fuel elements that may cause fission products to be present in the water. A Tracerlab MWP-1A Fission Product Water Monitoring System (or equivalent) detects the radioiodine isotopes from fission products accumulated in a resin ion exchange column. This device detects radioiodine in concentrations of 10^{-6} $\mu c/cm^3$.

authority and experience to maintain the safety of the operation. Adequate instrumentation for personnel and area monitoring will be provided the health physicist to carry out these duties.

Personnel Monitoring

A film-badge service is to be provided for the personnel working in the facility. The badges will have both neutron-sensitive and gamma-ray-sensitive film and will be processed quarterly, or a badge can be processed immediately in the event an exposure has occurred to an individual. Pocket dosimeters will supplement the film badges. The dosimeters will be read on a semiweekly schedule during the initial phases of facility operation; however, after the personnel have been suitably trained, this schedule will be on a weekly basis. The pocket dosimeters will measure the dosages of both thermal neutrons and gamma rays; any indication by the thermal-neutron detector will require that the neutron badge be processed immediately.

Area Monitoring

Two fixed, audible alarm, gamma-ray monitors located on the second floor of the building, 9 portable survey instruments, and 2 high-level ion chambers located in the reactor exposure room will be used for area monitoring. Readings from the fixed instruments will be recorded and an audible alarm will be actuated if the activity exceeds a preset level.

Personnel- and Area-Monitoring Instruments

Listed below are the various instruments to be used for monitoring personnel and various areas in DORF.

	Quantity	Description
Personnel Monitoring:		
	1	Dosimeter Charger Reader (equivalent to Victoreen Model 687)

	Quantity	Description
	50	Dosimeter, indirect-reading (equivalent to Victoreen Model 362)
	1	Hand and foot monitor (equivalent to Eberline Model HFM-2)
Contamination-boundary Monitoring:		
	1	Hurst Probe (equivalent to Reuter-Stokes RSN-48)
	4	G. M. Counters (equivalent to Eberline RM-1)
Area Monitoring:		
	1	Remote area-monitoring system with accessory equipment (equivalent to Victoreen Model 712)
	2	Air particulate monitor (equivalent to Victoreen Model 900-56)
Survey Instruments:		
	3	Radiation monitor, portable with 6 ft rigid probe and adapter (equivalent to Victoreen Model 2022)
	3	Alpha, Beta, Gamma Survey Meter (equivalent to Victoreen Model 740)
	3	Gamma Survey Meter, Miniature (equivalent to Victoreen Model M-50)
	1	Fission-product Water Moderator (equivalent to Victoreen Model 900-146)
	1	Scaler with high-voltage power supply (equivalent to Victoreen Model 764 and accessories)
	1	Gas-flow counters and shield (equivalent to N. Wood Model K-3)
	1	Fast-neutron dosimeter with built-in calibration (equivalent to Fairport Instrument Model 410)

Additional equipment are (1) remote-control lens turrets, (2) remote-controlled Auto-Zoom lens (equivalent to RCA MI-36189), (3) remote-controlled iris (equivalent to RCA MI-36140), (4) remote-control pan and tilt (equivalent to RCA MI-36110), (5) utility monitors (equivalent to RCA 14-in. IM-4A series), and (6) an anemometer with wind speed and direction transmitter and recorder (Braum 63906 Aero-Vane type, or equivalent).

Chapter 4

EXPERIMENTAL FACILITIES

EXPOSURE ROOM

The exposure room is adjacent to the core when the core is moved into a position near the tank wall (see Fig. 4). The dimensions of this room are approximately 20 ft by 20 ft by 8 ft high. One foot of wood lining will cover all six surfaces of the room to minimize the effects of induced secondary radiation from the surrounding concrete shielding; the only exception is the region where the aluminum projection of the reactor pool extends into the room. The floor will have plywood over the thick wood shield to facilitate maintenance of the surface and to permit unusual decontamination procedures. Normal decontamination will be accomplished by washing down the epoxy resin coating of the floor and walls.

Access to the exposure room will be through a rolling plug door whose minimum opening is 5 ft 8 in. wide by 6 ft 6 in. high. The plug will be tapered and stepped to interrupt radiation streaming paths. The movement of this door is restricted through a system of safety interlocks which have been previously described.

Maximum Radiation Doses in Exposure Room

For a 24 Mw-sec pulse, the leakage dose to the exposure room at the reactor centerline is shown in Table 4.1.

In the high-level exposure thimble, the neutron dose will be 3.9×10^5 rads for a 24 Mw-sec pulse, and the gamma dose will be 6.3×10^5 rads, which gives a total of 10^6 rads.

The energy distribution of the neutron dose at the tank surface (no lead reflector), as measured with threshold detectors, is

10 kev to 750 kev, 33%;

750 kev to 1.5 Mev, 16%;

1.5 Mev to 2.5 Mev, 14%;

Greater than 2.5 Mev, 37%.

Irradiation experiments in the exposure room may be performed at various distances from the reflector, to vary the dosage. At about 55 cm from the reflector surface, the dose will be down by a factor of ten. At a distance of about 1 meter from the core center, the total dose (neutron and gamma) will be 15,000 rads/pulse.

Table 4.1

LEAKAGE DOSE TO EXPOSURE ROOM

	Neutron Dose (rads)	Gamma Dose (rads)	Neutron-to-Gamma Ratio
Without Lead Reflector			
Tank surface	8.7×10^4	1.4×10^5	0.62
3 ft from surface	3.3×10^3	7×10^3	0.47
With 2-in. Lead Reflector			
Reflector surface	3.1×10^4	1.3×10^4	2.5
3 ft from surface	1.5×10^3	0.8×10^3	1.9

Activation in Exposure Room

Argon concentrations and doses have been calculated and are presented in Appendix I. In summary, these concentrations and doses for an external gamma dose from a 20-min exposure immediately after shutdown in the fast-neutron exposure room are 12.3 mrep at 100 kw steady-state and 4.1 mrep for repetitive 24 Mw-sec pulses.

With the high-intensity neutron and gamma-ray fluxes from the reactor, activation of the walls in the exposure room is considerable. Calculations were made to determine the activation of the permanent aluminum tank wall, and of the wood lining in this room.

Three reactions were considered in the tank wall: Al^{27} (n,α) Na^{24},

Al^{27} (n, p) Mg^{27}, and Al^{27} (n, γ) Al^{28}. Calculations were made for a steady-state power level of 100 kw, short 250-kw runs, and 20 Mw-sec pulses.

At steady-state, 100-kw operation, the activity of Al^{28} predominates, contributing about 6 r/hr at 1 meter from a 2-in. lead reflector surface. Since the half life of Al^{28} is 2.3 min, its radioactivity decays rapidly. After 20 min from shutdown, radioactivity from Al^{28} is less than that from Na^{24} and from Mg^{27}. For Na^{24}, the half life is 15 hr and its contribution is 25 mr/hr at 1 meter from the shield. Mg^{27} radioactivity contributes about the same amount. High power level operation will give proportionately higher doses.

The 24 Mw-sec pulse results in activation of Al^{28} and Mg^{27} to a lesser degree than the 100-kw operation. Pulsing the reactor at the maximum rate (5 pulses per hr) for extended periods of time is equivalent to 30-kw steady-state operation and results in doses approximately one-third that calculated for the 100-kw steady-state operation.

Calculations indicate that if the 2-in.-thick lead shield of the exposure room is removed, all doses from the aluminum tank wall will increase by a factor of about 10, with Na^{24} the chief contributor.

The calculations on the activation of the wood lining on the fast-neutron exposure-room walls were conservatively made, assuming that 0.4 wt-% of the wood is ash, of which 1 wt-% is sodium. The dose rate from the active sodium in the wood at the center of the room, after shutdown from 100-kw operation for one shift operation, will be about 10 mr/hr. Activity of other constituents of the wood was found to be negligible.

An upper limit estimate of the dose rate in the center of the exposure room, from activation of the sodium in the concrete behind the wood, was made. This dose rate will be about the same as the dose rate from the sodium in the wood, i.e., 10 mr/hr. The steel used for reinforcing the concrete has been designed so that the dose rate from the manganese impurities in the steel is smaller than that from the sodium in the concrete.

Table 4.2 summarizes the doses from various components which give a total dose of 26 mrem/hr, 1 meter from the wall of the aluminum tank, for repetitive 24 Mw-sec pulsed operation 20-min after shutdown.

Table 4.2

	Dose Rate (mrem/hr)		
Component	Tank Wall	Room Wall	Air
Argon	--	--	4
Na^{24}	8	--	--
Wood	--	3	--
Concrete	--	3	--
Mg^{3}	8	--	--
Total	16	6	4

TYPES OF EXPERIMENTS

The types of experiments proposed for this facility will be performed in such a way that any one experiment, or all experiments, will contribute no more than 1.0% $\delta k/k$. The nature of the experiments to be conducted is described below.

Pool Irradiation

Irradiation of small waterproofed samples or specimens may be performed within the reactor pool. These irradiations would take place with the reactor isolated from the fast-neutron exposure room by the lead shield doors. Either the pulsing or steady-state operating characteristics of the reactor can be utilized. The pool facility can also be used to produce radioisotopes and to perform activation analysis. Experimental capability will exist within the pool for irradiating chemical compounds and solutions, small electronic and mechanical components, materials associated with chemical dosimetry, foils, and miscellaneous components. Most of these experiments will be small and will be sealed in individual waterproof capsules.

Exposure Room

The size of experimental configurations in the exposure room is limited by the dimensions of the rolling-plug-door opening. Irradiation studies may be performed on complete electrical or electronic systems.

Variation of the radiation intensity and gamma-to-neutron ratios may be obtained by selection of the specimen position, the shield configuration, and the reactor position.

Specimens may be placed anywhere within the room or in close proximity to a high-level exposure thimble. The thimble is a special indentation in the reactor tank, which extends into the outer circle of the fuel elements (the F-ring), thus creating a window for the escape of high-energy neutrons.

Gamma-to-neutron ratios may be modified by a water-cooled lead shield, which can be raised or lowered by a hydraulic lift. In the raised position, the shield will attenuate gammas to reduce the gamma-to-neutron ratio. An additional feature of this shield is a removable lead plug which covers the high-level exposure thimble, so that with the shield raised, collimation is available for shaping the radiation field. The water-cooled lead shield also serves to attenuate the radiation from induced activation of the aluminum of the tank, thus allowing longer working times for personnel in the fast-neutron exposure room. The movement of the shield is permitted only by the proper safety interlock sequence.

The gamma-to-neutron ratio, as well as the neutron spectrum, can be modified by moving the reactor from its closest approach of 1 in. from the tank wall to a position about 12 in. further into the water. This movement is restricted by the interlock system to prevent interference with the rotation of the lead shield doors.

Chapter 5

PERFORMANCE CHARACTERISTICS

The TRIGA Mark F reactor, which is the prototype for the DORF reactor, has been operating since July 2, 1960. A view of the Mark F core is shown in Fig. 21. The latest available data from the research and development program are given below.

SUMMARY OF TRIGA MARK F TEST RESULTS

The TRIGA Mark F reactor, since going critical, has operated at various steady-state power levels up to 1 Mw and has already been pulsed over 160 times, the largest step reactivity insertions being slightly less than 2.2% $\delta k/k$. The operating characteristics of the TRIGA Mark F have been determined to be substantially the same as those of the original Torrey Pines TRIGA Mark I reactor. Steady-state experiments conducted at 1 Mw after numerous periods of pulsing operation indicate that the maximum fuel temperatures will not exceed 530°C.

The transient operating characteristics of the TRIGA Mark F are essentially identical to those of the original Torrey Pines TRIGA reactor. In particular, the period-versus-peak-power curves of the two TRIGA reactors are almost identical, indicating very similar shutdown coefficients. The periods resulting from 2.2% $\delta k/k$ step reactivity insertions are somewhat shorter than those encountered in the Torrey Pines TRIGA, as would be expected in view of the shorter neutron lifetime. The largest pulses to date, slightly less than 2.2% $\delta k/k$, have had a peak power of about 2000 Mw and a pulse width at half maximum power of approximately 10 ms, and have resulted in a prompt energy release of approximately 24 Mw-sec. The maximum fuel temperatures during transients have been less than 530°C for pulses terminated after 1 sec, and the maximum fuel temperatures at

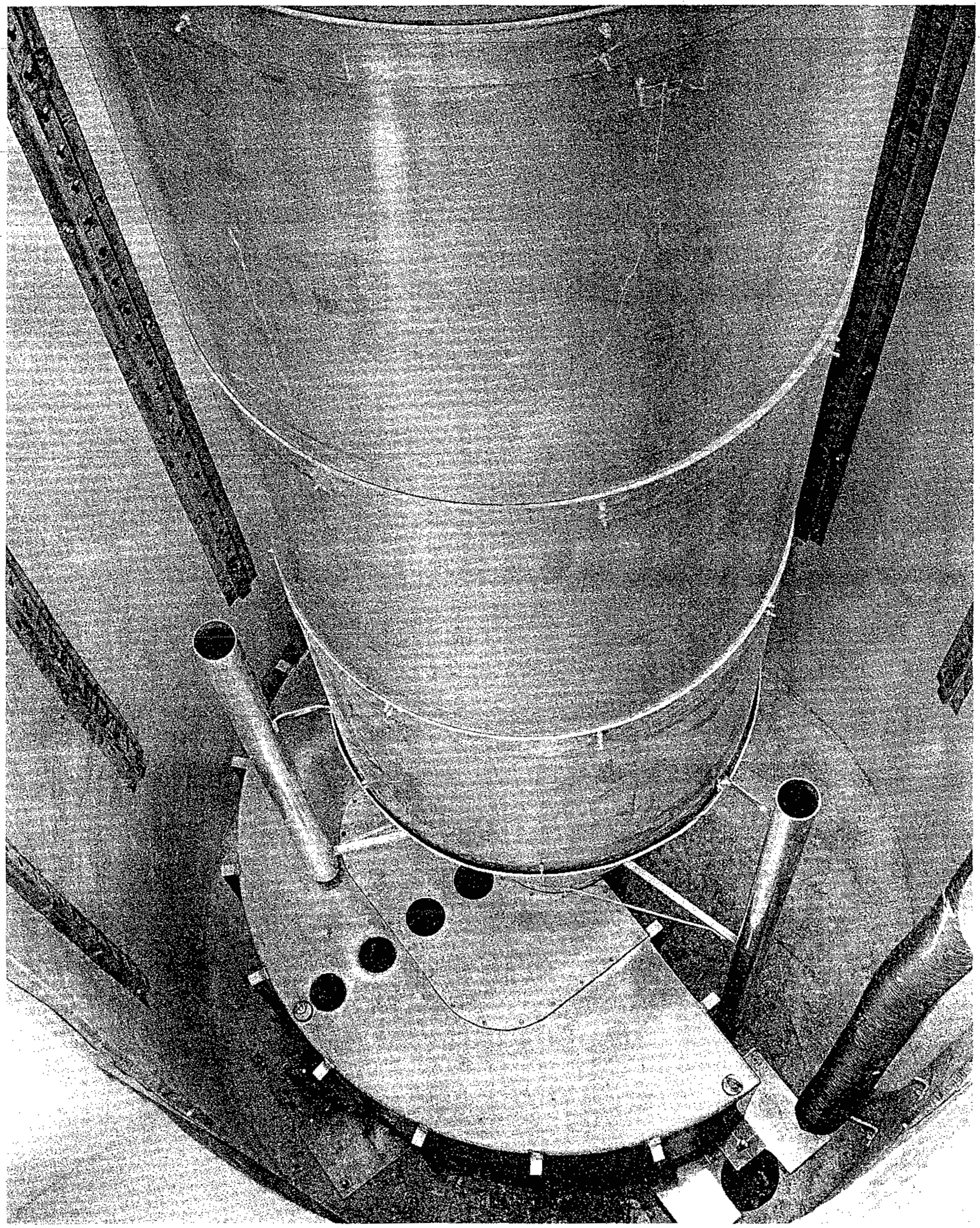


Fig. 21--Reactor core and suspension system of the TRIGA Mark F

1 Mw for steady-state operation were also less than 530°C. The reactor parameters for the TRIGA Mark F are given in Table 5.1.

Table 5.1

TRIGA MARK F REACTOR PARAMETERS

Core:	
Number of fuel elements	85
Active-lattice dimensions	17 in. diam. × 14 in. long
Nuclear characteristics:	
Initial excess reactivity allowance	2.9% δk/k
Fuel inventory	3.2 kg
Reactivity value in control system, % δk/k:	
Safety rods (2)	1.5 each, 3.0 total
Regulating rod	1.5
Shim-safety-transient (max)	
Pneumatically driven portion	2.2
Motor-driven portion	0.5
Total	2.7
Total control-system reactivity (4 rods)	7.2
Void coefficient of reactivity in core	-2×10^{-3} (% δk/k)/(% void)
Prompt neutron lifetime	44. μsec
Energy shutdown coefficient	$2.6\pm0.2\times10^{-5}$ watt^{-1} sec^{-2}
Thermal characteristics:	
Maximum fuel temperature	530°C during transient
Reactor shutdown fuel temperature	530°C

DETAILED RESULTS OF TRIGA MARK F TESTS

General Atomic was issued an operating license for the TRIGA Mark F reactor on July 1, 1960, and the reactor attained its first criticality on July 2, 1960, with a fuel loading of 74 conventional TRIGA uranium—zirconium hydride elements containing 2720 g of U^{235}. Initial criticality was reached with the reactor core surrounded by an infinite water shield, which is equivalent to the reflector conditions which will exist in the DORF-TRIGA reactor when the reactor is used in the pool irradiation facility.

The TRIGA Mark F research and development program had as one of its objectives the determination of experimental changes in reactivity that will occur in the DORF-TRIGA reactor when the reactor is moved into a position adjacent to the exposure room. Such a movement will change the reactor reflector from infinite water to approximately 2 in. of water on 180° of its perifery. Measurements made by General Atomic during subcritical experiments using TRIGA-type fuel elements indicated that a reduction of 0.8% $\delta k/k$ was associated with this change in position. A condition equivalent to the reactor's being adjacent to the exposure room was simulated with the TRIGA Mark F by using a gas- and styrofoam-filled tank (hereafter called the void tank) as shown in Fig. 22. The result of this test reveals that placing the reactor adjacent to the void tank causes a maximum decrease in reactivity of 0.44% $\delta k/k$. The addition of an air-filled thimble in conjunction with the void tank decreases the reactivity an additional 0.23% $\delta k/k$, giving a total decrease in reactivity of 0.67% as compared with an infinite water reflector.

The typical relative worth of fuel elements is dependent on their position in the core. In the outer rings fuel worth is influenced by the presence or absence of elements in the adjacent fuel positions. The TRIGA Mark F grid plate contains six concentric rings of fuel-element positions (B through G rings), as compared with the five rings of fuel positions proposed for the DORF-TRIGA (B through F rings).

The fuel worth as measured in the Mark F reactor is given in Table 5.2.

Steady-state Power Experiments

In the research and development program conducted by General Atomic relative to the DORF-TRIGA reactor, considerable data have been collected at various steady-state power levels up to 1 Mw. In order to calibrate the ion chambers for power level, the heating rate was determined with known energy input to the pool in experiments using electrical heaters

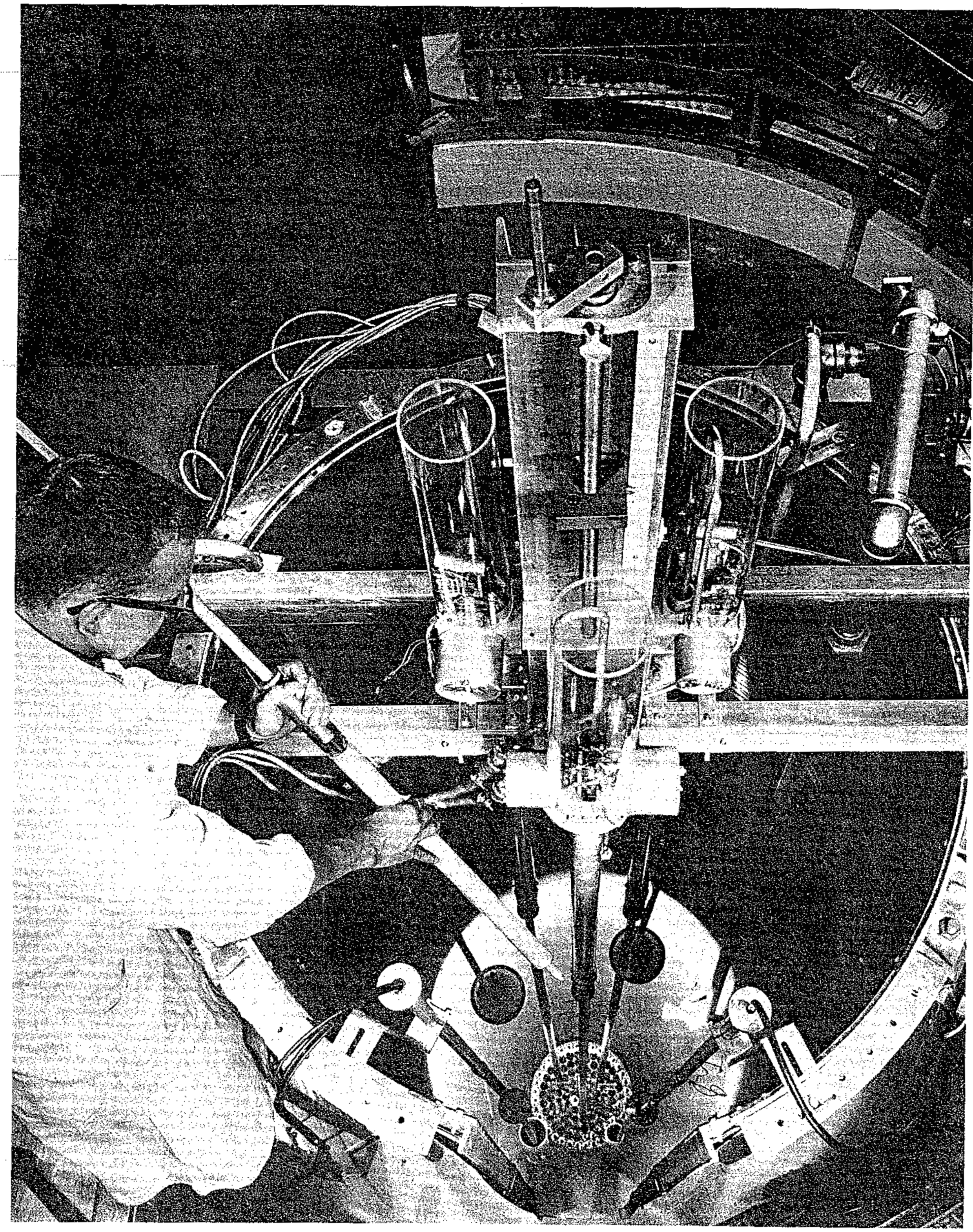


Fig. 22--Reactor tank and core suspension system of the TRIGA Mark F

as a heat source. Once the heating rate of the pool was known, the necessary ion chambers were calibrated for power-level indication for a known rate of temperature rise. During steady-state experiments, power level was monitored by two and sometimes three ion chambers.

Table 5.2

FUEL WORTH IN MARK F REACTOR

Ring Location	Maximum Fuel Positions In Ring	Worth of Typical Fuel Element Compared with Water	
		$	% $\delta k/k$
B	6	0.87	0.64
C	12	0.71	0.52
D	15	0.62	0.45
E	24	0.48	0.35
F	30	0.34	0.25
G	36	0.20	0.15

The Mark F reactor power-coefficient data are shown in Figs. 23 and 24. Figure 23 shows the reactor power versus reactivity curve for power levels up to 200 kw and for three different reactor-core-loading and void-tank configurations. It is seen from Fig. 23 that the void tank, which simulates the reactor exposure room in the DORF, increases the reactor power coefficient by about 10%.

The reactor power versus reactivity curves of the core before and after numerous transients of 2.2% $\delta k/k$ are compared in Fig. 24. The data for the Mark F reactor indicate an increase of about 20% in the power coefficient after the reactor has been subjected to numerous transients. A comparison of this value with results previously obtained in the Torrey Pines TRIGA Mark I reactor and experience with General Atomic's TRIGA Mark I reactor indicates that additional transients of up to 2.2% $\delta k/k$ will not cause a significant further alteration in the curve. This has been attributed to an increased gap between the fuel-element cladding and the fuel material

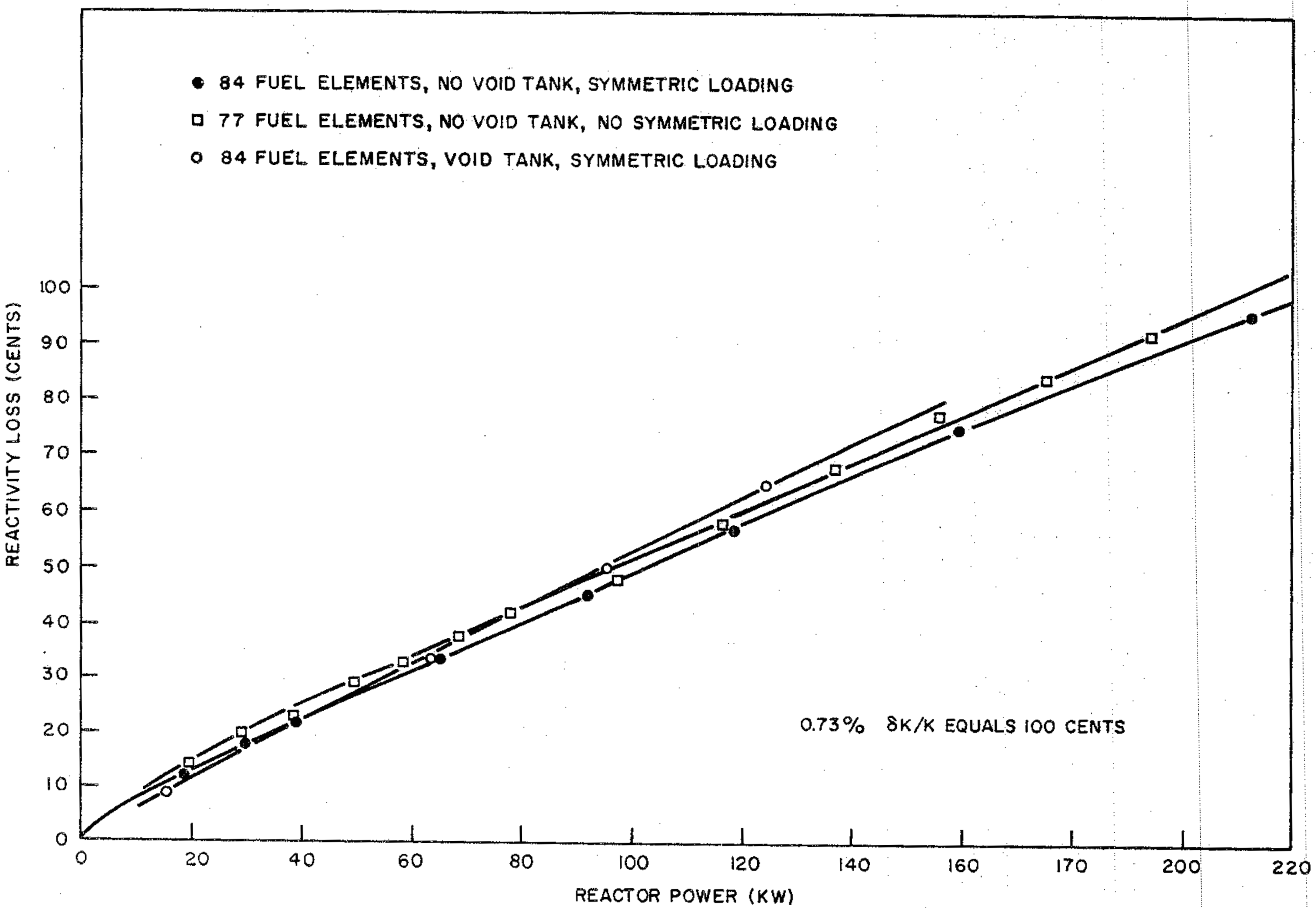


Fig. 23--TRIGA Mark F reactor-power coefficient (before pulsing)

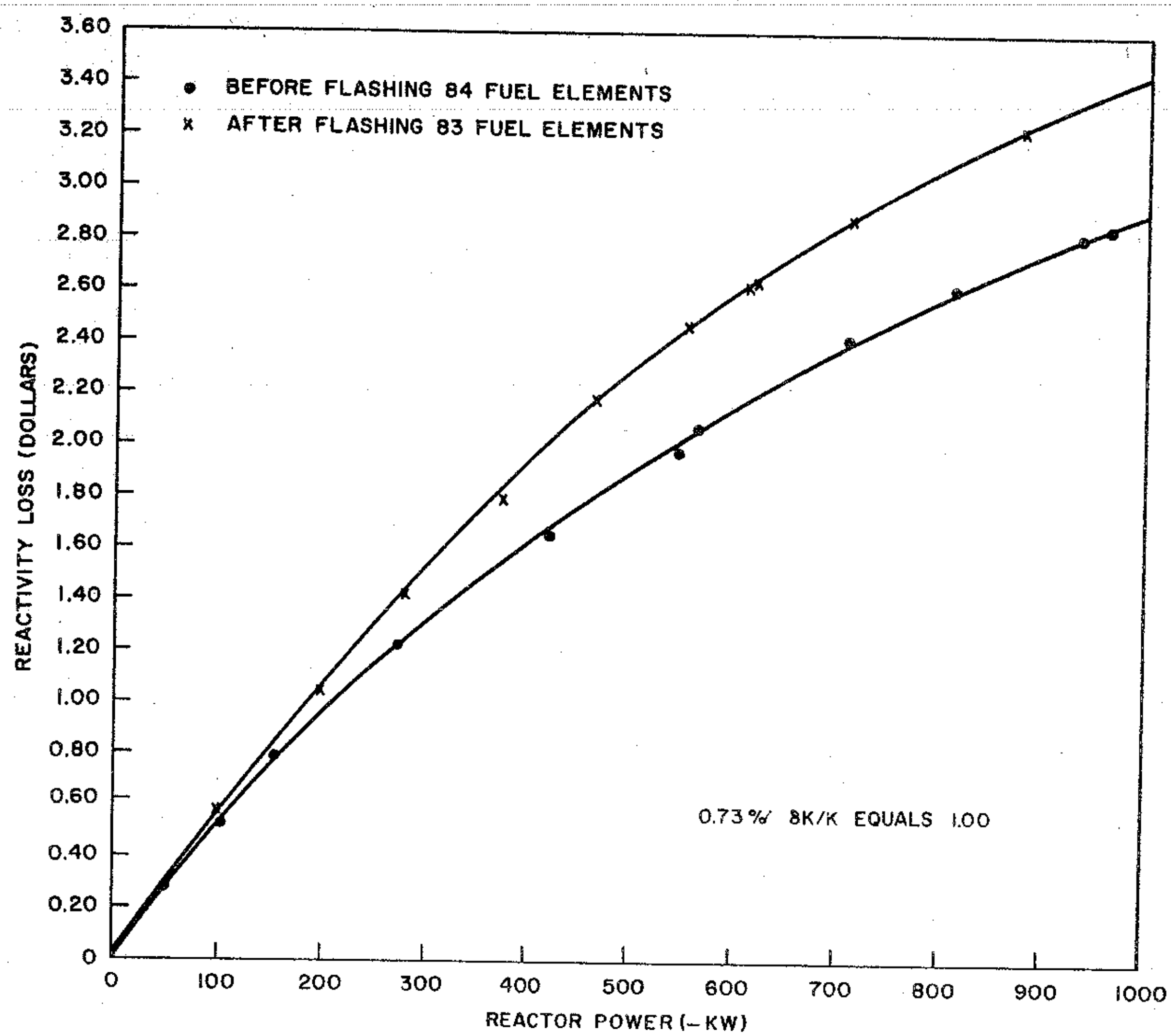


Fig. 24--TRIGA Mark F reactor-power coefficient (void tank not used)

which occurs as a result of the initial transient and results in a slight change in the heat-transfer coefficient of the fuel element.

Data were obtained on the temperature of several fuel elements as the TRIGA Mark F reactor was operated at various steady-state power levels. The maximum temperature rise in a central fuel position is plotted versus reactor power for the same fuel element before and after numerous transients of approximately 2.2% $\delta k/k$ in Fig. 25. The increase in measured fuel temperature after the transients is further support of the theory that the over-all fuel-element heat-transfer coefficient changes slightly during the initial stages of transient operations. However, it is significant that the results of the TRIGA Mark F tests to date have given maximum fuel temperatures at 1-Mw steady-state operation of considerably less than the 530°C fuel temperature that is being established as the upper limit for the DORF-TRIGA reactor. The basic design of the TRIGA fuel element is such that some small variation of heat-transfer coefficient from element to element would be expected. The data collected in the TRIGA Mark F test program to date permit a comparison of power level versus fuel temperature for three elements, as shown in Fig. 26.

The isothermal temperature coefficient (bath coefficient) of the TRIGA Mark F reactor was measured following a high-power-level calibration run during which the reactor pool water temperature was raised to 60°C. The critical point was measured as the reactor water was cooled to ambient temperature. Corrections were made for the experimentally determined decay of xenon poisoning which took place during the same time interval. The results of this experiment are shown in Fig. 27. It is seen from this figure that the isothermal temperature coefficient is essentially zero from 20° to 40°C, and that the coefficient becomes negative at higher pool temperatures.

Transient Operation

A major part of the research and development program performed by General Atomic consisted of a series of step reactivity insertions using

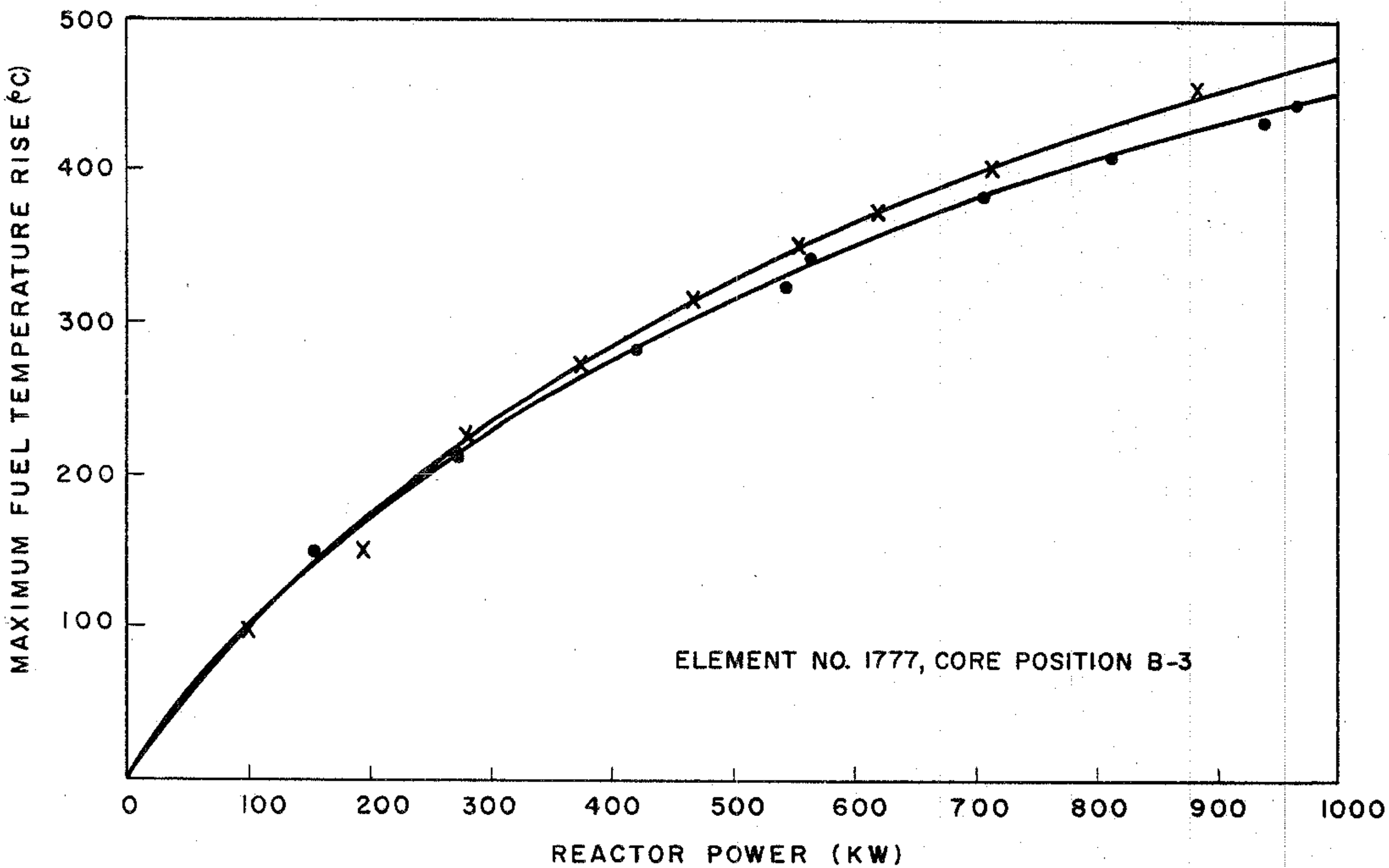


Fig. 25--TRIGA Mark F maximum fuel temperature rise

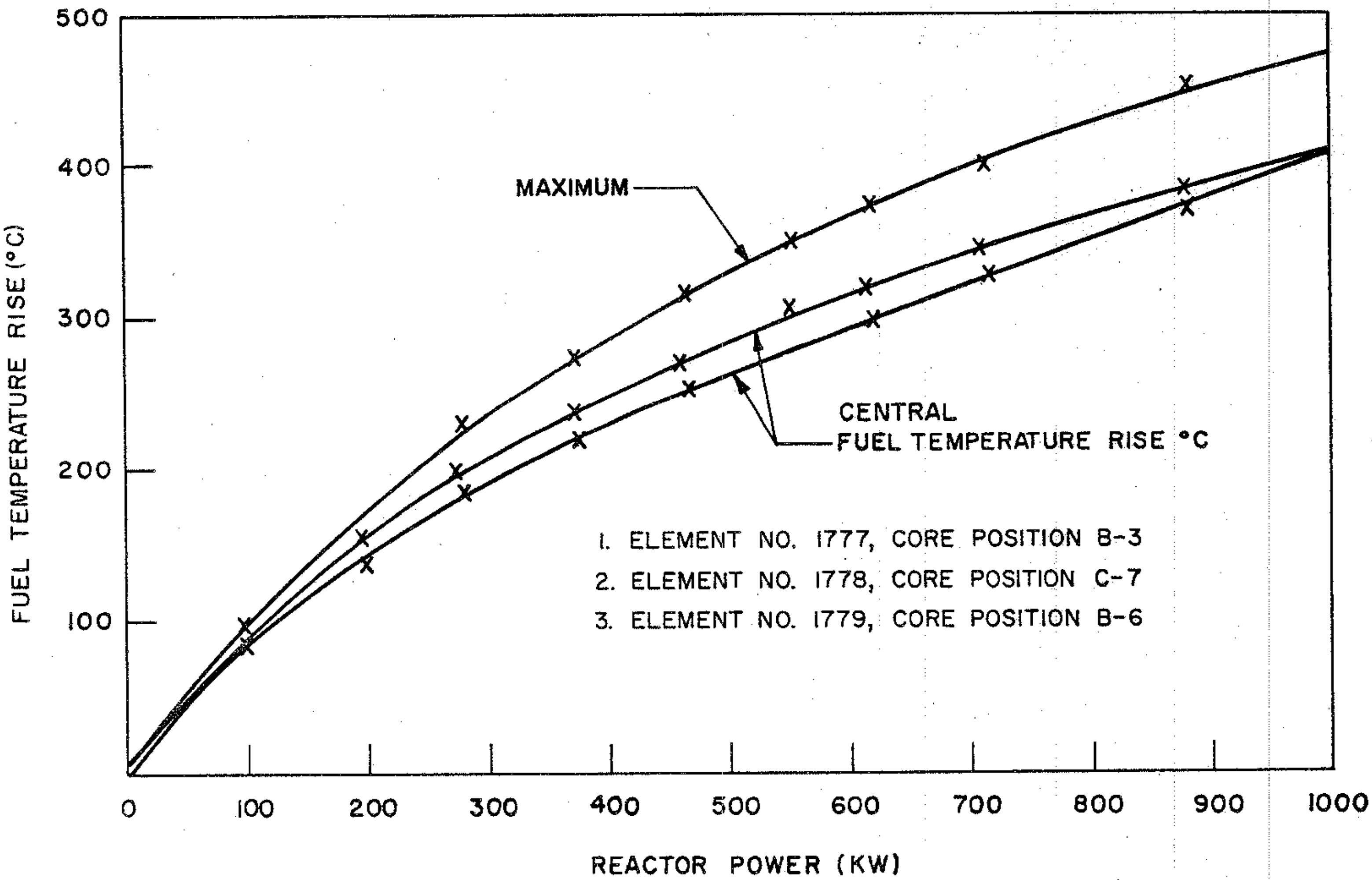


Fig. 26--Central fuel temperature rise, TRIGA Mark F

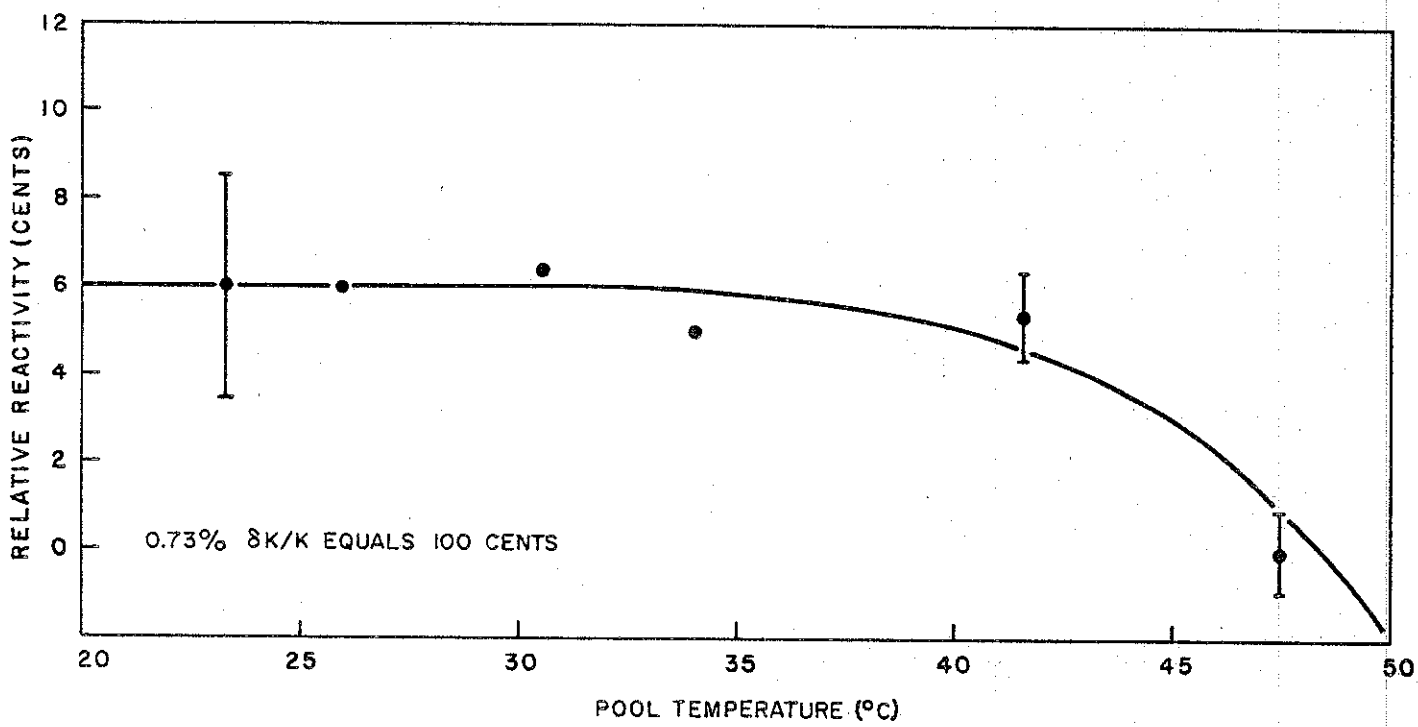


Fig. 27--Variation of reactivity with pool water temperature, TRIGA Mark F

the TRIGA Mark F to determine the transient characteristics of this type of reactor. The reactor parameters of principal interest included reactor power as a function of time, fuel-element temperatures, reactor pulse shape, and integrated energy per pulse.

The reactor transients were initiated by the rapid withdrawal of the central control rod by a pneumatic cylinder similar in principle to the control rod to be used in the DORF-TRIGA reactor. The amount of reactivity insertion is determined prior to the transient by manually adjusting the initial position of the transient rod. Before initiating a step insertion of reactivity, the reactor is brought to criticality at a power level of approximately 100 w. During the transient experiments, data were collected on as many as five ion chambers for power-level information, and numerous temperatures were recorded using a fast-response, 36-channel, galvanometer recorder. These data have permitted the development of information on pulse intensity, pulse shape, and fuel-element temperatures.

Both uncompensated ion chambers and fission current chambers are satisfactory for measurements during transient experiments. Compensated chambers are not usable in transient experiments, since the apparent power is strongly dependent on the compensating voltage. The power registered by uncompensated chambers is believed to be accurate, inasmuch as no change in apparent peak power was noticed when the ion-chamber sensitivity was changed by a factor of thirty. Ion chambers are located several feet from the reactor core at a position where the sensitivity is 2 to 5×10^{-7} amp/Mw.

Pulse shape was also studied both with ion chambers connected directly into the recording galvanometers and with a Keithley micromicroammeter as a preamplifier. No variation in response was determined between these two systems for the less sensitive ranges of the micromicroammeter.

The results of forty-five different step reactivity insertion experiments performed by General Atomic on its TRIGA Mark F reactor are given in Figs. 28 through 33. These transients were all performed under the same

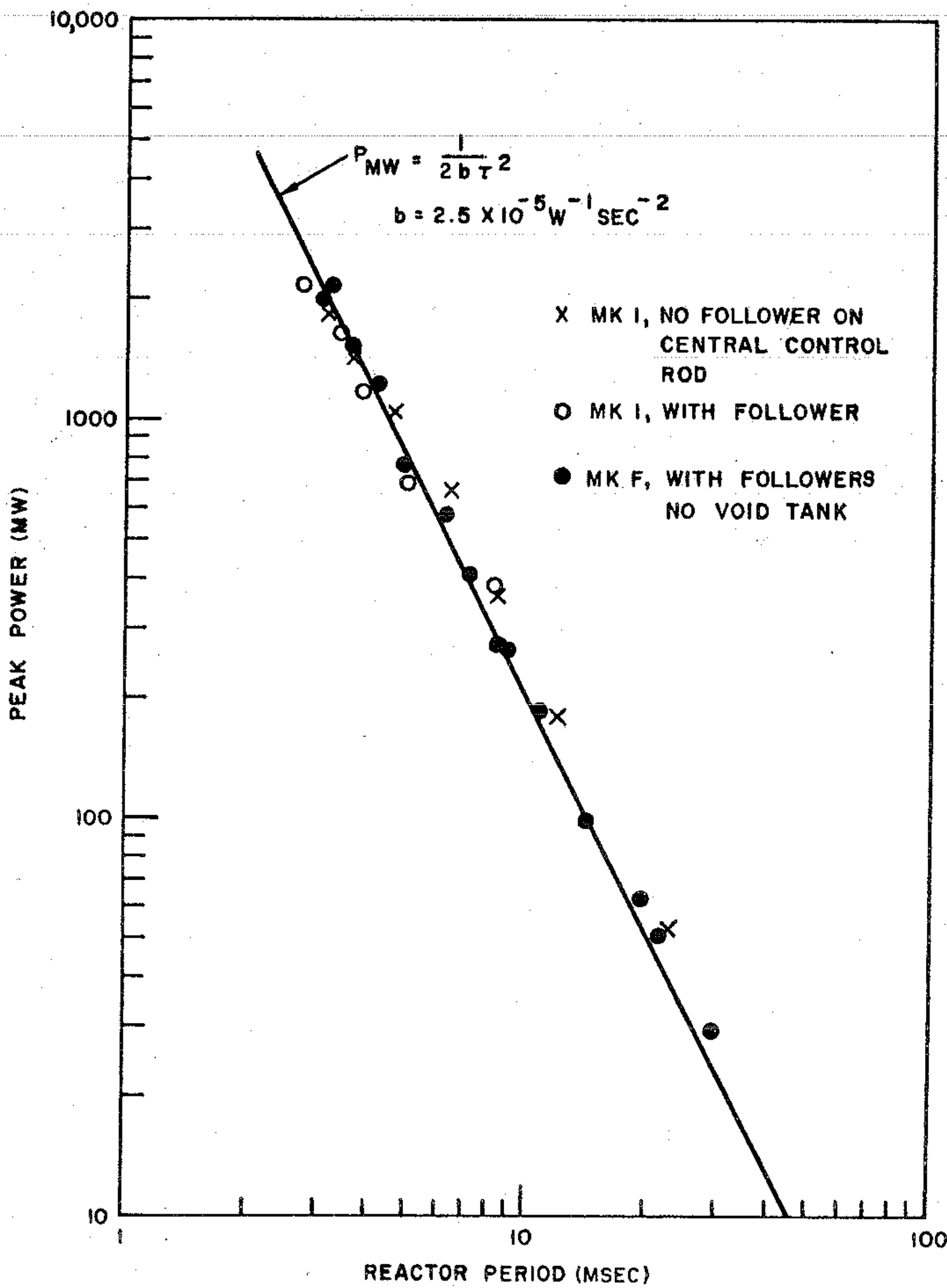


Fig. 28--TRIGA Mark F peak power transient

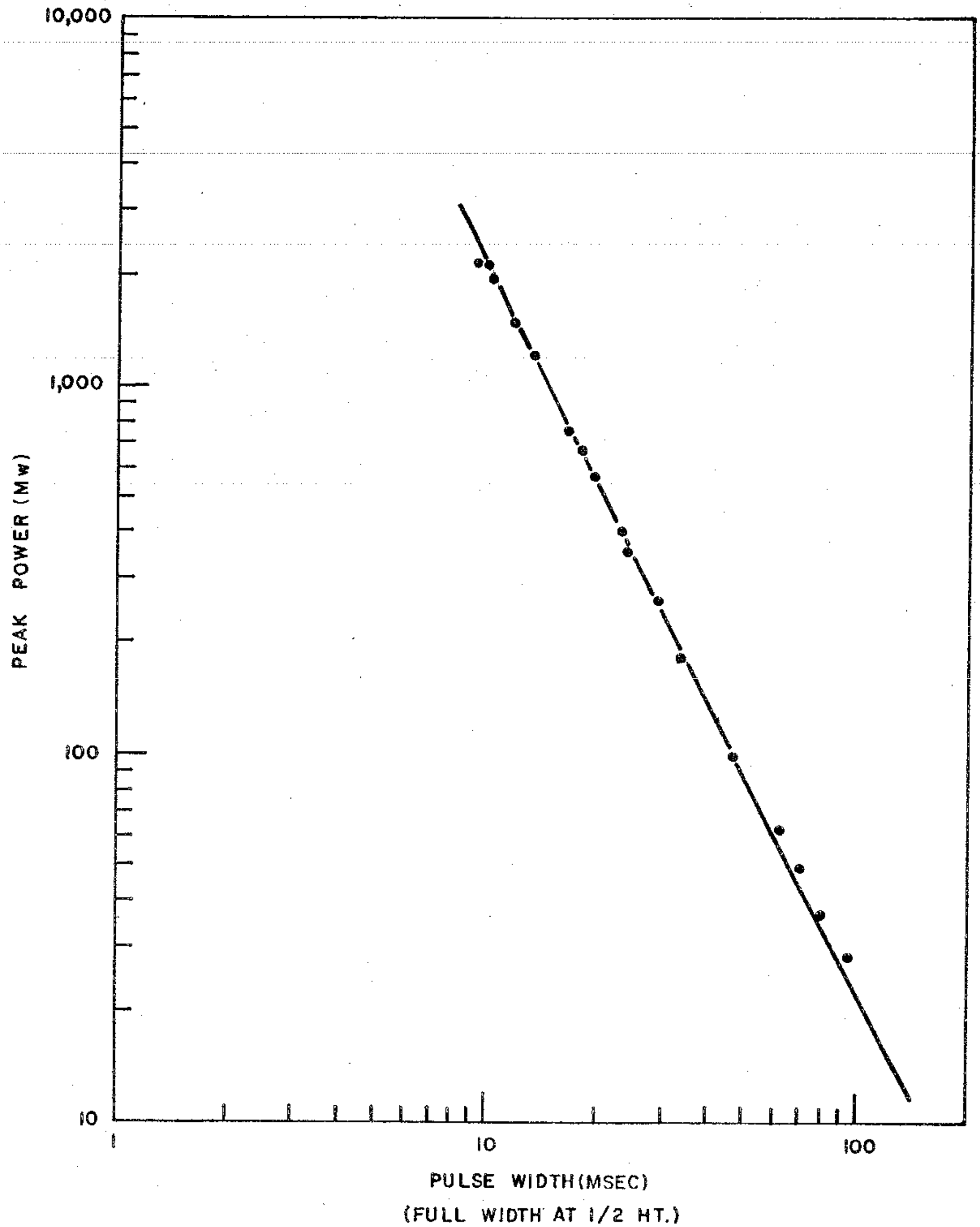


Fig. 29--TRIGA Mark F peak power pulse width

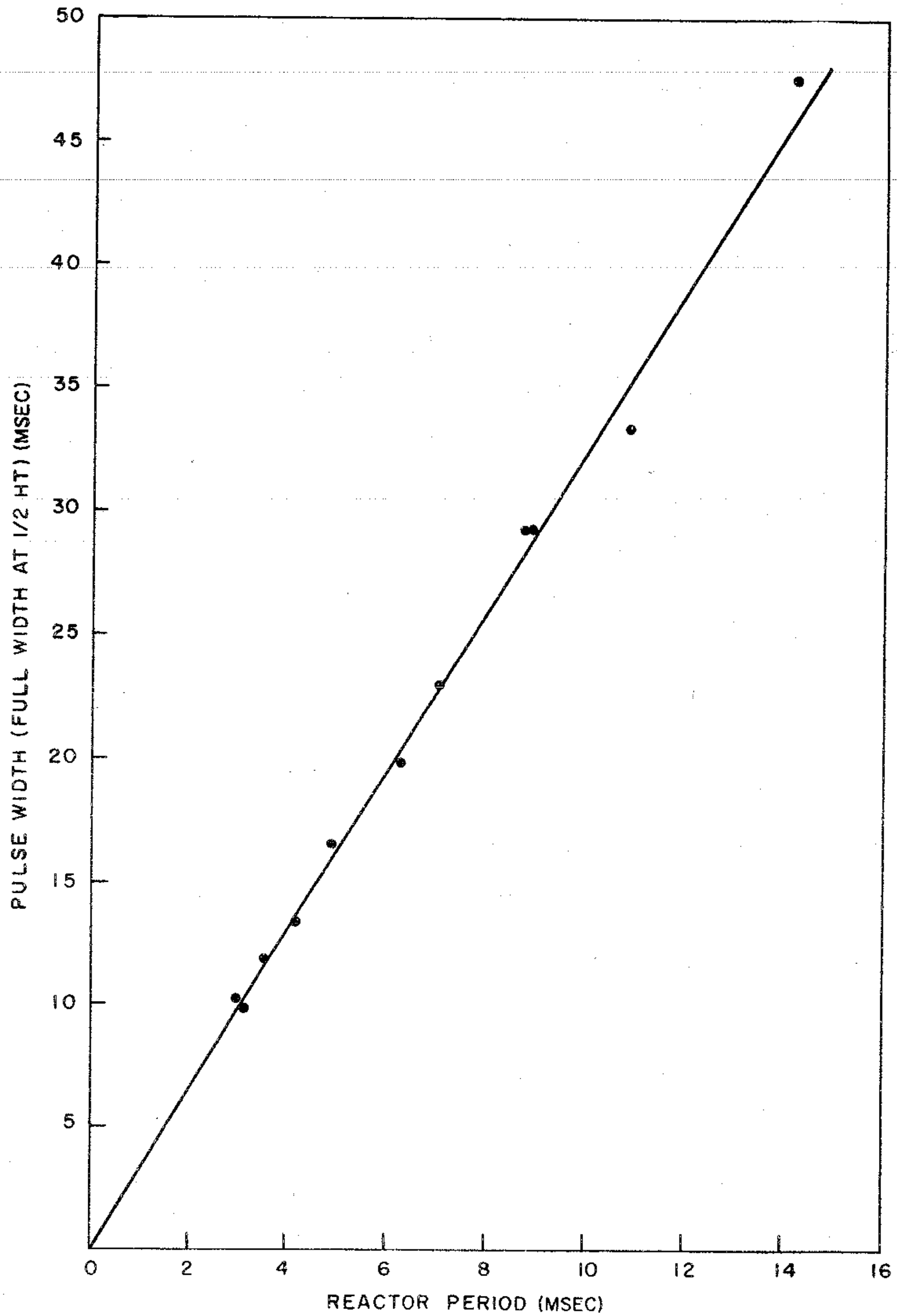


Fig. 30--TRIGA Mark F period—pulse relation

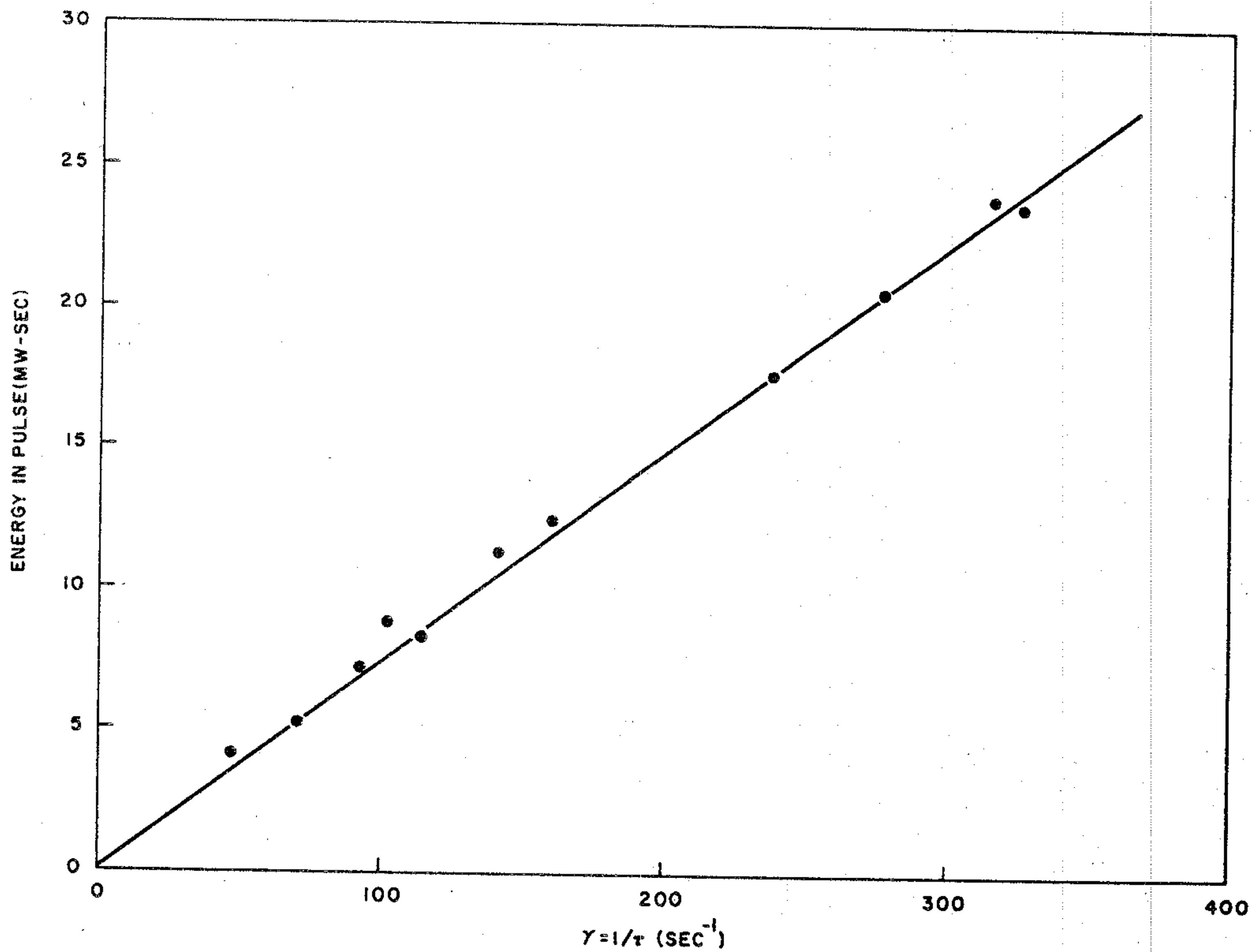


Fig. 31--TRIGA Mark F energy in prompt burst (inverse period)

conditions and with the void tank 18 in. from the edge of the core. Each point plotted on the curves represents an average of several power-period and pulse-width determinations.

The peak transient power plotted as a function of reactor period is shown in Fig. 28 for the TRIGA Mark F reactor and the Torrey Pines TRIGA Mark I reactor. This figure indicates that the Mark F reactor gives a curve essentially identical to that of the time-proven TRIGA Mark I reactors.

The peak power points plotted in Fig. 28 were plotted in Fig. 29 as a function of pulse width. This plot results in a smoother curve because the pulse width is easier to measure accurately. In Fig. 30, pulse width is plotted as a function of reactor period, which shows their linear relationship. Figures 31 and 32 indicate the integrated energy and prompt bursts as a function of inverse period and inverse pulse width for the TRIGA Mark F.

The $1/\tau^2$ behavior, where τ is the prompt period ($\tau = 1/\gamma$), of the peak power demonstrates remarkably good agreement with the shutdown characteristics of the Fuchs model. From the peak power curve, the energy shutdown coefficient, b, is 2.5×10^{-5} watt^{-1} sec^{-2}. From the prompt-burst energy versus $1/\tau$ curve, the value of b is 2.7×10^{-5} watt^{-1} sec^{-2}. Thus, the average value is $2.6 \pm 0.2 \times 10^{-5}$ watt^{-1} sec^{-2}, compared with a value of $2.5 \pm 0.2 \times 10^{-5}$ watt^{-1} sec^{-2} for the TRIGA Mark I. This corresponds to a temperature coefficient of $1.0 \pm 0.1 \times 10^{-4}$ $\delta k/k/C^{o}$.

During transient operation of the TRIGA Mark F, the reactor shutdown mechanism is activated by the peak fuel-element temperature, which protects the fuel material from excessive temperatures. Therefore, the data plotted in Fig. 33 are of interest as they give the measured peak transient fuel temperature as a function of reactor period for a reactor of the TRIGA Mark F type. In order to have these experiments on a comparable basis, the reactor was manually shut down 1 sec after the transient rod was ejected which also tended to reduce the heating of the fuel element by energy liberated after the initial portion of the pulse.

Examination of the experimental inhour curve indicates that the larger transients of the TRIGA Mark F can be well fitted to a curve with $\ell/\beta = 6 \times 10^{-3}$ sec. Using a β value of 0.0073, which has recently been calculated as the correct value for the Mark F reactor, ℓ is approximately equal to 44 μsec. The corresponding number for the TRIGA Mark I is approximately 60 μsec. Experiments to determine the effects of the void tank and the air-filled thimble on the transient characteristics of the reactor have been performed. Examination of the peak power and pulse shape during 25 transients indicated no measurable change in the shutdown coefficient. The shape of pulses remained identical.

Chapter 6

HAZARDS ANALYSIS

The DORF reactor will have the inherent safety characteristics of the TRIGA reactor because it uses the same fuel-moderator elements. The reactor will therefore be able to withstand large step reactivity insertions up to a limit set by the total heat capacity of the reactor core.

This hazards analysis assumes that administrative control of reactor operations, as set forth in Chapter 7, will ensure that the following conditions are met at all times:

1. The excess reactivity available in the reactor will be limited to 2.9%, $\delta k/k$, measured in the pool location (Position 3).
2. The reactor operations will be monitored by individuals trained in the detection and evaluation of radiological hazards.

In addition to an analysis of argon activation during normal operation in the exposure room, the following hazards are reviewed:

1. Improper fuel-loading procedures and variations in excess reactivity.
2. Loss of shielding water.
3. Production of radioactive gases in those areas of the reactor that contain air.
4. Reactor power transients involving rapid insertion of excess reactivity.

This chapter contains information on the effects on the reactor of various highly unlikely sequences of events which could permit the step insertion of large amounts of reactivity. This information is based on a preliminary evaluation of the experimental results obtained in the research and development program of the General Atomic TRIGA Mark F reactor.

IMPROPER FUEL LOADING

The initial loading of the reactor will be approximately 2.9% $\delta k/k$ excess reactivity when the core is in the most reactive position in the pool. This core loading should use 85 of the 87 fuel positions available in the grid plate. Should the grid plate inadvertently be fully loaded with fuel, the maximum excess reactivity available would be about 3.3% $\delta k/k$.

The control-rod worth for the safety and regulating rods will be approximately 1.5% $\delta k/k$ each, and the pneumatic, electromechanical transient rod will have a maximum of 2.2% $\delta k/k$ available for insertion in the pulsing mode. This maximum is determined by the preset maximum stroke of the piston. An additional 0.51% $\delta k/k$ will be available in the motor-driven portion of the rod, which will make a total of 2.7% $\delta k/k$ available in the transient rod. This would make it possible to operate the reactor at 1 Mw steady-state by the programmed pneumatic insertion of approximately 0.66% $\delta k/k$, and the insertion of the remainder by the motor-driven ball-nut lead screw.

During the normal maximum transient operation (2.2% $\delta k/k$ with a 1-sec automatic shutdown), the maximum fuel temperature will be less than 530°C. These maximum operating temperatures are maintained by a scram circuit by at least two thermocouples that are attached to fuel elements in the central region of the core. The various reactivity figures presented above (developed from TRIGA Mark F test results) have been used in the analysis of irregular operating conditions of the reactor control rods. The maximum step insertion of reactivity that is conceivable would occur in connection with the transient rod under conditions that would cause complete mechanical failure of the transient rod and the resulting rapid insertion of the entire 2.7% $\delta k/k$ available. Should these unlikely conditions occur, the reactor would rise on a 2.2-ms period and reach a peak fuel temperature between 650° and 700°C, assuming that the temperature scram mechanism failed simultaneously and allowed complete afterheating of the fuel elements by delayed neutrons. These conditions would possibly produce some slight bending of the central fuel elements in the core; however, the hydrogen pressure at these temperatures would be negligible. It is possible that some of the fuel elements in the core might have to be replaced because of the bending. This sequence of events should cause no other damage to the reactor and should not jeopardize the health or safety of the operating personnel or the public. This postulated sequence of events is felt to be the maximum conceivable in connection with the transient rod and would occur only in case of complete mechanical failure of the threaded and pinned connection between the shock absorber and the pneumatic piston.

As indicated previously, the reactor will have a maximum available excess reactivity in its most reactive position of 2.9% $\delta k/k$ and the transient rod will have a reactivity of 2.7% $\delta k/k$. Thus, if the three safety and regulating rods, which have a cumulative reactivity of approximately 4.5% $\delta k/k$, were withdrawn while the transient rod remained in the reactor, the reactor would be critical with a compensated excess reactivity of about 0.2% $\delta k/k$ at a power level of approximately 50 kw and a maximum fuel temperature of less than 100°C. If, under these conditions, a 2.2% $\delta k/k$ transient inadvertently occurred, the resulting total reactivity of 2.6% $\delta k/k$ would have a less hazardous effect than the previous sequence of events that resulted in the 2.7% $\delta k/k$ step insertion. The inadvertent type of transient would also be less severe than the rapid insertion of 2.6% $\delta k/k$ from a low-power level because of the initial steady-state heating of the fuel elements.

Therefore, it can be concluded that no sequence of events relating to the reactor control system can have a more adverse effect than the first case presented, which indicated no possibility of endangering the health or safety of the operating crew or the public.

Variations in reactivity of only 1% $\delta k/k$, which is no hazard, can also arise from the presence, the insertion, or the withdrawal of experiments, either in the pool or in the exposure room. The insertion of fissile materials will be controlled by the Reactor Safety and Planning Committee.

It is concluded that no hazardous variations in excess reactivity can result from moving the reactor core or from the presence of experiments adjoining the reactor.

LOSS OF SHIELDING WATER

Even though the possibility of the loss of shielding water is remote, a calculation has been performed to evaluate the radiological hazard associated with this type of accident. Assuming that the reactor has been operating for a long period of time at 100 kw prior to losing all of its shielding water, Table 6.1 gives the radiation dose rate at two different locations. The first location is the top of the core tank, 16 ft above the unshielded reactor core. The second is also at the top of the reactor, but shielded from direct radiation and subjected only to the scattered radiation from a thick concrete ceiling 9 ft above the top of the reactor tank. The assumption that there is a thick concrete ceiling maximizes the reflected radiation dose. Normal roof structures would give considerably less back-scattering. Time is measured from the conclusion of a period of 100-kw operation.

Table 6.1 shows that if an individual does not expose himself to the core directly he could work for approximately 1 hr at the top of the shield tank one day after shutdown without being exposed to radiation in excess of the radiation dose permitted by U.S. Atomic Energy Commission Regulations. This would permit sufficient time to view the interior of the shield tank with a mirror and to make emergency repairs.

Table 6.1

Time	Direct Radiation (r/hr)	Scattered Radiation (r/hr)
10 sec	1.1×10^4	6.1
1 day	5.7×10^2	0.32
1 week	3.7×10^2	0.21
1 month	1.8×10^2	0.09

The radiation from the unshielded core would be highly collimated by the shield structure and therefore would not be a public hazard.

The afterheat in this reactor following a water-loss accident is such that system temperatures will be far below those required to melt the aluminum fuel-element cladding. Therefore, no dispersal of fission products will take place.

The following summarizes conditions should a water-loss accident occur:

1. The probability of the loss of all of the shielding water is extremely small.
2. The concrete reactor shield provides adequate protection for nearby personnel at the time of a hypothetical loss of all shielding water.
3. There is no dispersal of fission products as a direct result of the loss of shielding water.
4. The reactor shielding is sufficient to allow emergency repair of the leak one day following shutdown, even after prolonged operation at a power of 100 kw.
5. There is negligible public hazard associated with the loss of shielding water.

PRODUCTION OF RADIOACTIVE GASES BY THE REACTOR

Preliminary calculations indicate that argon concentrations in the reactor vessel are within the U.S. Atomic Energy Commissions's quoted maximum permissible concentration (MPC) for the operating conditions being proposed. Argon concentrations in excess of the quoted MPC have never occurred in the reactor room during extensive operation of the TRIGA reactor at General Atomic.

Significant quantities of 7-sec N^{16} are produced in the water of the reactor core. However, the transport time from the reactor core to the surface of the shielding water has been measured as 42 sec when the reactor is operating at 100 kw. This corresponds to six half lives of N^{16} and provides a large attenuation factor. Experiments on the General Atomic TRIGA reactor show the following:

1. Negligible quantities of N^{16} are transferred from the reactor shielding water to the air in the reactor building.
2. N^{16} makes small contribution to the residual radiation flux at the top of the shielding water during operation.

ARGON ACTIVATION IN THE EXPOSURE ROOM

In the exposure room, a volume of air that is relatively large for research reactors is exposed to an appreciable thermal-neutron flux. Thus, the production of radioactive A^{41} by neutron capture in the natural A^{40} in air must be evaluated. Argon-41 has a half life of 1.8 hr and hence may be released to the atmosphere if certain precautions are taken to ensure that adequate dilution is provided to hold the concentration to permissible levels. This is usually accomplished by releasing the A^{41} through a stack at a sufficient height above the ground. Since A^{41} is an inert gas and has a short half life, it presents no health hazard through retention by the body, but only through direct exposure to its gamma radiation. The MPC recommended by the Atomic Energy Commission regulations is based on continuous submersion in an infinite atmosphere of air containing A^{41}. If release through a stack ensures that the maximum average concentration at <u>any</u> point downwind from the stack is less than the MPC, then there is an additional factor of safety because a stack effluent plume would not, on the average, be considered infinite in size.

It is shown in Appendix I that under normal operation (1) the employees who work in the exposure room will receive doses far lower than the occupational tolerances, even if they enter the room immediately after reactor shutdown, and (2) the average maximum concentration of A^{41} reaching the ground from the stack is less than the nonoccupational tolerance by over a factor of two. Therefore, the conclusion is that at a stack height of 45 ft, A^{41} from the DORF research reactor presents no undue hazard to the health and safety of the public or the employees on the reactor site.

REACTOR POWER TRANSIENTS

An extensive experimental program has been conducted with the TRIGA Mark F prototype to measure the transient behavior of the reactor. This has consisted of a series of step reactivity insertions supplemented by kinetics calculations.

From the data obtained from experiments with the Mark F (Chapter 5), it is concluded that repeated step reactivity insertions of 2.2% will not adversely affect the fuel material and that such insertions may be routinely conducted. The maximum transient temperature of 500°C during these steps does not cause a phase transition of the fuel material.

Transients in the 2.4% to 3.1% reactivity region have been demonstrated to cause no hazard to operating personnel. Nevertheless, routine operation in this range of reactivity insertions is not recommended because some bending of the central elements has been observed after such steps, as a result of a metallurgical phase transition of the fuel material. Elements bend away from the center of the reactor, resulting in a slight decrease in reactivity.

Exposure to radiation in the immediate vicinity of the top of the reactor tank is small--less than 5 mrem for the largest transients conducted at General Atomic.

Calculations have shown that certain materials, such as enriched uranium, plutonium, B^{10}, or Li^6, if present in the reactor in bulk, may partially melt and vaporize when exposed to the high-intensity neutron bursts prevailing during pulses. Pertinent administrative procedures will be established to prevent the presence of these materials whenever the reactor is being pulsed.

The maximum size of a transient can be effectively controlled by limiting the worth of the transient rod. For the DORF reactor, the transient portion of the rod will be worth not more than 2.2% $\delta k/k$. This limits the maximum transient to this value, regardless of the amount of excess reactivity available in the reactor.

MAXIMUM CREDIBLE ACCIDENT

The maximum credible accident which can occur with this reactor is a defect in the cladding of a fuel element either prior to or simultaneously with a 2.2% $\delta k/k$ transient--the largest amount of reactivity controlled by any single rod. From operating experience with the TRIGA prototype, it is concluded that this maximum credible accident causes no significant hazard to individuals near the reactor.

The 2.2% $\delta k/k$ reactor transient itself will cause no hazard. In the unlikely event that a defect exists in the cladding of a fuel element at the time of pulsed operation, there will be some release of fission products to the atmosphere of the reactor room. Operating data have been obtained from the General Atomic TRIGA reactor to indicate the magnitude of such release.

During a higher transient than that mentioned above (3.1% $\delta k/k$), a cladding failure resulted in the release of a minor quantity of fission products. The following is a summary of the measurements made:

1. The activity in the cooling-water tank reached a maximum of 0.2 $\mu c/cm^3$. It decayed very rapidly and was 5×10^{-5} $\mu c/cm^3$ 24 hr after the cladding failure.
2. The activity in the air of the reactor room reached about ten times the MPC for fission products. It decayed very rapidly and experiments were resumed 2 hr after the activity release. The maximum dose to operating personnel was less than 1 mr.
3. The noble gases were not collected on the filter samples used,

but it may be inferred from the decay of the daughter products observed that only the noble-gas fission products escape from the TRIGA pool in significant proportions when a fuel-element cladding fails.

It is concluded that a cladding failure, or even the failure of the cladding of several fuel elements, would not constitute an undue hazard to the operating crew or the general public.

Chapter 7

ORGANIZATION

A well-defined system of administrative control is required for the efficient operation of a reactor facility. Although the TRIGA reactor is of itself inherently devoid of undesirable operating characteristics and the structural and safety interlock features of the facility normally ensure the necessary radiation protection, administrative control is also necessary to further guarantee against exposure of personnel to high-intensity fields and to possible ingestion hazards. This involves establishment and execution of operational procedures which bring to full utilization the built-in protective features of the facility and prescribes procedures for initiating modifications to the facility, performance of experiments, and the handling of derangements which might occur as the result of this work. Experienced and responsible individuals who have the necessary background and knowledge will administer such controls.

The responsibility for administrative control will be vested through normal DOFL channels to the following personnel:

Chief of Nuclear Vulnerability Branch
Reactor Safety and Planning Committee
Reactor Administrator
Reactor Supervisor
Reactor Operator

1. Chief of Nuclear Vulnerability Branch

The Branch Chief is ultimately responsible for the safe, competent, and efficient operation and use of the reactor facility. This responsibility manifests itself in

a. The selection of responsible and competent personnel as members of the Reactor Safety and Planning Committee, the Reactor

Administrator, and Reactor Supervisor and the approval of appointments of all other reactor operating personnel.

b. The establishment of administrative controls consistent with all applicable Governmental regulations and licenses and any other pertinent (state, military, or local) regulations.

c. The enforcement of all such regulations.

2. The Reactor Safety and Planning Committee

The Reactor Safety and Planning Committee is directly responsible for reviewing all experimental programs to ensure that such programs are carried out in a safe and competent manner, with particular control upon new or untried operations or procedures. To this end the committee will

a. Review and approve all experiment plans prior to execution.

b. Review and approve all changes in existing experiment plans which involve any changes in the reactor core, core loading, control rods, exposure room, or shielding components.

c. Ensure that all activities and experiments shall conform with all applicable Governmental regulations and licenses with regard to operation of the reactor facility, handling of special nuclear material, and possession and handling of activated by-product materials.

3. Reactor Administrator

The Reactor Administrator is directly responsible for promulgation and enforcing administrative rules and operating procedures to ensure safe and competent operation, use, and maintenance of the reactor facility. He is also responsible for the selection of reactor facility operating personnel subject to the Branch Chief's approval. He is responsible for seeing that there is adequate personnel and experiment dosimetry. To this end he will

a. Review, approve, and promulgate administrative procedures and changes thereto.

b. Review and approve changes in the function or design of the electrical or mechanical components of the reactor or facility.

c. Promulgate safe operating procedures of the reactor and facility.

d. Promulgate emergency procedures.

e. Review and approve in writing all changes in the loading of the reactor core, unusual uses of the reactor, and any changes in the normal operating procedure of the reactor.

f. Consult with the manufacturer concerning any change in normal operating procedure or proposed modification to the nuclear, electrical, or mechanical components of the reactor.

4. The Reactor Supervisor

The Reactor Supervisor is directly responsible for

a. Scheduling activities at the reactor facility and keeping adequate records of all experiments and operations.

b. Enforcing operating procedures and ensuring that the facility is operated in a safe and competent manner at all times.

c. Maintaining adequate personnel and experiment dosimetry and other required radiation-monitoring equipment at the reactor facility.

d. Adequate maintenance of the facility.

e. Enforcing administrative rules and operating procedures at the reactor facility.

f. Ensuring that all conditions of applicable Governmental licenses and regulations are fulfilled.

g. Executing emergency procedures.

5. The Reactor Operator

The Reactor Operator is responsible for

a. The safe and proper operation of the reactor; he shall be in a position to operate the controls at all times that the reactor is in operation.

b. Filling out all log sheets and following all checkout procedures.

c. Informing the supervisor of all unusual or unexpected incidents, any apparent or real operational error, any real or apparent equipment or instrument failure, or any malfunction.

d. Ordering emergency procedures and calling the supervisor in the event a nuclear incident should occur or any unsafe condition should be present.

Health Physics will be administered by the Radiation Safety Officer of the Walter Reed Medical Center, by agreement with its Commanding General. A chart of the DORF reactor organization is given in Fig. 34.

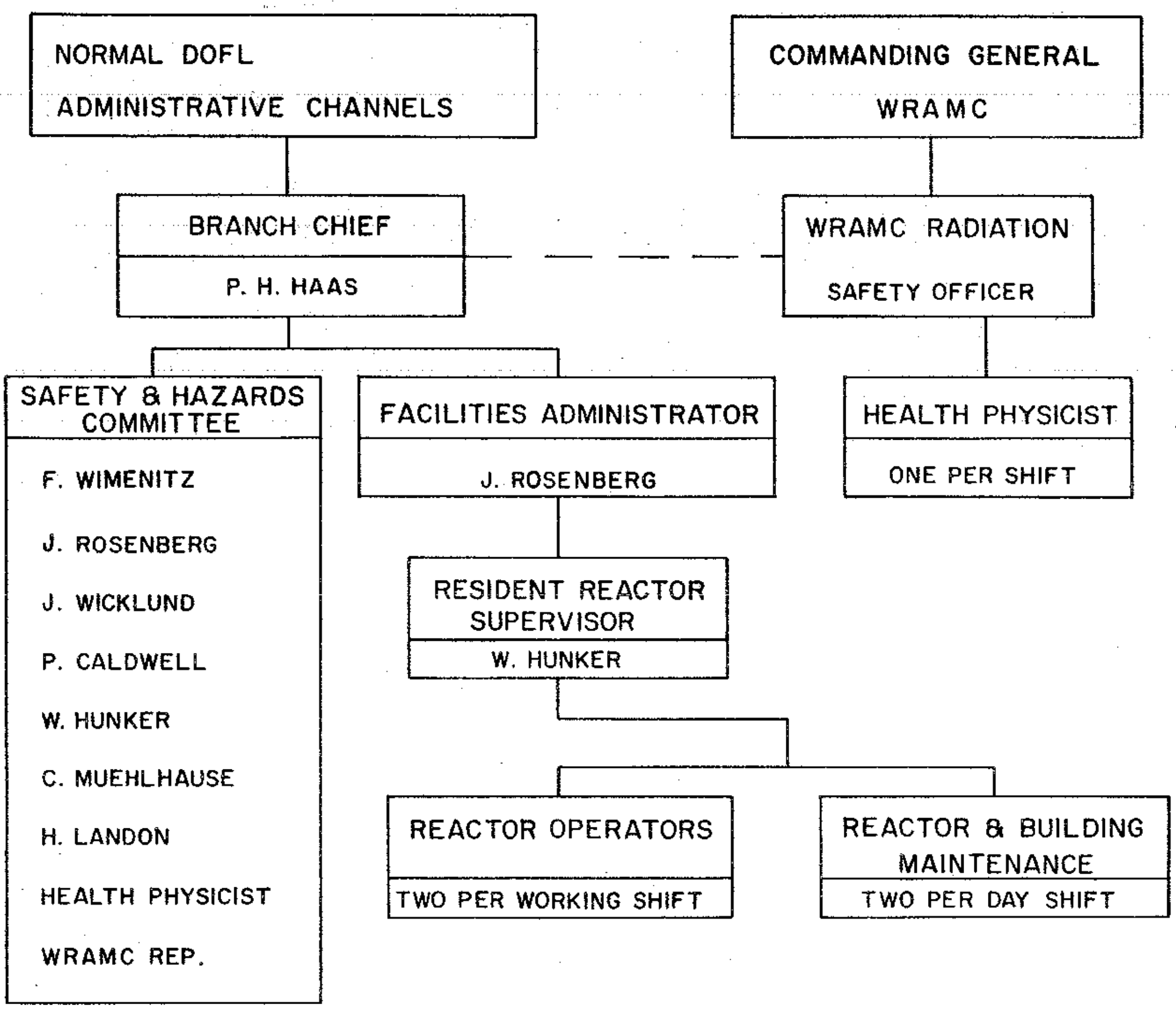


Fig. 34--Organization chart of DORF

Appendix I

ARGON ACTIVATION

ARGON ACTIVATION IN EXPOSURE ROOM AIR

Activation of argon-40 in the air of the exposure room results from the thermal neutron flux from the reactor. Calculations used here indicate the level and effects of this activity based on atmospheric air pressure, a temperature of 70°F, and an A^{40} content of 0.94% in air.

As the reactor core "sees" the room from one of the walls, the room was approximated by a hemispherical volume equal to the room volume, i.e., the neutron source was assumed to be located at the center of a sphere having a volume twice that of the room volume. The radius of this sphere is designated R_0. The neutron flux, as a function of distance from the source, is given by

$$\phi n(r) = \frac{S_0}{4\pi r^2} \quad \text{neutrons/cm}^2\text{-sec,} \tag{I-1}$$

where S_0 is the neutron source strength in neutrons/sec.

The average neutron flux within the hemispherical volume is obtained by dividing the volume integral of $\phi n(r)$ by the total volume, i.e.,

$$\overline{\phi n} = \frac{\int_V \phi n(r)\, dV}{\int_V dV} = \frac{3\, S_0}{4\pi\, R_0^2} . \tag{I-2}$$

The production rate of A^{41} as a function of production time is given by

$$\frac{dN}{dt} = \overline{\phi n}\, \Sigma_a - \lambda N, \tag{I-3}$$

where Σ_a = macroscopic absorption cross section of A^{40}, cm^{-1},

λ = decay constant for A^{41}, sec^{-1},

N = atomic density of A^{41}, atoms of A^{41} per cm^3.

Integrating from $N = 0$ at $t = 0$ to $N = N_1$ at $t = t_1$ yields

$$N_1 = \frac{\overline{\phi}n\ \Sigma_a}{\lambda}\left(1 - e^{-\lambda t_1}\right) \quad \text{atoms } A^{41}/\text{cm}^3 \tag{I-4}$$

and the corresponding activity of A^{41} is

$$A_1 = \overline{\phi}n\ \Sigma_a\left(1 - e^{-\lambda t_1}\right) \quad \text{disintegrations/cm}^3\text{-sec.} \tag{I-5}$$

If the room is ventilated at the rate of Q cubic feet per second and the room volume is V cubic feet, then the differential equation must be modified so that

$$\frac{dN}{dt} = \overline{\phi}n\ \Sigma_a - \left(\lambda + \frac{Q}{V}\right) N \tag{I-6}$$

and the corresponding solution in terms of A^{41} activity is

$$A_1 = \frac{\overline{\phi}n\ \Sigma_a\ \lambda}{\lambda + Q/V}\left\{1 - \exp\left[-\left(\lambda + \frac{Q}{V}\right)t_1\right]\right\}. \tag{I-7}$$

Steady-state Operation

If the reactor operates at a steady-state power level for a long time (i.e., $t_1 >> 1/\lambda$ or $1/(\lambda + Q/V)$), then the production and removal rates of A^{41} will be in equilibrium and

$$A_1 = \overline{\phi}n\ \Sigma_a \quad \text{disintegrations/cm}^3\text{-sec} \tag{I-8}$$

for the room without ventilation and

$$A_1 = \frac{\overline{\phi}n\ \Sigma_a\ \lambda}{\lambda + Q/V} \tag{I-9}$$

with ventilation.

Pulsed Reactor Operation

In power bursts lasting a few milliseconds, the numerical value of λt_1 or $(\lambda + Q/V)t_1$ is small, and hence the approximation may be made

$$N_1 = \overline{\phi}nt_1 \, \Sigma_a \qquad \text{atoms/cm}^3 \tag{I-10}$$

for both the ventilated and nonventilated cases. The corresponding activity is then

$$A_1 = \overline{\phi}nt_1 \, \Sigma_a \, \lambda \qquad \text{disintegrations/cm}^3\text{-sec.} \tag{I-11}$$

The term $\overline{\phi}nt_1$ is the time-integrated thermal neutron flux over the duration of the pulse and will be defined by the expression

$$\overline{\phi}n' = \overline{\phi}nt_1 = \frac{3\,S_0 t_1}{4\pi\,R_0^2} = \frac{3\,S_0'}{4\pi\,R_0^2}\,, \tag{I-12}$$

where S_0' is the total number of thermal neutrons escaping from the reactor face in a pulse. Thus,

$$A_1 = \overline{\phi}n' \, \Sigma_a \, \lambda \qquad \text{disintegrations/cm}^3\text{-sec .} \tag{I-13}$$

Exposure to A^{41}

The dose received from A^{41} in the exposure room by an individual who enters the room immediately after reactor shutdown may be evaluated by assuming the room to be a sphere with the individual at the sphere's center during the whole exposure time. Since 0.991 disintegrations of A^{41} result in a 1.3-Mev gamma photon, the volume source of gamma rays at the time of shutdown is

$$S_\gamma = 0.991 A_1 \qquad \text{photons/cm}^3\text{-sec,} \tag{I-14}$$

where A_1 is as previously defined and represents the A^{41} activity at shutdown.

The photon flux at the center of the sphere is given by

$$\phi_\gamma = \int_0^{R_1} \frac{0.991 A_1}{4\pi r^2} \, e^{-\mu r} \times 4\pi r^2 \, dr \qquad \text{photons/cm}^2\text{-sec ,} \tag{I-15}$$

where μ is the absorption coefficient for 1.3-Mev gammas in air and R_1

Table I. 1

20-MIN-EXPOSURE DOSES
(in mrep)

	No Ventilation	With Ventilation
100-kw steady state	12. 3	0. 026
24 Mw-sec pulse	0. 314	0. 042

ARGON ACTIVATION IN REACTOR WATER

The argon activity in the reactor pool water results from the argon dissolved in water. To evaluate the activation of argon in the reactor pool, the following reasoning and assumptions were used.

The amount of argon which is dissolved in water is calculated, assuming that argon follows Henry's law. If the water temperature is taken to be $70^{\circ}F$, then the corresponding water vapor pressure is 26 mm Hg. The partial pressure of air is then 760 - 26 = 734 mm Hg. The argon content of air is 0. 94%, by volume, and hence the partial pressure of argon is $734 \times (9.4 \times 10^{-3}) = 7$ mm Hg.

The saturated concentration of argon in water, according to Henry's law, is

$$X = \frac{P}{K} , \qquad \text{(I-25)}$$

where X = mole fraction of argon in water,

P = partial pressure of argon above water,

K = Henry's constant = 2.84×10^{7} at $70^{\circ}F$.

Thus, $X = 2.46 \times 10^{-7}$ mole A^{40} per mole of $(H_2O + A^{40})$ or $X = 1.367 \times 10^{-8}$ mole A^{40} per 1 cm^3 H_2O.

Argon-41 production, assuming saturated conditions and irradiation time, t, is given by

$$N_1 = \frac{\bar{\phi}_n \Sigma_a (1 - e^{-\lambda t})}{\lambda} . \qquad \text{(I-26)}$$

A corresponding activity for small values of λt is

$$A_1 = \overline{\phi}\lambda t \Sigma_a \, . \tag{I-27}$$

The average thermal flux in the reactor core is estimated to be 1.0×10^{12} neutrons/cm^2-sec. The circulation of water in the core occurs through natural convection and it is estimated that it changes completely in 4 sec. Since the core holds 3.5×10^4 cm^3 of water, the rate of flow of water through the core is 0.9×10^4 cm^3/sec. Substituting appropriate values for Σ_a and taking irradiation time as 4 sec, there results

$$A_1 = 1.87 \quad \text{disintegrations/cm}^3\text{-sec.} \tag{I-28}$$

The total activity from the core is thus

$$Q_0 = 0.454 \ \mu c/\text{sec} \, . \tag{I-29}$$

The travel time of A^{41} from the core to the water surface, a distance of 16 ft, was measured as 42 sec. Applying the decay law, A^{41} activity reaching the surface is

$$Q = 0.450 \ \mu c/\text{sec} \, . \tag{I-30}$$

Under saturated, steady-state conditions, the rate at which A^{41} will escape from the water surface will correspond to 0.450 μc/sec, and an equivalent amount of A^{40} will dissolve in water in place of A^{41}. At increased water temperature, the partial pressure of water vapor will increase and the amount of dissolved A^{40} will decrease.

The radioactive argon escaping from the reactor pool will dissipate in the air of the reactor-room level. The volume of air at this level is estimated to be 71,600 ft^3; the air is recirculated through an air-conditioning system at a rate of 9,070 ft^3/min. Though the reactor-room level is partitioned into various offices and working areas, because of the high air-circulation rate, it may be assumed that A^{41} is evenly distributed throughout the entire level. The concentration of A^{41} in the air at the reactor-room level may be obtained by considering the material balance of A^{41} (in $\mu c/sec$).

Letting Q be the production rate, P the exhaust rate, and D the decay rate, the accumulation rate will be

$$A = Q - P - D \; . \qquad \text{(I-31)}$$

If the reactor is operated for a long time, equilibrium conditions in the air may be assumed. The accumulation rate, then, is equal to zero. Hence,

$$Q - D = P \; .$$

From Eq. (I-26), $Q = 0.450 \; \mu c/sec$.

The <u>average</u> decay rate may be expressed as

$$D = Q - \frac{1}{\theta} \int_0^\theta Q e^{-\lambda t} \, dt \qquad \mu c/sec \; , \qquad \text{(I-32)}$$

where θ is the time in which decay occurs and is the average time of residence of an atom of A^{41} at the reactor-room level before it is vented through the stack, i. e. ,

$$\theta = \frac{71,600 \; ft^3}{6.3 \; ft^3/sec} = 1.14 \times 10^4 \; sec \; . \qquad \text{(I-33)}$$

Integrating Eq. (I-32) and substituting in Eq. (I-31) yields

$$P = Q \, \frac{1 - e^{-\lambda\theta}}{\lambda\theta} \qquad \mu c/sec \; . \qquad \text{(I-34)}$$

Substituting for λ and θ yields

$$P = 0.261 \; \mu c/sec \; . \qquad \text{(I-35)}$$

Since equilibrium conditions are assumed, the same concentration of A^{41} will exist in the air of the reactor-room level as in the exhausted air. Concentration in the exhausted air is $(0.261)/(6.3)(2.832 \times 10^4) = 1.46 \times 10^{-6} \; \mu c/cm^3$. This is less than the maximum permissible occupational tolerance, as defined by Part 20 of the Code of Federal Regulations.

ROUTINE RELEASE OF A^{41} TO THE ATMOSPHERE

An expression for the concentration of A^{41} in the exposure room was presented earlier. Although higher concentrations will prevail without ventilation, contaminated air under these conditions will not be discharged to the atmosphere. As soon as ventilation starts, the concentration will drop to a few percent of the stagnant-air value. It is evident, therefore, that from the standpoint of discharged radioactivity, the most severe case is a long operation of the reactor at the 100-kw power level.

The radioactive argon activity discharged from the exposure room was calculated from Eq. (I-9) multiplied by the volume of air discharged per unit time with

$$S_0 = 2.56 \times 10^{14} \text{ neutrons/sec at 100 kw,}$$
$$\bar{\phi} = 4.98 \times 10^{8} \text{ neutrons/cm}^2\text{-sec,}$$
$$\Sigma_a = 1.25 \times 10^{-7} \text{ cm}^{-1},$$
$$Q = 21 \text{ ft}^3/\text{sec},$$
$$\lambda = 1.06 \times 10^{-4} \text{ sec}^{-1},$$
$$V = 3200 \text{ ft}^3,$$
$$R = 351 \text{ cm},$$

which gives the activity discharge rate as

$$q = 16\ \mu c/\text{sec}.$$

Contaminated air from the exposure rooms is discharged through a stack after passing through an absolute filter. A requirement of the regulations set forth by the U. S. Atomic Energy Commission is that the discharged radioactivity from the stack does not exceed the nonoccupational maximum permissible concentration in the atmosphere. It is extremely difficult to estimate the effect of meteorological variables and the varying terrain on the dilution of the stack effluent. One conservative method of doing this calculation is to use the procedure of the AEC's publication "Meteorology and Atomic Energy" for a continuous point source. The relation between source-strength elevation and concentration on the ground as a function of atmospheric conditions is given by Sutton's equations. These equations have been differentiated and minimized to obtain the maximum concentration on the ground downwind from the stack. The results as given in "Meteorology and Atomic Energy" for a continuous point source are

$$\chi_{max} = \frac{2q}{e\pi \bar{u} h^2} \quad \mu c/cm^3 , \tag{I-36}$$

$$d_{max} = \left(\frac{h^2}{C^2}\right)^{\frac{1}{2 - n}} \quad \text{meters} , \tag{I-37}$$

where χ_{max} = average maximum concentration of radioactivity in air,

d_{max} = distance downwind from stack where χ_{max} occurs,

q = radioactive source, $\mu c/sec$,

h = stack height, meters,

$\bar{u}$ = wind velocity, meters/sec,

e = base of natural logarithm,

n = Sutton's coefficient of stability,

C = Sutton's coefficient of diffusion.

Conservative average weather conditions are assumed--namely, moderate temperature inversion and 5 mph wind velocity. This corresponds to $n = 0.33$. Values of C are obtained from Fig. 9.4 of "Meteorology and Atomic Energy" as a function of source elevation, stability, and wind speed.

The maximum ground concentration and downwind distances from stack, where the maximum occurs, are shown in Table I.2 and are plotted in Fig. 35. These values are for A^{41} generation in the exposure room.

It is concluded that the average maximum ground concentration from the discharge from the 45-ft height above ground level is less than the

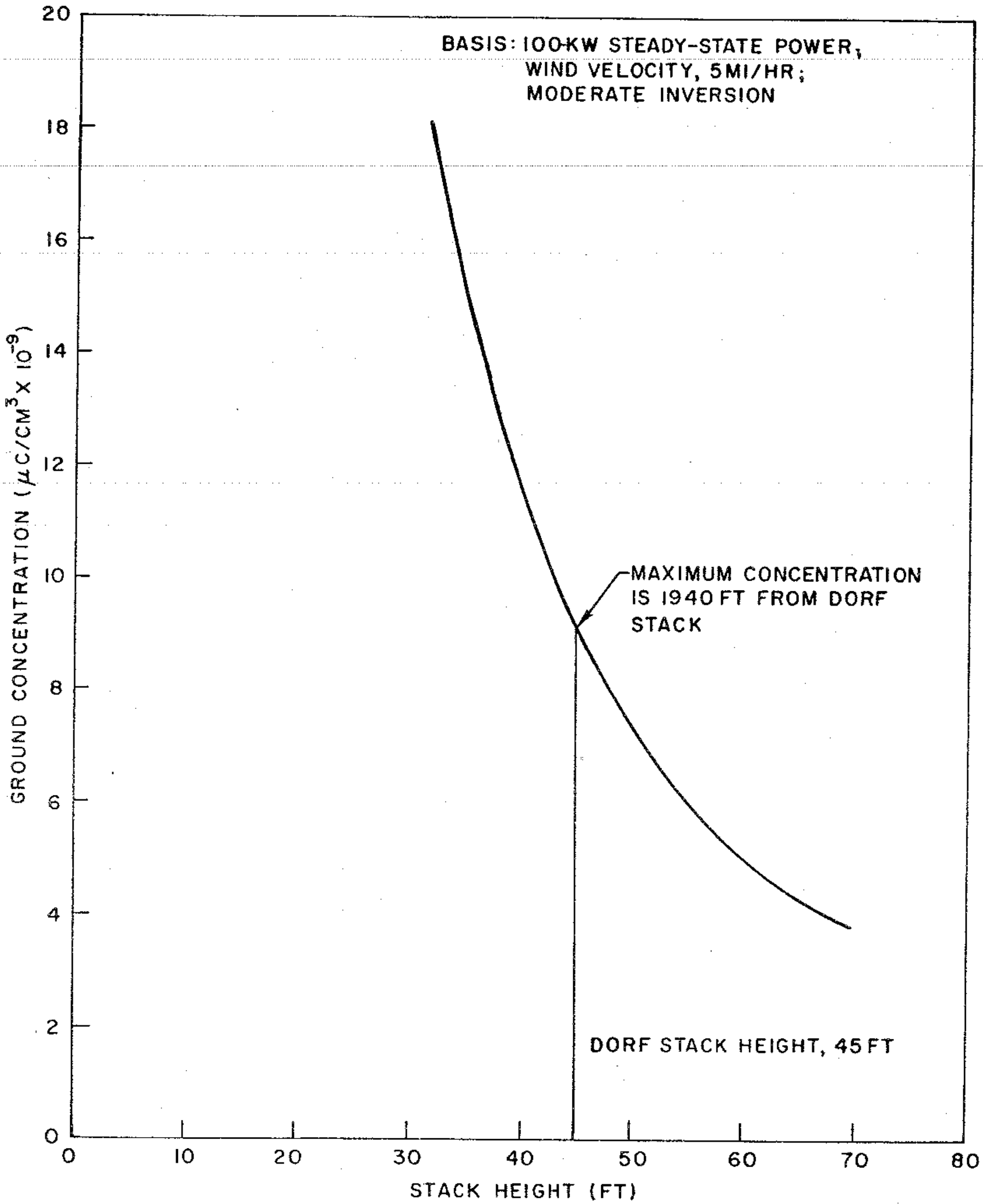


Fig. 35--TRIGA Mark F stack height versus A^{41} ground concentration

Table I-2

MAXIMUM GROUND CONCENTRATION AS A FUNCTION OF STACK HEIGHT

Stack Height		χ_{max}	d_{max}	
Meters	Feet	$\mu c/cm^3$	Meters	Feet
10	32.8	1.675×10^{-8}	390	1279
12	39.4	1.16×10^{-8}	490	1610
14	45.9	0.854×10^{-8}	610	2001
16	52.5	0.654×10^{-8}	720	2362
18	59.1	0.517×10^{-8}	850	2788

nonoccupational maximum permissible concentration (MPC) as given in the National Bureau of Standards Handbook No. 69, which gives 4×10^{-7} $\mu c/cm^3$ as the nonoccupational criterion for whole-body immersion in A^{41}.

Appendix II

SOIL ACTIVATION

An analysis of the soil taken from test drillings on the site was obtained through the U. S. Department of the Interior, Geological Report No. TDS-A526. From this analysis, the concentration of parent elements which yield radioactive isotopes was determined. In making the soil-activation determination it was assumed that

1. Only fast neutrons leak from the concrete shield and that all of those that enter the soil are eventually absorbed by the soil as thermal neutrons.
2. The facility produces 48 to 20 Mw-sec pulses per day each day for a full year.
3. The integrated soil activity over the year does not exceed the total disintegrations from 12 monthly burials made in compliance with 10 CFR 20.304.
4. The neutron activation results in a saturated activity of the soil.

PERMISSIBLE ACTIVATION

The activity restriction for continuous irradiation of isotopes in the soil was determined from the monthly burial levels as specified in 10 CFR 20.304. The following method was used to establish this restriction:

1. The permissible isotopic burial tolerances were obtained from Table C of 10 CFR 20 and were corrected by a factor of 1000, as specified--this tolerance is denoted as $C_i \times 10^3$ μc.
2. The number of active atoms of the isotope is related to the activity by

$$N_i = \frac{3.7C_i \times 10^7}{\lambda_s} \quad \text{(active atoms at each burial),}$$

where λ_s is the decay constant, in sec^{-1}.

3. The total number of disintegrations of isotope i in a year resulting from 12 successive monthly burials would be

$$q_i = N_i \sum_{n=1}^{n=12} (1 - e^{-n\lambda t}); \qquad \lambda = (month)^{-1} .$$

Expressing λ_s in terms of $\tau_{1/2}$, the isotopic half-life in months, and substituting for N_i, we have

$$q_i = 1.4 \times 10^{14} \times C_i \tau_{1/2} \left[12 - \left(\frac{1 - e^{-12\lambda t}}{e^{+\lambda t} - 1} \right) \right] ,$$

where t is equal to 1 month.

DETERMINATION OF MACROSCOPIC CROSS SECTIONS

The results of the soil analysis were used to determine the total soil absorption and isotopic activation cross section.

Activation Calculation

The radioactive isotopes in the soil are considered to be in equilibrium; therefore, the decay and creation rates are equal.

Considering the source of S neutrons per second to be at an equivalent point and isotropic in distribution, the following describes their probability of arrival at some distance from their source and the absorption in parent isotopes of the soil:

$$N(r)^* = \frac{\Sigma_i S e^{-\Sigma_s r} F(b)}{4\pi r^2} \quad \text{active isotopes at radius r, } cm^{-3},$$

where Σ_i = macroscopic neutron activation cross section of isotope i, cm^{-1},

Σ_s = macroscopic total neutron absorption cross section of soil, cm^{-1},

r = radius, cm,

F(b) = shield attenuation factor.

Describing the volume in which absorption occurs as

$$dV = r^2 \, d\Omega \, dr \;\; cm^3 ,$$

where $d\Omega$ is the neutron escape angle, in sterads, the integration over r from zero to infinity yields

$$N_t^* = F(b) \int_0^\infty \frac{\Sigma_i \, S e^{-\Sigma_s r} \, r^2 \, dr \, d\Omega}{4\pi r^2} = F(b) \, \frac{\Delta\Omega}{4\pi} \, \frac{S\Sigma_i}{\Sigma_s} ,$$

where N_t^* is the total active isotopes in $\Delta\Omega$ sterad resulting from activation of a single parent isotope, and F(b) is the shielding factor, introduced as an average value associated with the applicable geometry.

Since the source was introduced as neutrons per unit time, the value N_t^* is the creation of active isotopes per unit time. Also, since an equilibrium state has been specified, N_t^* also represents the constant decay rate.

Examination of the expression for N_t^* shows that

$\frac{\Sigma_i}{\Sigma_s}$ = absorptions and also disintegrations of isotope i per neutron absorbed in the soil per unit time.

The total neutron absorption cross section of the soil, summing over all elements, is

$$\Sigma_s = \sum \left(m\rho \, \frac{A_0}{A} \, \sigma_a \right) \; cm^{-1} ,$$

where m = weight fraction of element in soil,

ρ = density of the soil, g/cm^3,

A = atomic weight of the element, g,

A_0 = Avogadro's number = 0.602×10^{-24},

σ_a = microscopic absorption cross section, cm^2.

The neutron activation cross section of an isotope is

$$\Sigma_i = am\rho \, \frac{A_0}{A} \, \sigma_i ,$$

where a is the abundance of the parent isotope in the element, and σ_i is the microscopic activation cross section, cm^2.

Determination of Activation Criterion

The reactor leakage rate has been determined to be approximately $ST = 2 \times 10^{21}$ fast neutrons per year (based on the previously assumed duty cycle), where T is seconds per year.

The leakage angle $\Delta\Omega$ is 3.36 steradians into the soil and the effective neutron attenuation factor through the concrete walls is

$$F(b) = 3.34 \times 10^{-6} .$$

Therefore, the activation of each isotope will be

$$F(b) ST \frac{\Delta\Omega}{4\pi} \frac{\Sigma_i}{\Sigma_s} ,$$

and the summation for all isotopes will be

$$F(b) ST \frac{\Delta\Omega}{4\pi} \sum \frac{\Sigma_i}{\Sigma_s} .$$

We note that $F(b)ST\ \Delta\Omega/4\pi$ will be neutrons entering the soil and will equal 1.92×10^{15} fast neutrons per year, and since Σ_i/Σ_s equals the number of disintegrations of isotope i per neutron entering the soil, we can establish the fraction

$$F(b) ST \frac{\Delta\Omega}{4\pi} \frac{\Sigma_i/\Sigma_s}{q_i} ,$$

which gives the fraction of permissible disintegrations of isotope i taking place per year. The summation of all similar fractions for the radioactive isotopes in the soil should not exceed 1--if the postulated permissible burial criteria are not to be exceeded.

In Table II.1 the value Σ_i/Σ_s appears in the third column, the calculated values of q_i appear in the fourth, and the ratio of Σ_i/Σ_s to q_i appears last. It will be noted that the value

$$\left(F(b)ST\,\frac{\Delta\Omega}{4\pi}\right)\left[\sum \frac{\Sigma_i/\Sigma_s}{q_i}\right] = 0.353 \; .$$

This indicates that by using the assumption originally stated, the soil activation is 35.3% of the permissible.

Table II.1

Elements in the Soil	Mass Number of Active Isotope	Σ_i/Σ_s p	Permissible Disintegrations Per Year, q_i	p/q_i
Al[a]	---	-----	-----	-----
B[a]	---	-----	-----	-----
Ba	137[b]	0.967×10^{-5}	0.100×10^{12}	0.967×10^{-16}
Be[a]	(c)	-----	-----	-----
Ca	45	0.65×10^{-4}	0.475×10^{-17}	0.347×10^{-21}
Co	60	0.795×10^{-3}	0.728×10^{16}	0.109×10^{-18}
Cr	51	0.577×10^{-4}	0.721×10^{17}	0.800×10^{-21}
Cu	64	0.276×10^{-3}	0.148×10^{16}	0.186×10^{-18}
Fe	55	0.915×10^{-2}	0.310×10^{18}	0.295×10^{-19}
Fe	59	0.207×10^{-3}	0.215×10^{16}	0.963×10^{-19}
Ga	72	0.141×10^{-4}	0.328×10^{15}	0.430×10^{-19}
K	42	0.497×10^{-2}	0.287×10^{15}	0.173×10^{-16}
La	140[b]	0.981×10^{-4}	0.928×10^{14}	0.106×10^{-17}
Mg[a]	---	-----	-----	-----
Mn	56	0.145×10^{-1}	0.298×10^{15}	0.487×10^{-16}
Na	24	0.623×10^{-2}	0.346×10^{15}	0.180×10^{-16}
Ni	59	0.225×10^{-3}	0.759×10^{16}	0.296×10^{-19}
Ni	63	0.433×10^{-4}	0.756×10^{16}	0.573×10^{-20}

Table II. 1--continued

Elements in the Soil	Mass Number of Active Isotope	Σ_i/Σ_s p	Permissible Disintegrations Per Year, q_i	p/q_i
Pb[a]	---	-----	-----	-----
Sc	46	0.454×10^{-3}	0.338×10^{16}	0.134×10^{-18}
Si[a]	---	-----	-----	-----
Sr	89[d]	0.140×10^{-6}	0.235×10^{16}	0.596×10^{-22}
Ti[a]	---	-----	-----	-----
V[a]	(c)	-----	-----	-----
Y	90[d]	0.304×10^{-4}	0.148×10^{14}	0.205×10^{-17}
Yb[a]	---	-----	-----	-----
Zr[a]	---	-----	-----	-----
H_2O[a]	---	-----	-----	-----
Total of ratio p/q_i				0.184×10^{-15}

[a] Element not present in this soil.

[b] The limit for $Ba^{137} + Cs^{137}$ was used, although Cs is not present in this soil. Ba^{140} is produced by secondary reaction only. The lower limit was chosen for La^{140} because Ba^{140} is present in the soil.

[c] Activity produced by second- and higher-order reactions only is negligible.

[d] The limit shown is for Sr^{89}; Sr^{90} is produced by secondary reaction only. The lower limit was used for Y^{90} because Sr is present in the soil.

Appendix III

PROCEDURES

The nuclear characteristics of the DORF-TRIGA are expected to be similar to those of the TRIGA prototype, which has undergone extensive testing. Further, the TRIGA Mark F reactor at General Atomic serves as the prototype for the DORF-TRIGA and provides experimental verification of the core characteristics.

The operating behavior of the DORF-TRIGA will be determined at the site by General Atomic personnel during the initial start-up phase of reactor operation. The procedures for this operation are described below.

INITIAL LOADING PROCEDURES

Before fuel is loaded into the reactor, the satisfactory performance of the mechanical and instrumentation systems of the reactor, including the action of all scrams, will be demonstrated.

The initial approach to criticality will be made with the reactor tank filled with water and with the reactor in such a position as to have an infinite water reflector (greater than 6 in. of water). The fuel elements will be added one at a time, and inverse multiplication plots will be kept on all available channels. When criticality is reached, a number of measurements will be made at low power. These will include

1. Control-rod calibration (all rods);
2. Measurement of fuel-element worth;
3. Adjustment of excess reactivity to 1.6% $\delta k/k$;
4. Reactivity changes as a function of reactor position and reflector composition, i.e., 2 in. H_2O, 2 in. H_2O + 2 in. Pb.

STEADY-STATE EXPERIMENTS

After criticality is reached, an excess reactivity of not more than 1.6% δk/k will be inserted in the reactor.

After low-power calibration experiments have been completed, power-level calibration experiments will be performed, using calorimetric and heat-balance methods. A schedule of power-level increases in approximately half-decade steps will be followed, from 1 w to 250 kw. During this phase, instrument linearity will be checked, and reactivity loss will be measured as a function of power level.

The fuel temperature coefficient will be checked by measuring the loss of reactivity versus temperature of at least two fuel elements in the B-ring, and then comparing these measurements with the same measurements made with the TRIGA Mark F prototype reactor. Provided that the measured temperature coefficient lies within 30% of the Mark F measurements, the proposed experimental program will be continued.

The following conditions will apply to high-power operation:

1. Maximum excess reactivity: 2.9% δk/k over cold, clean critical;
2. Maximum mixed mean pool temperature: 60°C;
3. Maximum water temperature above the fuel elements: 90°C or less, as measured by at least two thermocouples placed in the triflute region of the central fuel elements;
4. Maximum reactor power: 250 kw, for sustained periods; maximum integrated 250-kw operation not to exceed 1 Mw-hr per day.
5. Maximum fuel temperature: not to exceed 530°C.

Fuel-element temperature and reactivity-loss measurements will be made as a function of power up to 250 kw, at 25-kw increments. The reactor will be moved to a position with a 2-in. H_2O reflector, and the measurements will be repeated.

Following transient operation of the reactor, the power coefficient of reactivity will be remeasured to a power level of 250 kw, to determine the effect of previous flashing on steady-state characteristics.

TRANSIENT EXPERIMENTS

A series of step-reactivity-insertion experiments will be performed. This will be a repetition of transient experiments with the existing General Atomic pulsing reactor prototype, the TRIGA Mark F, in which the central control rod will be rapidly withdrawn (~ 0.1 sec) from the core by a pneumatic drive. The transient temperature in at least two central fuel elements will be measured and will activate a scram if it exceeds $530^{\circ}C$.

The series of power-transient experiments will start with a step insertion of 0.8% $\delta k/k$ and will continue in larger steps; the transient temperature rise will be increased in steps of $50^{\circ}C$ or less.

The limiting conditions that will apply are as follows:

1. The maximum transient temperature at the center of the inner fuel elements will be $530^{\circ}C$.
2. No transient experiments will be performed while damaged fuel elements are present in the core. Procedures will be instituted to periodically inspect the fuel elements.

The radiation dosage will be measured in the vicinity of the core with ion chambers, Hurst dosimeters, and threshold foil detectors.

The transient experiments will be initiated in the all-water-reflected geometry, with the reactor in the condition of maximum reactivity. The experiments will be repeated with the 2 in. H_2O + 2 in. Pb reflector.

Dose measurements in the exposure room will be made under steady-state and transient conditions, with ion chambers, Hurst dosimeters, and threshold detectors.

EMERGENCY PROCEDURES

In the event of an accident which contaminates the reactor room, it will be necessary to evacuate personnel for a short time. The general procedures leading up to such an evacuation are as follows: The reactor operator and/or supervisor will be continuously monitoring the control room, and it is most likely that they will be the first to observe any abnormalities associated with either the reactor or the radiation level of the area. In the event that abnormal behavior is observed by personnel other than the reactor supervisor or operator, the reactor supervisor should be advised immediately and it will be his responsibility to ensure that appropriate action is taken.

Should there by any indication that evacuation of the reactor room is advisable, the reactor supervisor will immediately issue such an order. The evacuation procedure calls for the carrying of adequate monitoring equipment for the purpose of detecting any leakage of contaminated air. The room will be re-entered only on the advice of the reactor supervisor, and the decision will be based on observations made by the health physicist.

In the unlikely event that evacuation of the building should be required, the reactor supervisor will put the procedures for evacuation into effect. To increase the effectiveness of evacuating a building, information on the direction and velocity of the prevailing winds is continuously available in the reactor room.

Appendix IV

CALCULATED MAXIMUM FISSION-PRODUCT RELEASE AFTER A FUEL-ELEMENT FAILURE

Calculations and a related experiment have been made to determine theoretically the maximum concentration of fission products that might be present in the reactor-room air following a fuel-element cladding failure.

The calculations are based on the fact that as the reactor operates, fission products will be built up in the uranium—zirconium hydride fuel mixture until an equilibrium concentration is reached for each nuclide, dependent on (1) the total energy release in the reactor, (2) the decay process for each nuclide, and (3) the yield of the species from fission. Of the various fission products produced in the fuel material, only certain nuclides will migrate into the gap between the fuel material and the fuel-element cladding. These nuclides are the iodines, the xenons, and the kryptons and their decay products.

CALCULATION OF GASEOUS FISSION PRODUCTS

In the event of a rupture of the fuel-element cladding, the fission products released to the water in the reactor tank will be limited to all or a portion of those fission products that have collected in the gap between the fuel material and the aluminum cladding. A portion of these fission products will then go into the air above the water, this fraction depending on the solubility of the species in water.

The quantity of gaseous fission products produced in the fuel element was determined by Blomcke and Todd (Report ORNL-2127). The amounts of krypton, xenon, and iodine produced in a typical element after infinite operation at 100 kw are given in Table IV. 1. These data are based on a loading of 36 g of U^{235} (9.4×10^{22} nuclei) and an average flux of

10^{12} thermal neutrons/ cm^2-sec. N_s is the number of nuclei of isotope S in the element, and $N_{25}o$ is the initial number of U^{235} nuclei (9.4×10^{22}).

Table IV. 1

GASEOUS FISSION PRODUCTS PRODUCED IN FUEL ELEMENT

Isotope	$N_s/N_{25}o$	N_s	Decay Constant (sec^{-1})	Activity (c)
Kr^{83m}	7.1×10^{-8}	2.7×10^{15}	0.101×10^{-3}	7.2
Kr^{85m}	3.2×10^{-7}	1.2×10^{16}	4.4×10^{-5}	14.
Kr^{85}	2.0×10^{-3}	7.6×10^{19}	2.2×10^{-9}	4.4
Kr^{87}	2.7×10^{-7}	1.0×10^{16}	0.15×10^{-3}	40.
Kr^{88}	7.7×10^{-7}	2.9×10^{16}	7.0×10^{-5}	54.
Kr^{89}	1.8×10^{-8}	6.8×10^{14}	3.6×10^{-3}	66.
Kr^{90}	3.8×10^{-9}	1.4×10^{14}	2.1×10^{-2}	82.
Kr^{92}	1.7×10^{-10}	6.4×10^{12}	2.3×10^{-1}	40.
I^{130}	9.5×10^{-7}	3.6×10^{16}	1.5×10^{-5}	14.
I^{131}	4.2×10^{-5}	1.6×10^{18}	1.0×10^{-6}	44.
Xe^{131m}	6.3×10^{-7}	2.4×10^{16}	6.7×10^{-7}	0.4
I^{132}	7.8×10^{-7}	2.9×10^{16}	8.0×10^{-5}	63.
I^{133}	1.0×10^{-5}	3.8×10^{17}	9.3×10^{-6}	94.
Xe^{133m}	6.3×10^{-7}	2.4×10^{16}	3.5×10^{-6}	2.4
Xe^{133}	6.3×10^{-5}	2.4×10^{18}	1.5×10^{-6}	96.
I^{134}	5.0×10^{-7}	1.9×10^{16}	2.2×10^{-4}	119.
I^{135}	2.9×10^{-6}	1.1×10^{17}	2.9×10^{-5}	84.
Xe^{135m}	3.6×10^{-8}	1.4×10^{15}	7.4×10^{-4}	27.
Xe^{135}	3.8×10^{-6}	1.4×10^{17}	2.1×10^{-5}	82.
I^{136}	5.6×10^{-9}	2.1×10^{14}	8.1×10^{-3}	46.
Xe^{137}	2.9×10^{-8}	1.1×10^{15}	3.0×10^{-3}	128.
I^{138}	4.2×10^{-10}	1.6×10^{13}	1.2×10^{-1}	52.

Table IV. 1--continued

Isotope	N_s/N_{25}^o	N_s	Decay Constant (sec^{-1})	Activity (c)
Xe^{138}	1.2×10^{-7}	4.4×10^{15}	6.8×10^{-4}	82.
I^{139}	9.8×10^{-11}	3.7×10^{12}	2.6×10^{-1}	25.
Xe^{139}	4.0×10^{-9}	1.5×10^{14}	1.7×10^{-2}	70.
Xe^{140}	1.3×10^{-9}	4.8×10^{13}	4.3×10^{-2}	56.

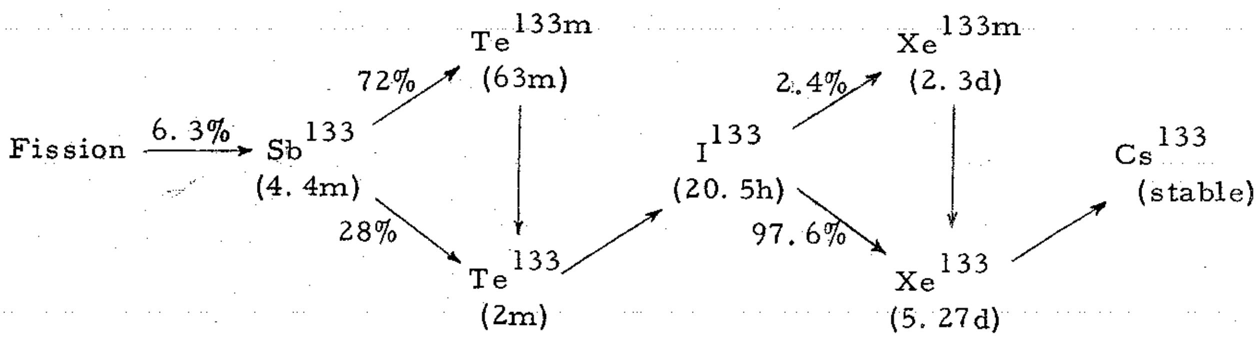
The sum of all the activities of the gaseous fission products in an element after it has reached equilibrium at 100 kw is therefore 1392 c.

EXPERIMENTAL DETERMINATION OF FISSION PRODUCTS IN GAP

In order to determine the actual percentage of fission-product gases which escape the fuel material and collect in the air gap between the cladding and the fuel material, the following experiment was conducted in the TRIGA reactor at General Atomic. For the purpose of this experiment a fuel element was fabricated with a sealed tube venting the gap to a charcoal trap at the surface of the reactor tank. All of the fission-product gases which accumulated in the gap were then collected in a liquid-air-cooled charcoal trap by purging the system with helium, and the traps were then analyzed. This measured amount of radioactive noble gases enables the determination of the fraction of the fission products which diffused through the uranium—zirconium hydride material into the gap.

In the experiment a total of 210 kw-hr of reactor operation occurred in 1 hr. The test element was exposed to an average thermal flux of 2.1×10^{12} neutrons/cm^2-sec during this time, and the Xe^{133} that collected in the gap in this period of operation was determined to be 78 μc. Through determination of the Xe^{133} that diffuses into the gap during the exposure of the element to a known amount of reactor flux, the percentage of Xe^{133}, and consequently of all other fission-product gases that collect into the gap, can be determined.

During the experiment, Xe^{133} was produced in the fuel element from the chain:



The activity of Xe^{133} present in the element at the end of one hour at 2.1×10^{12} neutrons/cm^2-sec was determined from the work of Bolles and Ballou.* In this work the number of Xe^{133} nuclei present after the fission of 10,000 U^{235} nuclei is given as a function of time. Integrating their data over a time of 1 hr gives 0.49×10^{-3} Xe^{133} nuclei present per fission. In the 1-hr experiment the number of fissions that occur is

$$\text{Number of fissions} = 9.4 \times 10^{22} \frac{U^{235}\text{ atoms}}{\text{element}} \times 5.83 \times 10^{-22} \frac{\text{cm}^2}{U^{235}\text{ atom}}$$

$$\times\, 2.1 \times 10^{12} \frac{\text{neutrons}}{\text{cm}^2\text{-sec}} \times 3.6 \times 10^3 \text{ sec}$$

$$= 4.15 \times 10^{17}.$$

Thus, the number of Xe^{133} nuclei is $0.49 \times 10^{-3} \times 4.15 \times 10^{17}$, or 2.03×10^{14} nuclei. This gives a total available Xe^{133} activity of

$$A = \frac{1.52\times10^{-6} \times 2.03\times10^{14}}{3.7 \times 10^4}\ \mu c = 8.36\times10^3\ \mu c\ .$$

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*Bolles, R. C., and N. E. Ballou, Calculated Activities and Abundances of U^{235} Fission Products, USNRDL-456, August 30, 1956.

As was indicated previously, the amount of Xe^{133} that was collected in the gap at the end of 1 hr was 78 μc. The percentage of the total inventory of Xe^{133} that escaped to the gap between fuel and cladding can therefore be calculated as follows:

$$\frac{Xe^{133} \text{ measured in gap}}{Xe^{133} \text{ total available}} = \frac{78.\ \mu c}{8.36 \times 10^{3} \mu c} = 0.9\% .$$

As shown from Table IV. 1, at 100 kw the total quantity of all fission-product gases in a TRIGA fuel element at equilibrium is 1392 c. For the purposes of this calculation it is assumed that the fraction of the iodine, krypton, and xenon isotopes produced that collect in the gap between fuel and cladding is the same as that determined for Xe^{133}. Thus, the total gaseous activity in the gap is calculated to be 13 c.

Therefore, the maximum amount of fission products that could be released in the event of a cladding failure is

Iodines	5.1 c
Xenons	5.0 c
Kryptons	2.9 c
Total fission products	13.0 c

The volume of water in the DORF reactor tank is approximately 5.77×10^{7} cm^{3}. For purposes of this calculation it is assumed that the total 13 c of gaseous fission products in the gap escapes into the water. However, when these fission products (the iodines, kryptons, and xenons) are released from the fuel element into the water, the iodines will be dissolved in the water. Although the kryptons and xenons are quite soluble in water, it has been conservatively assumed for purposes of this analysis that a major portion of these isotopes will escape into the reactor-room atmosphere. For the purpose of the calculations presented below, it is assumed that 95% of the xenon and 98% of the krypton will escape into the air and that all of the iodines will remain in the water. Therefore, the activity in the water will be

$$A_{water}\ \frac{\mu c}{cm^3} = \frac{A(\text{iodine})(\mu c) + 0.05\ A(\text{xenon})(\mu c) + 0.02\ A(\text{krypton})(\mu c)}{\text{volume of water } (cm^3)} ,$$

giving

$$A_{water} = \frac{5.1 + 0.05 \times 5.0 + 0.02 \times 2.9}{5.77\times10^7\ cm^3} \times 10^6\ \mu c = \frac{5.41\times10^6\ \mu c}{5.7\times10^7\ cm^3}$$

$$= 9.5\times10^{-2}\ \mu c/cm^3 ,$$

and the activity in the air will be

$$A_{air}\ \frac{\mu c}{cm^3} = \frac{0.95A(\text{xenon})(\mu c) + 0.98A(\text{krypton})(\mu c)}{\text{volumes of air } (cm^3)}$$

$$= \frac{0.95 \times 5.0 + 98 \times 2.9}{2.02\times10^9\ cm^3} \times 10^6\ \mu c = \frac{7.59\times10^6\ \mu c}{2.02\times10^9\ cm^3}$$

$$= 2.9\times10^{-3} \mu c/cm^3 .$$

Since krypton and xenon are inert gases, the hazard presented by their being in the air is from the dose a person in the room would receive from their decay. To estimate this dose, it is assumed that each decay of a krypton or xenon nucleus results in the emission of a 0.5-Mev gamma photon. Then the dose rate at the center of a hemispherical volume of 71,350 ft (equivalent radius = 25.9 ft) is

$$D = \frac{3.75\times10^{-9}\ c/cm^3 \times 3.7\times10^{10}\ \text{photons/sec-c}\ (1 - e^{-8\times10^{-5}\times25.9\times30.5})}{2 \times 8\times10^{-5}\ cm^{-1} \times 1.085\times10^6\ \text{photons}/cm^2\text{-sec/rad/hr}}$$

$$= 0.803\ (1 - e^{-0.0632}) = 0.049\ \text{rads/hr} = 49\ \text{mr/hr} ,$$

where the attenuation coefficient for air is taken as $8\times10^{-5}\ cm^{-1}$ and the flux-to-dose conversion factor is 1.085×10^6.

To estimate an integrated dose to a person remaining in the reactor room for 1 hr after the rupture, the xenon and krypton activities in the room were averaged over 1 hr. This average is 2.45 c, or an average concentration of 1.21×10^{-3} $\mu c/cm^3$.

If we assume that each disintegration is accompanied by a 0.5-Mev photon and that the room can be approximated by a hemisphere of equivalent volume, then by proportion, using the above initial dose value, the average dose rate will be

$$D = \frac{1.21 \times 10^{-9}}{3.75 \times 10^{-9}} \times 49 \text{ mr/hr} = 15.7 \text{ mr/hr} ,$$

and the integrated dose for 1 hr will be approximately 15.7 mr.

It can thus be observed that even on the basis of the conservative assumptions made herein, a person could remain in the reactor room for several hours after the fuel-element cladding failure without exceeding the permissible radiation dose limits specified in 10 CFR 20. Standard operating procedures for the facility will require prompt evacuation of the reactor room by the operating personnel on indication by the continuous air monitor of airborne radioactivity in the reactor room. Upon the detection of airborne radioactivity, the air in the building will be isolated from the surrounding atmosphere and confined until it is determined by the Health Physicist that the airborne radioactivity limits are safe for normal operations to resume.

Appendix V

LOSS-OF-COOLANT CALCULATIONS

HEAT REMOVAL

Calculations have been made to determine the amount of heat that will be removed from a central element in the TRIGA core by natural convection of air through the core in the event cooling water is lost. The determination of this heat-removal rate, coupled with the calculation of afterheat production rates in the element, was used to find maximum operating powers and times that could be permitted without endangering the structural integrity of the fuel element.

The heat-removal rate by natural convection of air between three central elements was determined. The flow channel considered was bound by a B-ring element and two C-ring elements. The flow area in the channel was 0.327 in.^{2} and the length of the channel was 2 ft. In the center regions of the bottom grid plate 0.625-in.-ID holes were made to provide coolant flow through the center of the core. One of these holes was located at the bottom of the channel considered. Flow out of the top of the channel is through the holes in the top grid plate through which the elements are loaded. These top-grid-plate holes are obstructed by the spacer used to center the element. The effective flow area through the top grid plate is approximately 0.5 in.^{2}.

To find the weight flow of air through the channel, the driving pressure was equated to the pressure loss in the channel. As a first approximation, the entrance and exit losses were neglected. The driving pressure is given by

$$P_d = (\rho_0 - \rho_a)L \ , \qquad \text{(V-1)}$$

where ρ_0 is the ambient air density (in lb/ft^3), ρ_a is the average density of the air in the channel, and L is the length of the channel (in ft.). The frictional pressure loss in the channel is given by

$$P_f = \frac{2L\mu w}{g_c R_h^2 A_c \rho_a} , \qquad \text{(V-2)}$$

where μ is the viscosity of the air (in lb/ft-sec), w is the air weight flow (in lb/sec), g_c is 32.2 ft/sec^2, R_h is the hydraulic radius, which is equal to channel flow area/wetted perimeter (in ft), and A_c is the channel flow area (in ft^2).

As the density and viscosity of the air are functions of the air temperature, the weight flow was determined for a range of average air temperature in the channel. For this weight flow, the heat-removal rate required to raise the average air temperature to the given value is

$$q_r = 2(T_a - T_0)c_p w,$$

where q_r is the heat absorbed by the cooling air, T_a is the average air temperature, T_0 is the ambient air temperature (75°F), and c_p is the specific heat of air (0.25 Btu/(lb)($^\circ$F).

For the same range of average air temperature, the heat-transfer film coefficient, h, was determined from a nomograph in the Chemical Engineer's Handbook. To find this film coefficient, it is necessary to know the surface temperature of the wall. Since the maximum temperature permissible for the fuel-element cladding is 1100°F, that temperature was used. By using this wall temperature and the film coefficient just determined, the rate of heat transferred to the air can then be found for the range of average air temperatures by

$$q_t = hA(T_w - T_a), \qquad \text{(V-3)}$$

where q_t is the heat-transfer rate from the fuel-element wall to the cooling air, A is the surface area of the wall (0.1635 ft^2) and T_w is the wall temperature (1100°F).

The temperatures at which these two heat-transfer rates are equal establishes the conditions that will exist. Figure 36 shows q_r and q_t plotted versus T_a, and at an average air temperature of $320^\circ F$, the heat-transfer rate to the air will be 163 Btu/hr in this channel.

The power level in the fuel elements of this channel was assumed to be twice the average power level in the reactor, which contains 85 elements. Therefore, when the fuel-element cladding temperature is $1100^\circ F$, the total heat-removal rate is

$$q_{total} = 75 \times 163 \text{ Btu/hr} \times 0.293 \text{ w/(Btu/hr)} = 3600 \text{ w} .$$

The energy release in the reactor required to raise a central fuel-element cladding temperature to $1100^\circ F$ is

$$E = 1/2\, C\, \delta T,$$

where C is the heat capacity of the reactor (55 kw-sec/$^\circ$C), per data from R. Stahl,* and δT is the temperature rise of the element. The fuel-element temperature is assumed to be $400^\circ F$ when the cooling water is lost, so $\delta T = 1100^\circ F - 400^\circ F = 700^\circ F$, and

$$E = \frac{55}{2} \times 700 \times \frac{5}{9} = 1.07 \times 10^7 \text{ w-sec.}$$

The factor of 2 in the denominator is the ratio of the power in the central element to the average power per element.

From the above the following criteria can be established: If at the time the afterheat power level in the reactor decreases to 3600 w a total of more than 1.07×10^7 w-sec of energy has been released after the water loss in the reactor, the reactor core may be damaged. Conversely, if at the time the power reaches 3600 w less than 1.07×10^7 w-sec has been released, the reactor core will not be damaged.

*R. Stahl, private communication.

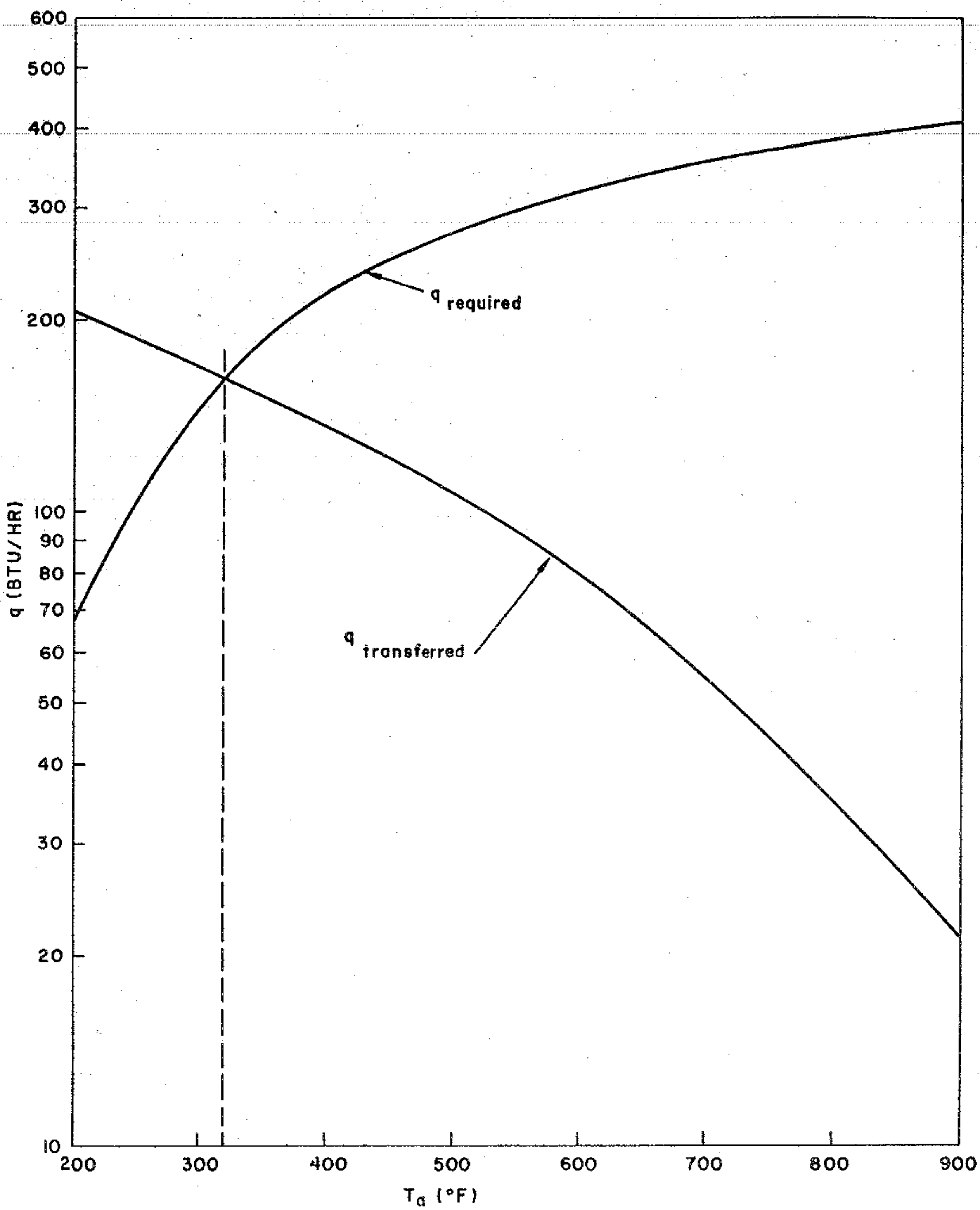

Fig. 36--Required heat-transfer rate to achieve T_a and heat-transfer rate at T_a versus average air temperature (T_a)

AFTERHEAT

The decay power after shutdown was calculated from the data of Stehn and Clancy.* Figure 37 shows the ratio of decay power to operating power as a function of time after shutdown for reactors that have been operating continuously for infinite time prior to shutdown and that have been operating 8 hr per day for infinite time prior to shutdown. Figure 38 shows the decay power for a reactor that has been operating at 100 kw continuously prior to operation for several different times at 1 Mw before shutdown. From these figures the time after shutdown that the decay power decreases to 3600 w can be found.

Integration of these curves over time after shutdown yields the energy release in the reactor. The decay-energy release as a function of time after shutdown for the three cases described above is shown in Figs. 39 and 40. The time after shutdown at which 1.07×10^7 w-sec of decay heat has been released can be found by reference to those figures.

PERMISSIBLE OPERATING POWERS AND TIMES

It is now possible to determine (1) the time at which the decay power decreases to the power that can be removed by natural convection of air (i.e., 3600 w) and (2) the time at which the fuel-element cladding temperature will rise to 1100°F (i.e., the time at which 1.07×10^7 w-sec have been released). These two times have been plotted as a function of operating power for the case of continuous operation and for 8 hr/day operation in Figs. 41 and 42. The point at which the two curves intersect establishes the maximum power permissible. For continuous operation, the maximum power is 250 kw. For operation of the reactor 8 hr/day, the maximum power is 380 kw.

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*J. R. Stehn and E. F. Clancy, "Fission-product Radioactivity and Heat Generation," Paper No. 1071, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, United Nations, Geneva, September, 1958.

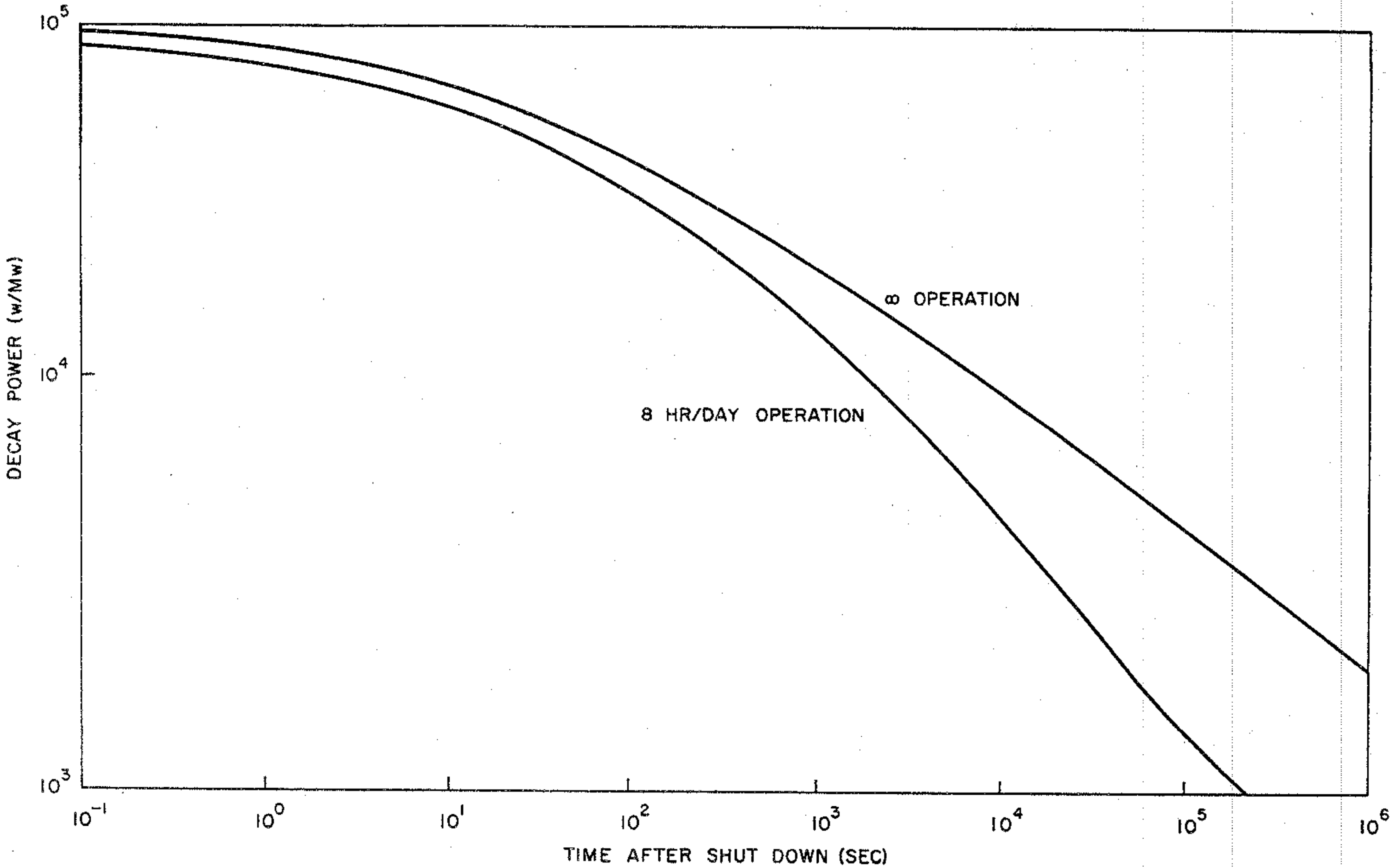

Fig. 37--Fission-product power release versus time after shutdown for infinite time operation and 8/hr per day

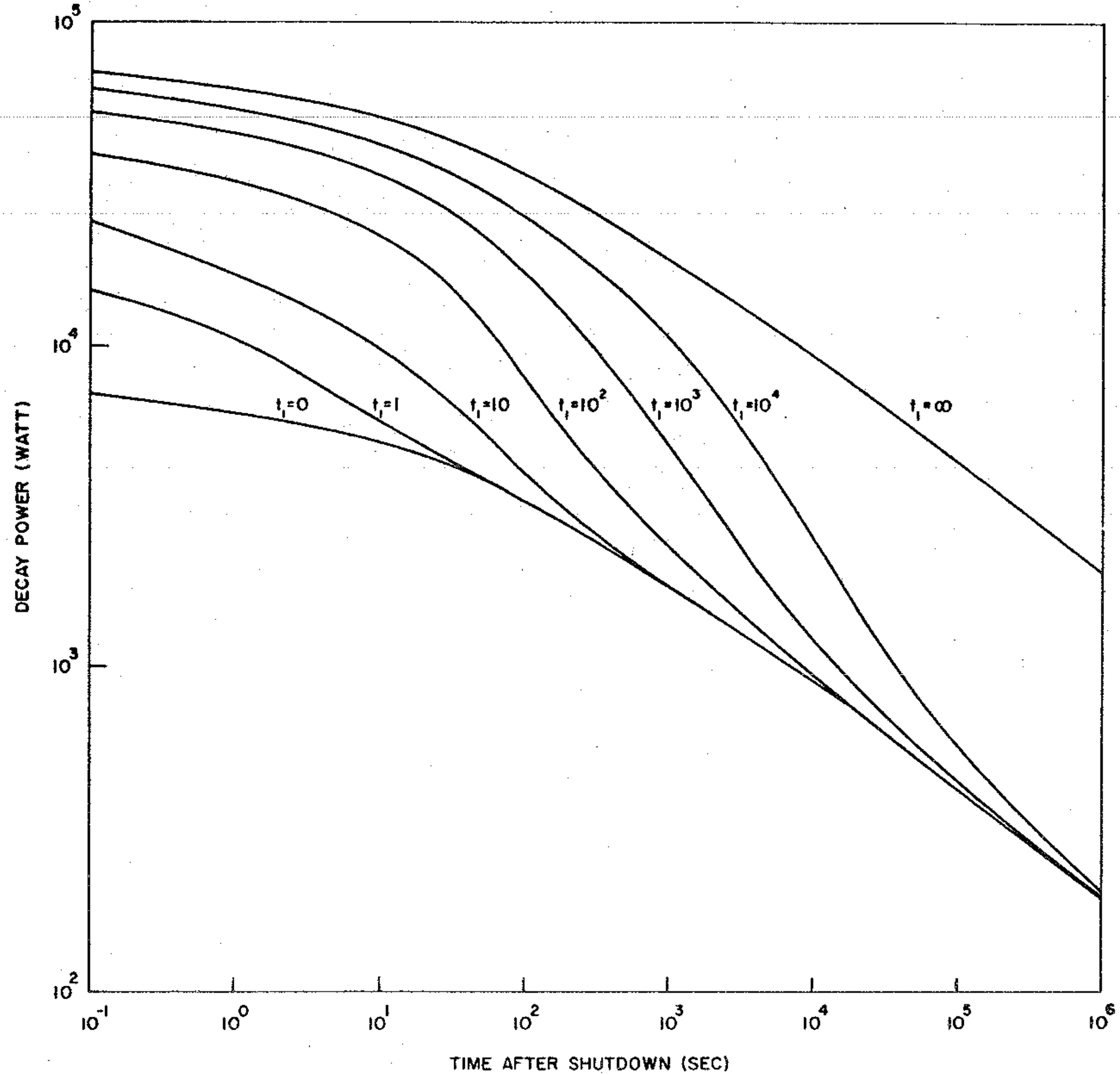

Fig. 38--Fission-product power release versus time after shutdown for 100-kw long-term operation followed by 1-Mw operation for various durations

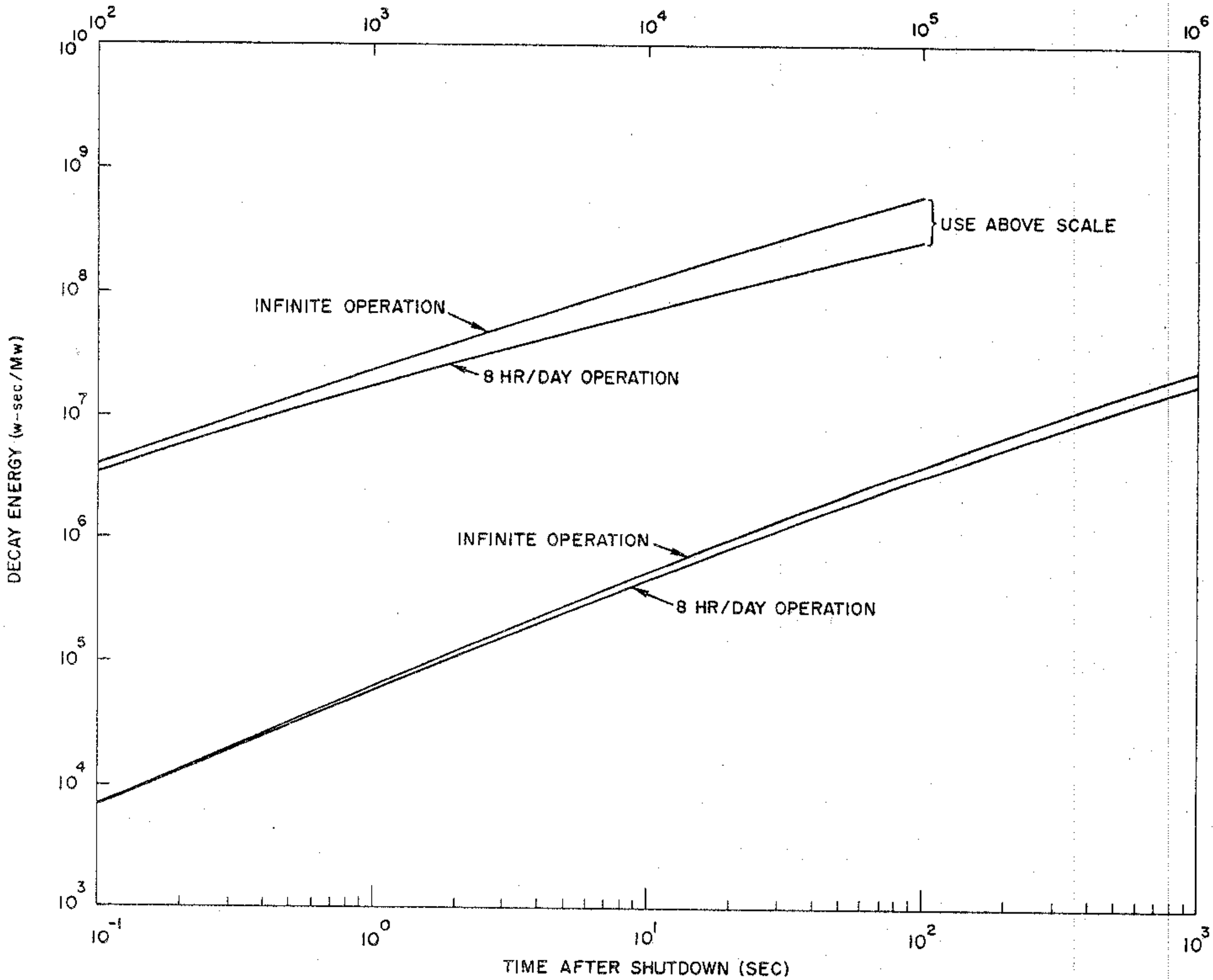

Fig. 39--Fission-product total energy release versus time after shutdown for infinite time operation and 8 hr per day

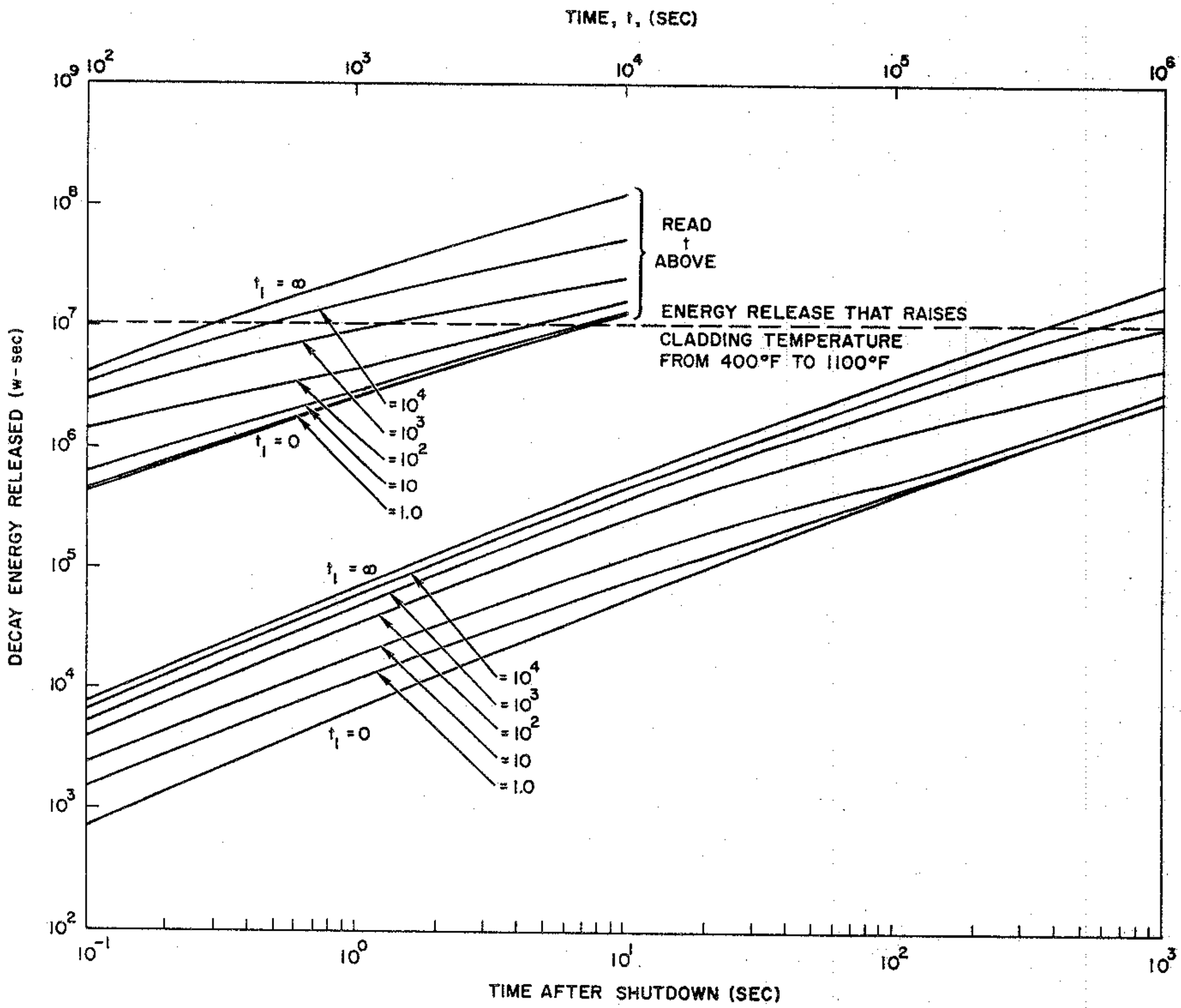

Fig. 40--Fission product energy release for infinite operation at 100 kw followed by t_i sec at 1 Mw

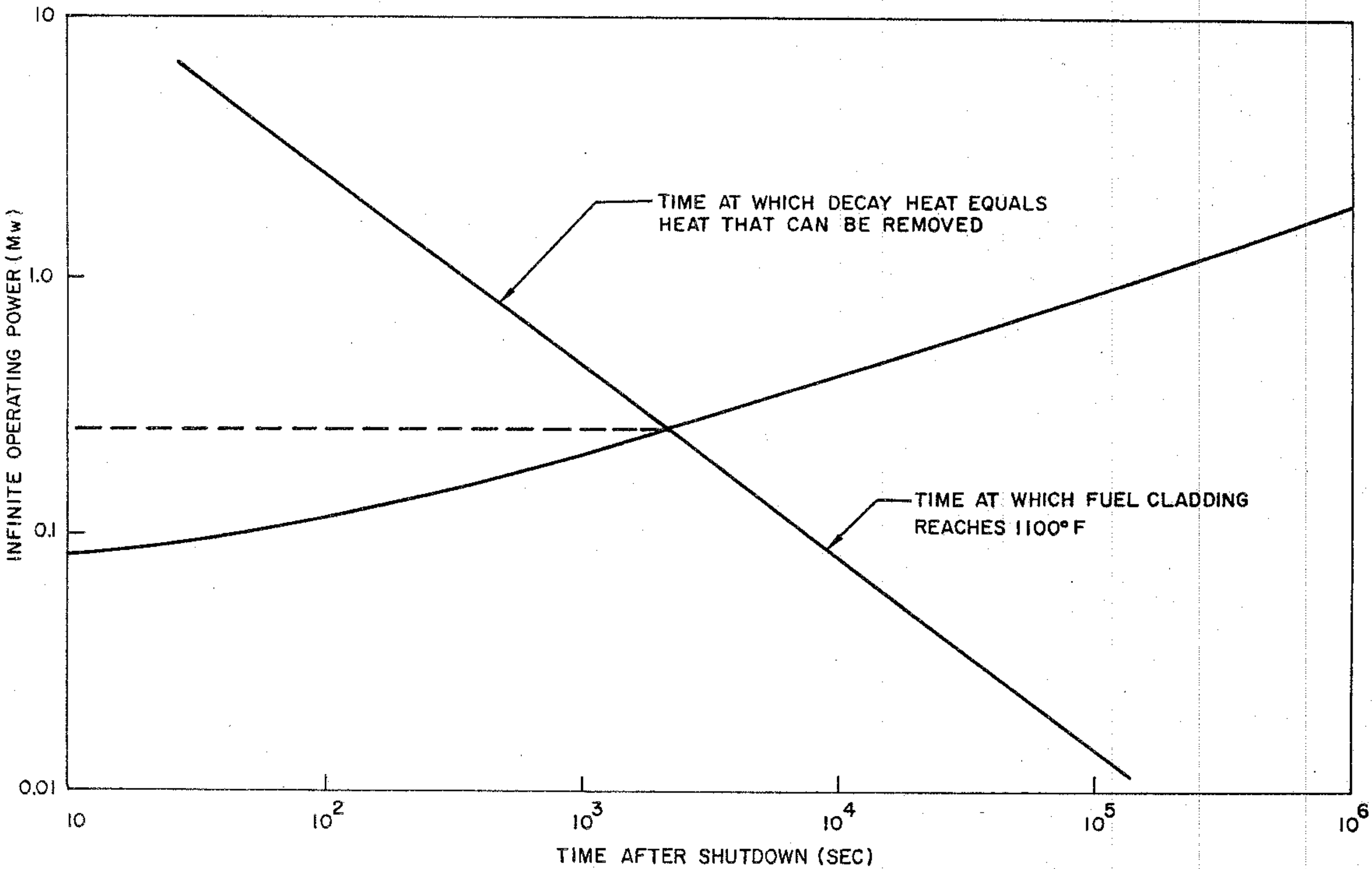

Fig. 41--Determination of safe infinite time operating power prior to loss of cooling water

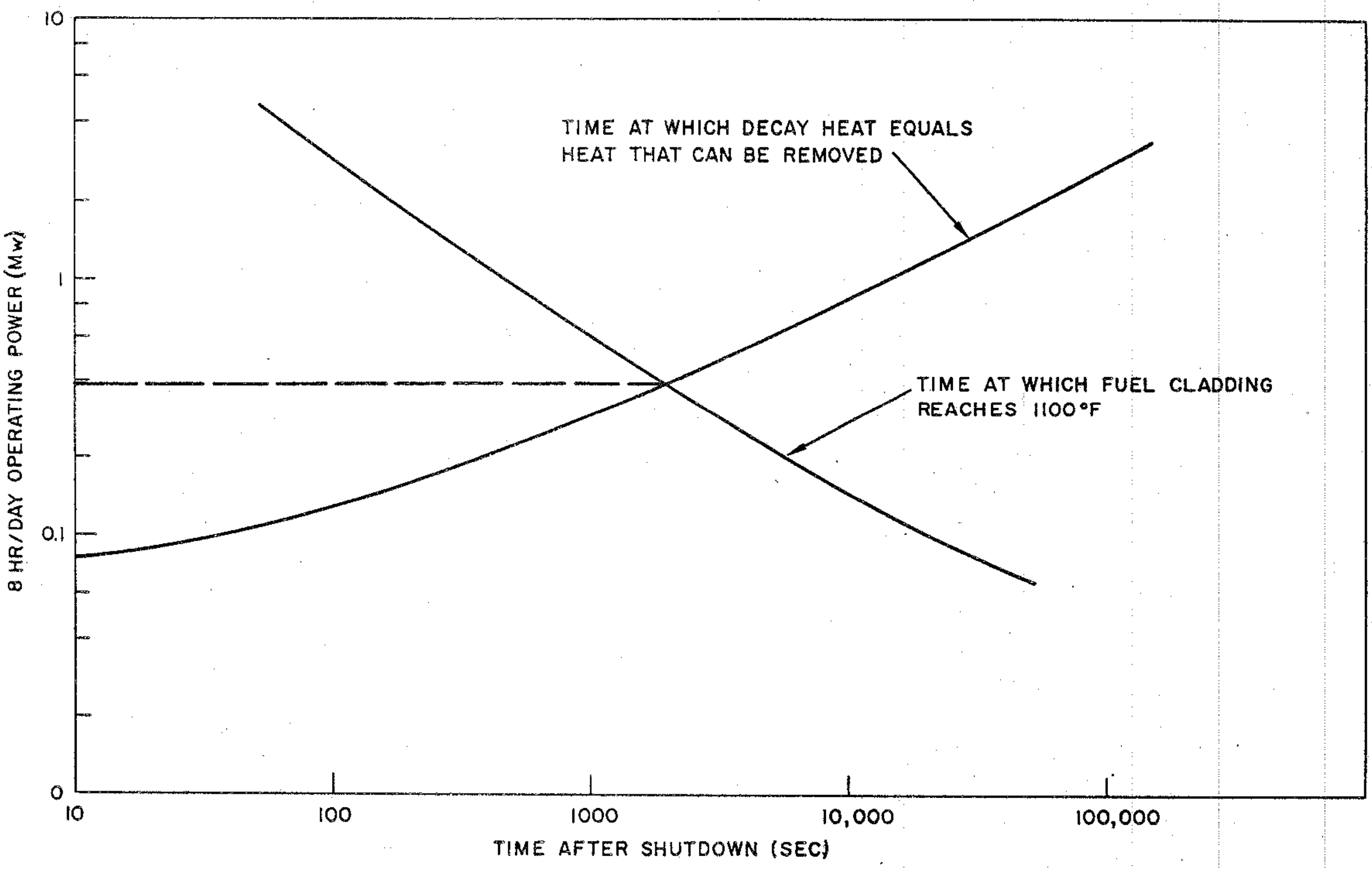

Fig. 42--Determination of safe 8-hr/day operating power prior to loss of cooling water

The time at which the decay power decreases to 3600 w and the time at which 1.07×10^7 w-sec is released are also plotted against the operating time at 1 Mw for the case of 100-kw continuous operation followed by 1-Mw operation. These data are in Fig. 43. The intersection of the two curves establishes the maximum operating time at 1 Mw after 100-kw continuous operation. This time is 1250 sec (~21 min). For 1-Mw operation after an 8 hr/day operation at 100 kw, interpolation of the data for continuous operation shows that the maximum 1-Mw operating time would be 1450 sec (~24 min), as indicated in Fig. 44.

Successive 1-Mw operating periods must be spaced in time to maintain the average power at 100 kw. Thus, 1-Mw operation for the maximum period of ~24 min should not be attempted more than twice in 8 hr, with at least 3-1/2 hr of cooling time in between. For shorter periods of operation, the cooling periods would be correspondingly shorter.

DISCUSSION OF RESULTS

These calculations have been made using some conservative assumptions. To substantiate the conclusions stated earlier, the following discussion of these assumptions is presented.

First, account has not been taken of any heat-removal mechanism other than natural convection of air through the core. Other mechanisms, however, will play a part, principally conduction to the grid plate and other structural members of the assembly. Heat removal by this means has been estimated roughly to be at the same rate as by convection. This means that the maximum permissible operating powers or times would be increased by 20% to 40%.

It has been assumed that the water has been lost instantaneously and completely from the core. Even if the water is lost by some inconceivable disaster that opens the bottom of the tank, it will still take a finite time for the out-rushing water to pass through the core. During this time afterheat will be removed.

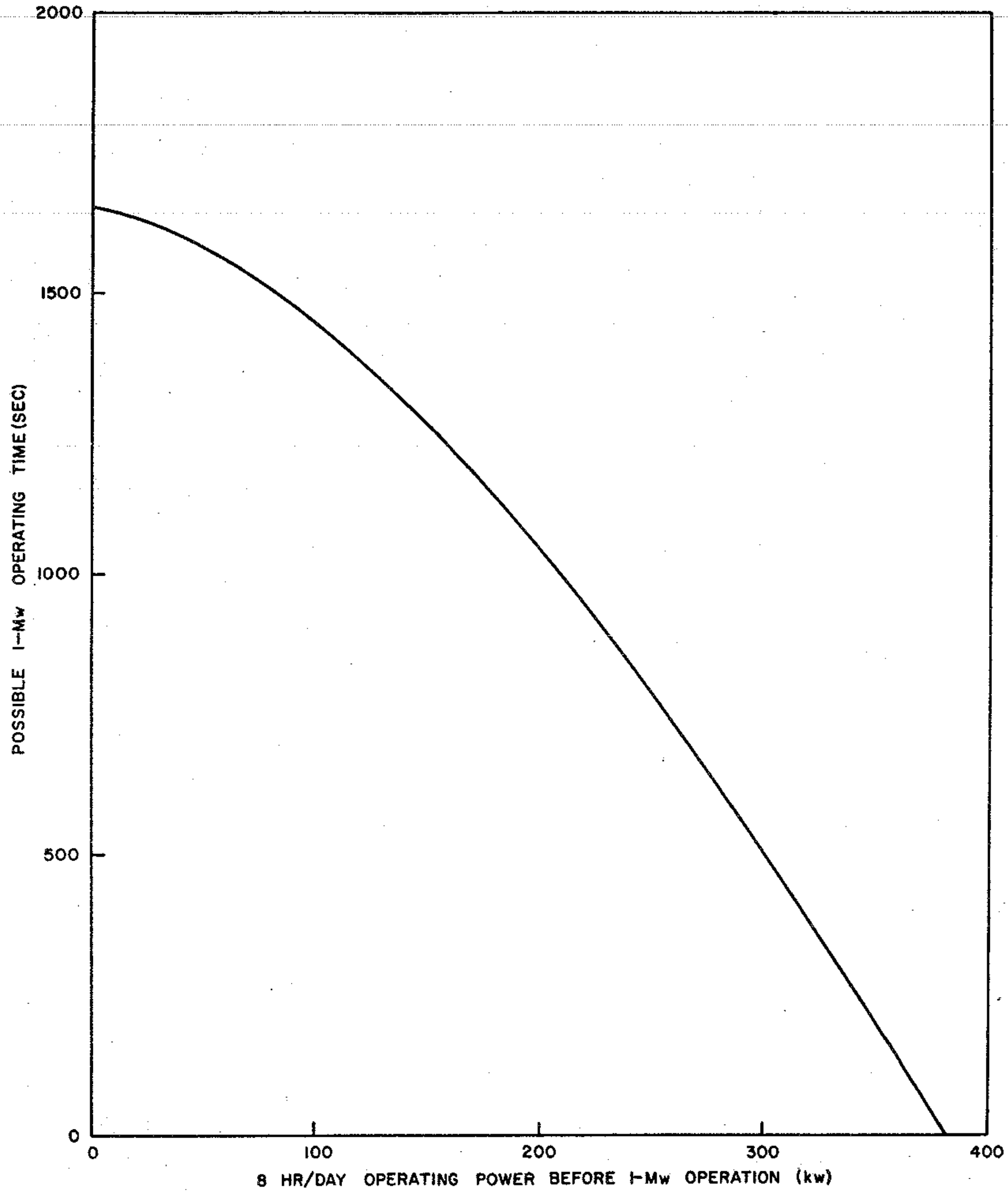

Fig. 44--Permissible operating time at 1 Mw after operation at average 8 hr/day power

Several other assumptions, that are conservative but have less effect on the results than those above, were made. One of these is that the driving force on the air passing up through the core has been calculated by assuming that only the air in the core is hotter than ambient. However, the air above the core will be hotter, creating an effective "stack" above the core. This means that the weight flow of air through the core will be somewhat greater than that calculated and, consequently, more heat can be removed by the air.

Also, it was assumed that no heat is removed by the air while the fuel-element temperature was rising to its peak value. Actually, the air would be cooling the element even as the element temperature rises, but this cooling will be small.

It may be noted that only pressure losses due to friction are considered in the flow through the channel. There will also be entrance and exit losses in the grid plates, but at the flow velocities being dealt with, these losses are so small they have been ignored.