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Certification Docket for the Remedial Action Performed at the Chapman Valve Site Indian Orchard, Massachusetts

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CERTIFICATION DOCKET FOR THE REMEDIAL ACTION PERFORMED AT THE CHAPMAN VALVE SITE IN INDIAN ORCHARD, MASSACHUSETTS

August 2003

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ACRONYMS

AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
BNI	Bechtel National, Inc.
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CHV	Chapman Valve Site
DCG	derived concentration guide
DOE	U.S. Department of Energy
EPA	U.S. Environmental Protection Agency
FIDLER	field instrument for detecting low-energy radiation
FUSRAP	Formerly Utilized Sites Remedial Action Program
HEPA	high-efficiency particulate air
IVC	independent verification contractor
MED	Manhattan Engineer District
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PPE	personal protective equipment
PIC	pressurized ionization chamber
RCRA	Resource Conservation and Recovery Act
RS S	radiological support subcontractor
SAIL	Science Applications International Corporation
TCLP	toxicity characteristic leaching procedure
USAGE	U.S. Army Corps of Engineers

UNITS OF MEASURE

µCi	microcurie
µR	microroentgen
cm	centimeter
dpm	disintegrations per minute
ft	foot
g	gram
h	hour
in.	inch
km	kilometer
m	meter
mi	mile
mL	milliliter
mrem	millirem
pCi	picocurie
yd	yard
yr	year

EXECUTIVE SUMMARY

The purposes of this certification docket are to provide a consolidated and permanent record of U.S. Department of Energy (DOE) activities at the former Chapman Valve site in Indian Orchard, Massachusetts, and to document radiological conditions at the property at the conclusion of remedial action. Remediation of the property was performed in 1995 under DOE's Formerly Utilized Sites Remedial Action Program (FUSRAP), a program established to identify and remediate, or otherwise control, sites where residual radioactive contamination remains from activities carried out by the Manhattan Engineer District/Atomic Energy Commission during the early years of the nation's atomic energy program.

FUSRAP was administered by DOE until October 1997, when the U.S. Congress reassigned responsibility for management of the program to the U.S. Army Corps of Engineers (USAGE). Completion of the certification process was delayed pending preparation of a Memorandum of Understanding between DOE and USAGE with regard to completed, remediated sites such as the Chapman Valve site.

Bechtel National, Inc. (BNI) was the project management contractor for work conducted at the Chapman Valve site. Thermo Nutech Services, Inc., served as the radiological support subcontractor for analyzing, sampling, and providing health physics technological support for site activities. Science Applications International Corporation (SAIL) was the environmental studies contractor responsible for assisting DOE in preparing the project environmental documentation. Oak Ridge National Laboratory was the independent verification contractor for the remediation work performed.

The material in this docket includes information and documents supporting certification that conditions at the property are in compliance with radiological guidelines in effect at the conclusion of remedial action. Furthermore, this certification docket substantiates that the future use of the property will not produce any significant radiological hazard or dose to the general public as a result of residual radioactivity remaining onsite that originated during activities conducted by DOE or its predecessor agencies.

Exhibit I of this docket is a summary of remedial activities conducted at the Chapman Valve site. The exhibit provides a brief history of the origin of the contamination at the site, the radiological characterization activities conducted, the remedial actions performed, post-remedial action survey and soil sampling results, and independent verification activities.

Exhibit II provides a listing of references documenting the entire remedial action process from designation of the site under FUSRAP to the certification that no radiological restrictions limit the future use of the site. Provided as Exhibit III is the DOE statement certifying that the property is in compliance with DOE radiological decontamination criteria and standards in effect at the conclusion of remedial action and the published Federal Register notice announcing the completion of remediation.

The certification docket and supporting documents identified in Exhibit II will be available for public review at the following locations:

U.S. Department of Energy
Public Reading Room
Room 1E-19
Forrestal Building
1000 Independence Avenue, S.W.
Washington, D.C. 20585

U.S. Department of Energy
DOE Information Center
475 Oak Ridge Turnpike
Oak Ridge, Tennessee 37831

Springfield Museum and Library
220 State Street
Springfield, Massachusetts 01103

Further information on the docket is available by contacting:

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U.S. Department of Energy
Washington, D.C. 20585-2040
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EXHIBIT I:

SUMMARY OF REMEDIAL ACTIVITIES AT THE FORMER CHAPMAN VALVE SITE

1.0 SITE HISTORY

The Chapman Valve site (CHV) is located at 203 Hampshire Street in Indian Orchard, a suburb of Springfield, Massachusetts (Figure I-1). The site was formerly owned and operated by the Chapman Valve Manufacturing Company. In 1948, this company set aside an area in the western end of Building 23 for the machining of uranium rods for the Atomic Energy Commission's (AEC's) Brookhaven National Laboratory (Figure I-2). According to a former supervisor at the site, the work area was segregated from the rest of the facility by a floor-to-ceiling wooden partition that was more than 50 ft high. The area was equipped with building shields, quenching tanks, suction systems, cranes, and special ducts. Uranium operations were terminated on November 8, 1948, at which time Chapman Valve had in its possession over 27,000 pounds of contaminated metal scrap, metal oxides, and sweepings. This material was removed from the site several months after contract work ended.

The Crane Company has owned the site since 1981; however, the company disconnected all utilities and vacated Building 23 in 1987. In 1991, Building 23 was still standing and in reasonably good condition, but harsh winter conditions had begun to deteriorate the building, and a structural inspection indicated that the roof was unsafe (Exh. II, Ref. 12).

2.0 RADIOLOGICAL HISTORY AND STATUS

2.1 RADIOLOGICAL AND CHEMICAL SURVEYS

A survey performed by ORNL in 1991 indicated that the residual uranium contamination found at CHV was typical of Manhattan Engineer District/AEC operations (Exh. II, Ref. 13). Results of this survey determined that the contamination was limited to the interior of the segregated area within Building 23 and included floors, walls, and overhead beams (see Figure I-2).

Additional radiological surveys were performed in November and December 1994 to supplement and refine existing survey information (Exh. II, Ref 12). Characterization confirmed the presence of contamination, predominantly in the western end of Building 23. In addition to confirming the ORNL 1991 survey results, these findings are in agreement with historical process information obtained during characterization. Characterization results show that contamination on the walls decreased significantly east of where a temporary wall had been constructed from floor to ceiling. Survey and sampling results from the east end of the building indicated near-background radiological conditions.

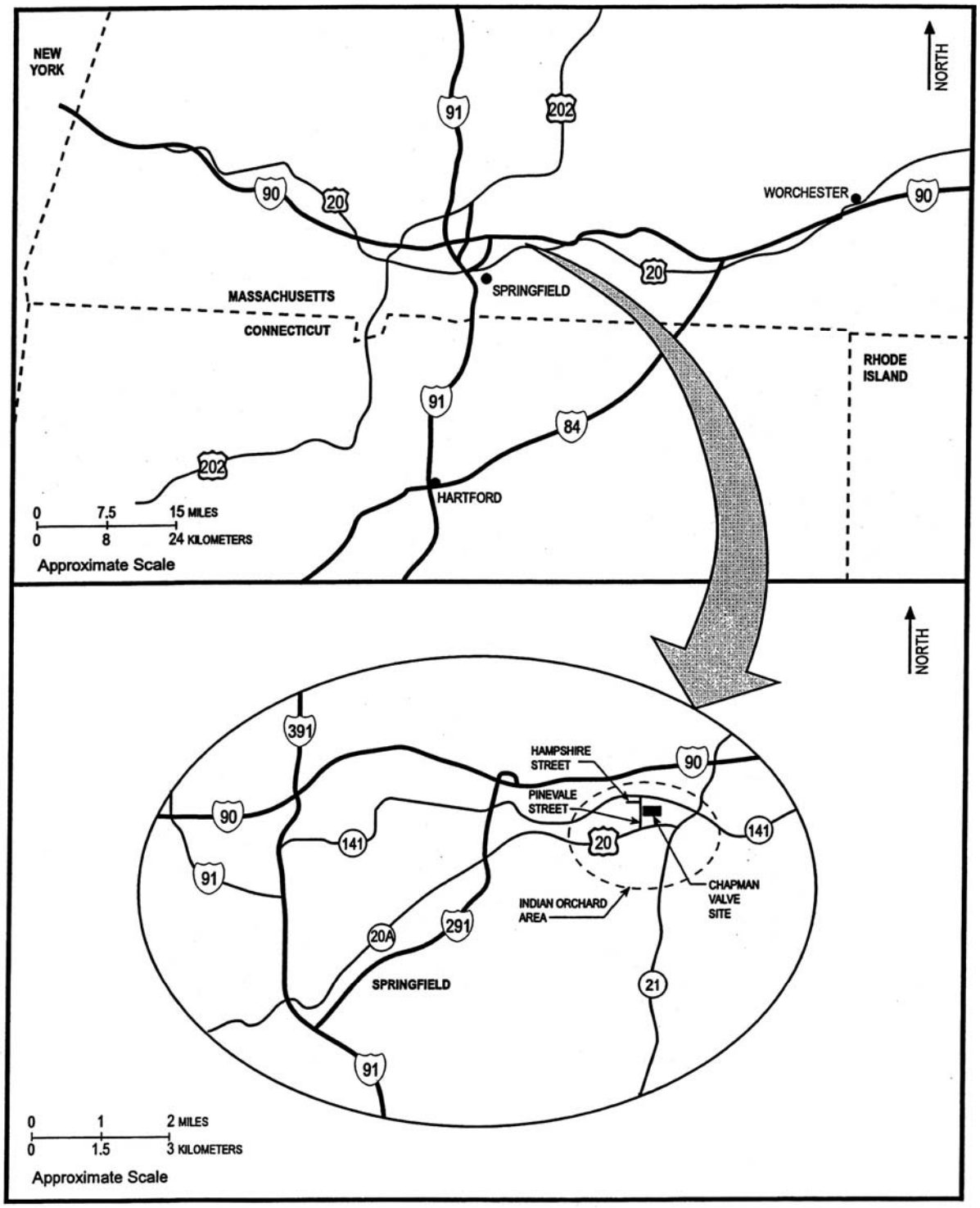
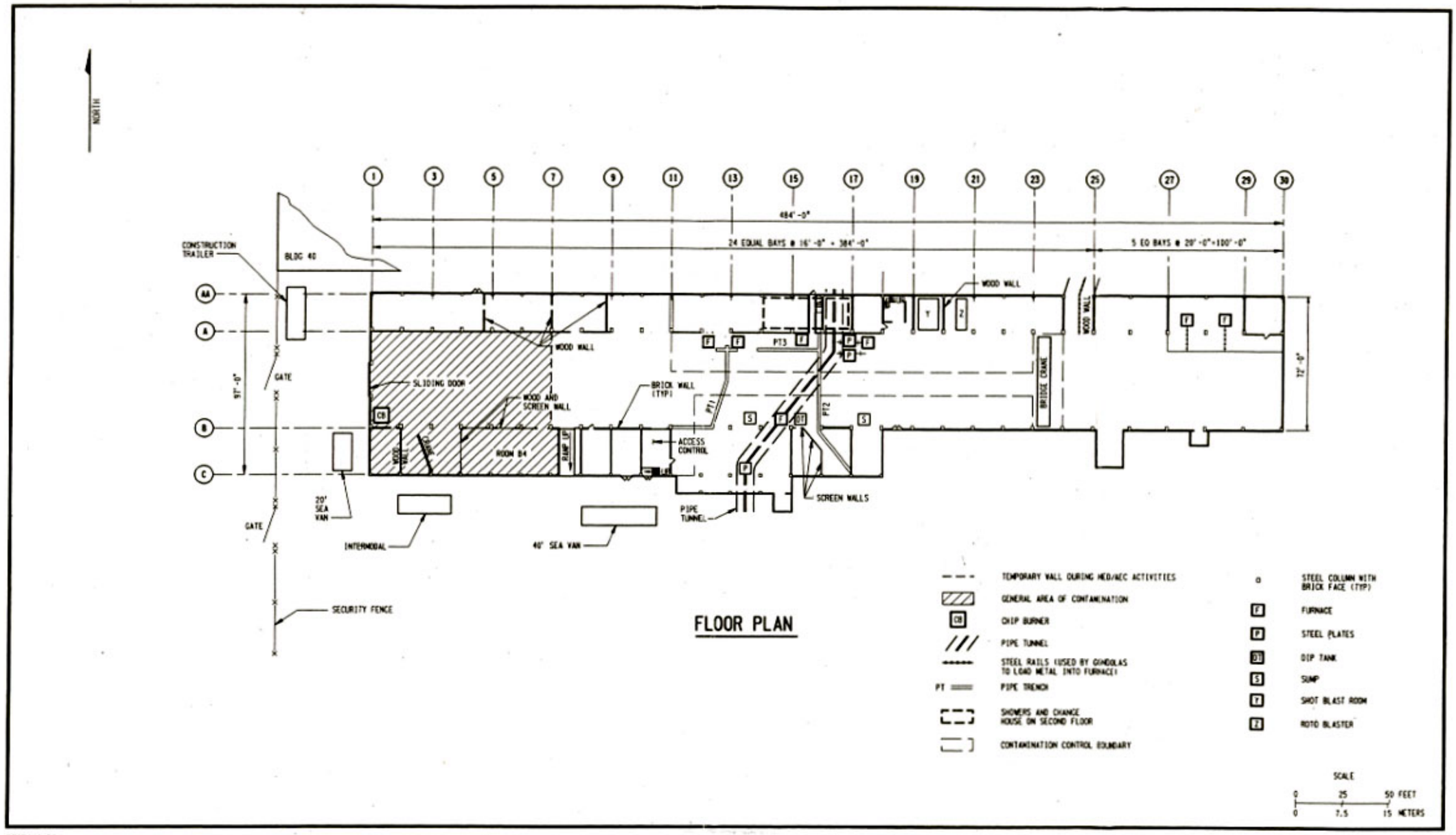


Figure I-1
Approximate Location of Chapman Valve Site



R76F009.DGN

Figure I-2
Chapman Valve
Building 23 - Plan View

Except for the roof and three small areas on the exterior (west and south) walls, the 1994 survey of the exterior of Building 23 indicated contamination below both the direct and removable criteria. It was determined that the contamination on the exterior walls and roof contributed only a negligible dose to potential future workers and therefore decontamination of these surfaces was not necessary (Exh. II, Ref. 17).

The exterior contamination originated from a chip burner in the southwest corner of the building that exhausted to the atmosphere through a nearby window. The chip burner was used for disposal (by burning) of uranium chips, turnings, scrap, and debris. The exhaust location and the shape of the roof caused more contamination to be deposited on the south roof than on the north roof (verified by characterization measurements). The uranium storage area was located in room B4 on the south side of the building. Characterization results did not indicate subsurface soil contamination beneath the floor slab.

Chemical sampling revealed the presence of radioactively contaminated lead-containing paint on the cranes, electrical boxes, and structural steel. Composite roof samples indicated the presence of bulk asbestos and electrical box insulation.

2.2 REMEDIAL ACTION GUIDELINES

Radioactive contamination at CHV consisted primarily of natural uranium. DOE residual contamination guidelines for release of formerly contaminated properties for use without radiological restrictions (Table I-1) were applied to the crane, floor, and drain lines. These guidelines were adopted by DOE based on compatibility with EPA criteria for remedial action in 40 CFR 192 and are contained in DOE Order 5400.5, "Radiation Protection of the Public and the Environment" (Exh. II, Ref. 7).

Supplemental guidelines were developed based on information contained in a technical study and preliminary hazard assessment (Exh. II, Ref. 11, 12). The following supplemental limits were approved by DOE in a memorandum that also outlined the remediation approach (Exh. II, Ref. 10):

- Remove all detectable contamination at ground level to the surface limits established in DOE Order 5400.5, Chapter IV.
- Perform gross decontamination of all horizontal surfaces above the ground level.
- Perform decontamination on all horizontal truss surfaces to meet an average surface level activity of no more than 15,000 dpm per 100 cm^2 of uranium activity for the entire truss.

Table I-1
Summary of DOE Residual Contamination Guidelines

Basic Dose Limits
The basic limit for the annual radiation dose received by an individual member of the general public is 100 mrem/y[a]. In implementing this limit, DOE applies as-low-as-reasonably-achievable principles to set site-specific guidelines.

Soil Guidelines [b,c,d]

Radium-226 Radium-228 Thorium-230 Thorium-232	5 pCi/ when averaged over the first 15 cm of soil below the surface and over any contiguous 100 m^2 surface area; 15 pCi/g when averaged over any 15-cm-thick soil layer below the surface layer and over any contiguous 100 m^2 surface area.
Uranium[c]	100 pCi/g for total uranium and 50 pCi/g for uranium-238 when averaged over any 15-cm-thick soil layer and over any contiguous 100 m2 surface area.

Structure Guidelines
Airborne Radon Decay Products: Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property that has no radiological restrictions on its use; structures that will be demolished or buried are excluded. The applicable generic guideline (40 CFR 192) is: In any occupied or habitable building, the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 Working Level [f] (WL). In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions are not required to comply with this guideline when there is reasonable assurance that residual radioactive materials are not the cause.
External Gamma Radiation: The average level of gamma radiation inside a building or habitable structure on a site that has no radiological restrictions on its use shall not exceed the background level by more than 20 μtR/h and will comply with the basic dose limits when an appropriate-use scenario is considered.

Indoor/Outdoor Structure Surface Contamination

Radionuclide [h]	**Allowable Surface Residual Contamination** [g] (dpm/100 cm^2)		
	Average[i,j]	Maximum[j,k]	Removable[j,l]
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, 1-124, 1-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, 1-126,1-131,1-133	1,000	3,000	200
U-Natural, U-235, U-238. and associated decay products	5,000 α	15,000 α	1,000 α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission except Sr-90 and others noted above)	5,000 β-γ	15,000 β-γ	1,000 β-γ

Notes:

[a] Department of Energy, 1990, Order 5400.5, "Radiation Protection of the Public and the Environment" (February 8).

[b] These guidelines take into account in growth of radium-226 from thorium-230 and of radium-228 from thorium-232, and assume secular equilibrium. If either thorium-230 and radium-226 or thorium-232 and radium-228 are both present, not in secular equilibrium, the guidelines apply to the higher concentration. If other mixtures of radionuclides occur, the concentrations of individual radionuclides must be reduced so that (1) the dose for the mixtures will not exceed the basic dose limit, or (2) the sum of ratios of the soil concentration of each radionuclide to the allowable limit for the radionuclide will not exceed 1 ("unity").

[c] These guidelines represent allowable residual concentration exceeding background levels averaged across any 15-cm (6-in.)-thick layer to any depth and over any contiguous 100-m^2 (1,076-ft^2) surface area, except as noted.

d If the average concentration in any surface or below-surface area less than or equal to 25 m^2 (269 ft^2) exceeds the authorized limit or guideline by a factor of $(100/A)^{1/2}$, where A is the area of the elevated region in square meters, limits for "hot spots" will also be applicable. Procedures for calculating these hot spot limits, which depend on the extent of the elevated local concentrations, are given in the supplement. In addition, every reasonable effort shall be made to remove any source of radionuclide that exceeds 30 times the appropriate limit for soil, irrespective of the average concentration in the soil.

[e] Guidelines are calculated on a site-specific basis using a DOE manual developed for this use.

[f] A working level (WL) is any combination of short lived radon decay products in 1 liter of air that will result in the emission of 1.3×10^5 MeV of potential alpha energy.

[g] 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 measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[h] Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

[i] Measurements of average contamination should not be averaged over more than 1 m^2 (10.8 ft^2). For objects of less surface area, the average must be derived for each such object.

[j] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm (0.4 in.).

[k] The maximum contamination level applies to an area of not more than 100 cm^2 (16 in $.^2$).

[1] The amount of removable radioactive material per 100 cm^2 (16 in $.^2$) of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 (16 in $.^2$) is determined, the activity per unit area should be based on the actual area or the entire surface should be wiped. The numbers in this column are maximum amounts.

Source:

DOE Order 5400.5 and 40 CFR 192

2.3 POST-REMEDIAL ACTION STATUS

Analytical results from post-remedial action surveys (Exh. II, Ref 17) indicated that levels of radioactivity in the remediated areas meet applicable cleanup guidelines and approved supplemental limits in effect at the conclusion of remedial action. Data supporting the remedial activities performed at the site were reviewed by the IVC to independently verify the adequacy of the remedial action and to confirm that the site is in compliance with applicable remedial action guidelines. The IVC also provided independent verification by collecting post-remedial action samples for independent radiological analysis and by conducting radiological walkover surveys. The IVC reviewed results of post-remedial action surveys and sampling and verified that the remediated areas comply with applicable cleanup guidelines for release of the site without radiological restrictions (Exh. II, Refs. 19,21).

3.0 SUMMARY OF REMEDIAL ACTION

The following discussion briefly describes the remedial action process and the measures taken to protect the public and the environment during this process.

3.1 PRE-REMEDIAL ACTION ACTIVITIES

Immediately before remedial action began, the contaminated areas were surveyed to accurately define the boundaries of radioactive contamination and to supplement existing characterization information. Any areas that were previously inaccessible were surveyed as they became accessible during remedial action.

3.2 DECONTAMINATION ACTIVITIES

Remedial activities at CHV lasted approximately eight weeks, from July to September 1995. All decontamination efforts were confined to the interior of Building 23. The following sections contain descriptions of decontamination techniques for each area remediated. Techniques used in the remedial action are summarized in Table I-2.

3.2.1 10-Ton Bridge Crane

Residual surface contamination was removed from the bridge crane by hand scraping and vacuuming. The crane was decontaminated without disturbing painted surfaces except for several small areas where it was necessary to remove the paint to decontaminate the surface. Before the paint was removed, the high-efficiency particulate air (HEPA) vacuums were emptied to

minimize the amount of material intermixed with the lead-based paint. Very-small-quantity-generator status permitted treatment and disposal of the paint onsite.

Table I-2
Decontamination Techniques Used at the Chapman Valve Site

Type	Description
HEPA Vacuuming	High-efficiency particulate air- (HEPA-) filtered vacuum cleaners were used to remove loose contamination and dust. HEPA vacuums were also used in conjunction with other techniques (grinding, wire brushing, etc.) to eliminate the airborne contamination associated with these techniques.
Jackhammering	Conventional jackhammers were used on small areas to remove anchor bolts from the concrete slab. Skid steer loaders equipped with hoe-ram attachments were used to remove the wooden blocks from the floor and to break up the concrete pads to expose the base slab.
Excavation	Contaminated concrete and debris were removed from the building with a skid steer loader. Contaminated soil from the west ramp and the pipe excavation was removed with shovels.
Wire Brushing	Small areas on the overhead trusses requiring rework were wire-brushed to remove contamination.
Scraping	Contamination was scraped from the surface of the 10-ton crane and the wooden planks removed from the crane deck.

The paint was dissolved in an acid solution and mixed with concrete. A survey of the concrete monoliths containing the paint solution showed that contaminant levels were below DOE residual surface contamination guidelines listed in Table I-1. The monoliths were left onsite, with the concurrence of the property owner and the State of Massachusetts (Exh. II, Ref. 15). The wooden decking of the crane was removed and decontaminated using the same technique that was used on the crane. The decontaminated boards were also left onsite.

3.2.2 Overhead Trusses

HEPA vacuums were used to remove contaminated dust from the upper horizontal surfaces of trusses in the main bay of Building 23 between trusses 1 and 8 (Figure I-2). In one area of truss 2, abrasive decontamination was required to meet the supplemental limits set for CHV. The radioactively contaminated dust generated during the operation was stabilized in concrete to eliminate the potential for fugitive dust during the transport and handling of the waste. The concrete/dust was then placed in an intermodal container for shipment to a licensed disposal facility.

3.2.3 Horizontal Interior Wall Surfaces

Horizontal surfaces on the walls of the main bay between columns 1 and 8, including the crane rail, window sills, radiators, and a large pipe on the north wall, were decontaminated with a HEPA vacuum (Figure I-2). Preliminary hazard assessment calculations indicated that the contamination on the wall would not contribute significantly to the dose to a future demolition worker, but the horizontal surfaces were decontaminated as part of FUSRAP's objective of keeping exposure to radioactivity as low as reasonably achievable (ALARA). The radioactive dust generated during the operation was solidified in concrete to eliminate the potential for fugitive dust during the transport and handling of the waste. The concrete/dust was then placed in an intermodal container for shipment to a licensed disposal facility.

3.2.4 Floor

A backhoe and skid steer loader equipped with a hoe-ram attachment were used to remove the wooden block and concrete pads above the base slab in the main bay and the B-series rooms on the south side of the building from column 1 to column 8 (Figure I-2). The wooden blocks and concrete pads were considered clean material based on volumetric sampling results from characterization (Exh. II, Ref. 12) and were left onsite, with the concurrence of the property owner. The floor was vacuumed with HEPA vacuums and the dust collected. Self-tapping steel floor anchors were removed with a jackhammer and surveyed. Any floor anchors containing above-background contamination were placed in an intermodal container for shipment to a licensed disposal facility. Areas of the floor that required further decontamination were mechanically cleaned with side grinders equipped with vacuum shrouds connected to HEPA vacuums.

During the survey of the floor, it was determined that part of the ramp at the west equipment door of Building 23 needed to be removed to provide access to contamination under the ramp. The portions of the ramp requiring removal were broken up with a jackhammer and a

skid steer loader equipped with a hoe-ram attachment. Shovels were then used to remove the contaminated soil below the ramp. All radioactively contaminated dust collected during the decontamination of the floor was mixed with concrete to eliminate the potential for fugitive dust during the transport and handling of waste. The concrete/dust was then placed in an intermodal container for shipment to a licensed disposal facility.

3.2.5 Drain Line

A ductile iron drain line discovered in Room B4 after removal of the wooden blocks was determined to be radioactively contaminated above criteria. The drain line was removed by breaking the concrete with jackhammers and a skid steer loader equipped with a hoe-ram attachment. Lead seals in the joints of the pipe were segregated by breaking the pipe away from the seal. A total of 145 ft of 4-in. drain line was removed and placed in the intermodal container for shipment to the disposal facility. The lead seals were surveyed, released, and taken to a local lead recycling company.

3.3 VOLUME REDUCTION

Volume reduction, waste minimization, and cost-saving techniques employed during the remedial action included segregation, sampling, and surveying of the wastes. Specific examples of waste minimization and cost-saving measures at CHV are described below:

- Concrete and debris were surveyed to determine whether contaminant levels were above criteria. The material below criteria was placed inside the building and left onsite, with property owner concurrence.

- Paint removed from the crane was treated by acid dissolution and solidified with concrete to reduce the content of leachable lead to the extent that toxicity characteristic leaching procedure (TCLP) results were below the limits specified by RCRA. The levels of lead in the resulting concrete monoliths were determined to be below cleanup guidelines, and the monoliths were left onsite, with concurrence of the property owner and the State of Massachusetts (Exh. II, Ref 15).

- Remediation workers wore washable Protech-2000 coveralls rather than standard Tyvek protective clothing. Using the washable coveralls reduced waste volumes associated with disposable protective clothing. An additional advantage was that workers found the Protech-2000 fabric coveralls more comfortable to wear than

Tyveks. Interstate Nuclear Services was contracted to provide laundry service for the washable protective clothing.

- An Eberline PCM-2 personnel contamination monitor was used at the access control point instead of a standard manual frisk, reducing the time required to ensure that workers exiting the controlled area are not contaminated. The PCM-2 monitor completes the contamination check in one minute; a standard hand frisk takes approximately five minutes. The PCM-2 also ensures a more consistent exit survey than the conventional hand frisk performed by an individual.

3.4 POST-REMEDIAL ACTION MEASUREMENTS

After each portion of the property was decontaminated, a radiological survey of the area was conducted to confirm that all radioactive contamination above cleanup criteria had been removed. Initial post-remedial action surveys were conducted by the radiological support subcontractor (RSS), Thermo Nutech Services, on behalf of the project management contractor, BNI. Figures delineating the surveyed areas and tables containing complete survey results are included in the post-remedial action report for CHV (Exh. II, Ref 17). Techniques used during post-remediation and verification surveys included direct surface and transferable contamination measurements, external gamma exposure rate measurements, and soil sampling. The initial post-remediation surveys were conducted in accordance with RSS procedures and BNI instruction guides. The IVC, Oak Ridge National Laboratory (ORNL), performed independent verification surveys of the floor, trusses, and bridge crane using survey techniques that were similar or identical to those used by the RSS. IVC survey data for CHV are presented in a separate verification report (Exh. II, Ref 19).

As excavations were completed, walkover surveys were conducted to determine whether soil had been removed to levels meeting cleanup criteria. Final walkover surveys were performed with a field instrument for detecting low-energy radiation (FIDLER) and a Geiger-Mueller counter. The walkover surveys provided immediate feedback so that additional excavation could be performed if residual contamination exceeded remedial action guidelines and the objective of maintaining exposures ALARA was not met.

External gamma radiation exposure rates were measured at 26 locations. Measurements were taken at 1 m (3 ft) above the ground surface with a pressurized ionization chamber (PIC). Direct-contact beta-gamma measurements were obtained with Geiger-Mueller counters; direct-contact alpha measurements were obtained with alpha scintillation detectors. Direct measurements were obtained by placing the probe on the surface to be surveyed and allowing counts to accumulate for at least 30 seconds on a scaler that was attached to the probes.

Measurements were converted, using appropriate calibration and conversion factors, to dpm/ 100 cm2 and compared with the DOE guidelines.

Transferable alpha and beta-gamma contamination was determined by wiping a 100-cm^2 (15.5 -$in.^2$) area with a smear and measuring alpha emissions from the smear with an alpha scintillation counter and a Geiger-Mueller counter, respectively. Transferable contamination was measured, at a minimum, at any location that exhibited direct alpha or beta-gamma contamination above the guideline for removable contamination (1,000 dpm/100 cm^2). Post-remedial action survey data for each area that was decontaminated are discussed in the following sections.

3.4.1 10-Ton Bridge Crane

Post-remedial action activities for the crane included surveys of direct and transferable contamination and sampling of paint and dust residues to determine whether the waste exhibited RCRA characteristics. A total of 169 locations were surveyed. Direct alpha and beta-gamma average surface contamination readings were 44 dpm/100 cm^2 and 520 dpm/100 cm^2, respectively. Average removable alpha and beta-gamma readings were 3 dpm/100 cm^2 and 42 dpm/100 cm^2, respectively. These readings indicate that the crane was decontaminated to levels that were in compliance with DOE residual removable contamination guidelines listed in Table I-1.

3.4.2 Overhead Trusses

Surveys of fixed and removable contamination were conducted on the trusses to determine the effectiveness of the decontamination effort. The maximum fixed average per truss for alpha contamination was 2,114 dpm/100 cm^2; for beta-gamma contamination, the average was 12,261 dpm/100 cm^2. Both of these readings, which were from truss 2, were below the supplemental limit of 15,000 dpm/100 cm^2 average per truss. All measurements of removable contamination were less than 1,000 dpm/100 cm^2. An additional survey conducted on non-horizontal surfaces and between welded angles of the trusses confirmed that the supplemental limits had not been exceeded and that these areas did not require decontamination.

3.4.3 Horizontal Interior Wall Surfaces

The ALARA approach to decontamination of the horizontal wall surfaces was to remove contamination that exhibited a potential for resuspension or migration. The maximum and average removable alpha measurements for the west wall were 164 dpm/100 cm^2 and 17 dpm/100 cm^2; measurements for the north wall were 8 50 dpm/100 cm^2 and 3 5 dpm/100 cm^2; and measurements for the south wall were 111 dpm/100 cm^2 and 17 dpm/100 cm,^{2}. The maximum and average removable beta-gamma measurements for the west wall were 191 dpm/100 cm^2 and 70 dpm/100 cm^2; measurements for the north wall were 3,197 dpm/100 cm^2 and 58 dpm/100 cm^2; and measurements for the south wall were 319 dpm/100 cm^2 and 45 dpm/100 cm^2. Except for a single reading on the north wall, all measurements were below the criterion of 1,000 dpm/100 cm^2 for removable contamination.

3.4.4 Floor

All survey results for the floor were below DOE 5400.5 criteria. The average fixed beta-gamma reading was 914 dpm/100 cm^2; the minimum was -872 dpm/100 cm^2; and the maximum was 4,934 dpm/100 cm^2. (Negative numbers indicate that the measurement was less than the minimum detectable activity and that after background was subtracted, the numerical value was negative.) A soil sample was also collected in the area where part of the west equipment door ramp was removed. Analytical results were 2.0 pCi/g for uranium-238; 0.47 pCi/g for radium-226; and 0.41 pCi/g for thorium-232. A site-specific uranium guideline for CHV was not developed; however, the level detected was well below the guidelines of 35 to 50 pCi/g typically derived for FUSRAP sites. Levels of radium-226 and thorium-232 were below the DOE 5400.5 criterion of 5 pCi/g for surface soils.

3.4.5 Drain Line

Following removal of 145 ft of 4-in. drain line in room B-4, three composite soil samples were collected in the trench to determine whether the decontamination effort was successful. Maximum levels detected were 0.62 pCi/g for uranium-238; 0.47 pCi/g for radium-226; and 0.50 pCi/g for thorium-232. A site-specific uranium guideline for CHV was not determined; however the results are well below the guideline of 35 to 50 pCi/g typically used for FUSRAP sites. The contaminated drain piping was placed in an intermodal container and shipped to a licensed disposal facility.

3.5 VERIFICATION ACTIVITIES

After remedial activities were completed, the IVC conducted a survey to verify that the site had been remediated to levels meeting cleanup criteria. The objective of the independent verification survey was to confirm that surveys, sampling, and analysis conducted during the remedial action process provided an accurate and complete description of the radiological status of the property.

IVC activities included reviewing the final radiological survey and soil data, conducting a visual inspection of the site, and performing radiological surveys and sampling. When verification activities were completed, the IVC prepared a verification report and submitted it to DOE (Exh. II, Ref. 19).

Due to the demolition and removal of Building 23, a second verification survey was performed to verify that the demolition activities did not redistribute any of the structural contamination. This survey was performed in May 2003. After completion of survey activities, the IVC prepared a verification report and submitted it to DOE (Exh. II, Ref. 21).

3.6 PUBLIC AND OCCUPATIONAL EXPOSURE

In accordance with the site-specific safety and health work instruction for CHV, engineering and administrative controls (such as dust control and hazardous work permits) and personal protective equipment (PPE) were used during remedial action to protect remediation workers and members of the public from potential exposure to radiation in excess of applicable standards.

All personnel working in contaminated areas were required to wear protective coveralls, hard hats, safety glasses, hearing protection, boots, and gloves. If conditions warranted,

additional PPE (e.g., face shields) was used. Site conditions did not necessitate the use of worker respiratory protection except in the HEPA vacuum changeout tent.

Workers leaving radioactively restricted work areas were scanned at the control point by a personnel contamination monitor followed by a hand and foot frisk. This ensured that protective clothing was not contaminated and prevented the spread of contamination to clean areas.

Potential exposure pathways for workers and members of the public were inhalation and ingestion of radioactively contaminated airborne dust generated during mechanical decontamination and excavation. HEPA filtration units were used to control the spread of dust and minimize the potential for contaminants to become airborne. In addition, a fine water mist was sprayed to control dust during removal of the floor and during transport of material to intermodal containers. All equipment used in the controlled area was surveyed before being released from the site.

Particulate air monitoring devices were placed in the areas being remediated at monitoring locations selected to provide data for the worst-case scenario. Uranium-238 concentrations ranging from 1.5×10^{-14} to 5.8×10^{-13} μCi/mL were conservatively derived by collecting air particulate samples daily from lapel air samplers worn by workers. After determination of gross activity per unit volume of air passed through the filter, uranium-238 was assumed to be the source of all activity on the filter. The derived air concentrations were then compared with the applicable DOE guideline of 2.0 x 10-11 μCi/mL for occupational exposures to airborne uranium-238 (DOE Order 5480.11, Radiation Protection for Occupational Workers, July 1989).

Area air particulate sampling was also performed adjacent to areas being remediated to ensure that no member of the general public was exposed to radioactivity above guidelines. Because all remediation took place within Building 23 and there were no open vents in the building, it was determined that area air particulate monitoring was not required for the exterior of the building. An Eberline RAS-1 high-volume monitor was used and collected daily and counted after four days to allow for radon decay. The limits specified in DOE Order 5400.5 are derived concentration guides (DCGs); a DCG is the concentration of a particular radionuclide that would provide an effective dose equivalent of 100 mrem/y (the DOE basic dose limit) to an individual continuously exposed to the radionuclide by one pathway for an entire year. Uranium-238 concentrations measured by area particulate monitors ranged from 1.5 x 10-15 to 3.1 x 1013 μCi/mL. The DCG for uranium-238 is 2.0×10^{-12} μCi/mL.

3.7 COSTS

The final costs associated with the remedial action performed at the subject property are presented in Table 1-3.

Table I-3
Cost of Remedial Action at the Chapman Valve Site

Description	Amount
Direct Costs	$269,000
Radiological Laboratory/HP Support	184,000
Chemical Laboratory	11,000
Direct Hire Labor	273,000
Transportation	12,000
Disposal	24,000
Final Engineering Reports	51,000
Home Office Support	143,500
Total RA Costs	$968,000

4.0 CONCLUSION

Post-remedial action surveys and samples demonstrated, and DOE certified, that the locations remediated are in compliance with applicable radiological standards and criteria in effect at the conclusion of the remedial activity for protecting members of the general public as well as occupants of the site and the environment.

EXHIBIT II:

DOCUMENTS SUPPORTING THE CERTIFICATION OF THE REMEDIAL ACTION PERFORMED AT THE CHAPMAN VALVE SITE

1.0 INTRODUCTION

The material in Exhibit 11 consists of documents supporting DOE certification that conditions at the subject property are in compliance with radiological guidelines and standards in effect at the conclusion of remedial action. It consists of the letters, memos, and reports that were produced to document the entire remedial action process, from designation of the property for cleanup under FUSRAP to the certification that no radiological restrictions limit the future use of the site. Exhibit I of this docket should be consulted for a brief history of the origin of site contamination, radiological characterization, the remedial action performed, and post-remedial action verification activities.

2.0 FUSRAP GENERAL

1. DOE, "Description of Formerly Utilized Sites Remedial Action Program," ORO-777, Oak Ridge, Tenn., September 1980.

2. DOE, *FUSRAP Summary Protocol*, CCN[1] 35692 and 046176, Oak Ridge, Tenn., March 1986.

3. DOE, "FUSRAP Designation/Elimination Protocol," Supplement No. l to the FUSRAP Summary Protocol, CCN 35692, Oak Ridge, Tenn., January 1986.

4. DOE, "FUSRAP Verification and Certification Protocol," Supplement No. 2 to the FUSRAP Summary Protocol, Rev. l, CCN 35692, November 1985.

3.0 DESIGNATION DOCUMENTATION

5. Memorandum, W. A. Williams (DOE-HQ) to File, "Authority Determination-Former Chapman Valve Manufacturing Company Facility, Indian Orchard, Massachusetts," CCN 098808, December 15, 1992.

6. Memorandum, James W. Wagoner II (DOE-HQ) to L. Price (DOE-OR), "Authorization for Remedial Action at the Chapman Valve Manufacturing Company Facility, Indian Orchard, Massachusetts," CCN 098808, December 15;1992 [Attachment: "Designation Summary for Chapman Valve Manufacturing, Indian Orchard, Massachusetts"].

[1] Communications Control Registry (CCN).

4.0 DECONTAMINATION CRITERIA

7. DOE Order 5400.5, Radiation Protection of the Public and the Environment, Chapter IV, "Residual Radioactive Material," CNN 067327, Office of Environment, Safety, and Health, Washington, D.C., February 1990.

8. DOE, "Design Criteria for Formerly Utilized Sites Remedial Action Program (FUSRAP) and Surplus Facilities Management Program (SFMP)," 14501-00-DC-01, Rev. 2, Oak Ridge, Tenn., March 1986.

9. Memorandum from J. W. Wagoner (DOE-HQ) to L. Price (DOE-FSRD), "Supplemental Limits for Residual Uranium at the Chapman Valve Site, Indian Orchard, Massachusetts," CCN 132753, July 27, 1995.

10. Memorandum from A.S. Johnson (DOE-HQ) to L. Price (DOE-FSRD), "Supplemental Limits for Residual Uranium at the Chapman Valve Site," CCN 144918, July 31, 1996 [Attachment: Bechtel National Inc., (BNI), "Chapman Valve Site - Draft Technical Memorandum Post Remedial Action Report," CCN 143419, June 20,1996].

11. BNI, "Hazard Assessment for Chapman Valve, Rev. 2, Calculation Number 133-CV-001," December 19, 1995.

5.0 RADIOLOGICAL CHARACTERIZATION REPORTS

12. BNI, "Technical Study for the Remedial Action at the Chapman Valve Site, Indian Orchard, Massachusetts," CCN 132076 and 129507," Oak Ridge, Tenn., May 1995 [Attachment: BNI, "FUSRAP Technical Memorandum -- Chapman Valve Characterization Results," May 8,1995].

13. Oak Ridge National Laboratory (ORNL), "Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts," ORNLIRASA-9211, CCN 098808, Oak Ridge, Tenn., July 1992.

6.0 REGULATORY DOCUMENTATION

14. Memorandum, Robert W. Poe (Assistant Manager for Environment, Safety and Quality) to Robert D. Dempsey (Assistant Manager for Environmental Management, EW-90), "Categorical Exclusion (CX) Determination-Removal Action at the Former Chapman Valve Manufacturing Company," CCN 131479, June 19, 1995.

15. Massachusetts Department of Environmental Protection, "Very Small Quantity Generator of Hazardous Waste Registration," CCN 132040, June 30, 1995.

7.0 REAL ESTATE LICENSES

16. Letter from Katy Kates (DOE Realty Officer) to Anthony D. Pantaleoni (Vice President, Crane Company), "Real Estate License REORDOER-7-95-0139, Chapman Valve Mfg. Co., Indian Orchard, MA," CCN 130873, June 6, 1995.

8.0 POST-REMEDIAL ACTION REPORTS

17. Bechtel National Inc., "Post-Remedial Action Report for the Chapman Valve Site, Indian Orchard, Massachusetts," DOE/OR/21949-408, CCN 148447, Oak Ridge, Tenn., November, 1996.

9.0 VERIFICATION DOCUMENTATION

18. Letter from R. E. Rodriguez (ORNL) to Dr. W. Alexander Williams (DOE-HQ), "Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts," CCN 134984, October 2, 1995.

19. Oak Ridge National Laboratory, "Results of the Independent Radiological Verification Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts," ORNL/RASA-95/17; CCN 133-LOA-TAA-0001, May 1997.

20. Memorandum from L. Price (DOE-FSRD) to J.W. Wagoner (DOE-HQ), "Chapman Valve - Remediation Success Definition," CCN 133219, June 9, 1995.

21. Letter from M. Murray (ORNL) to Tomiann McDaniel (USACE), "Final Radiological Verification Survey at the Chapman Valve Site, Indian Orchard, Massachusetts," 191-LOA-GET00060, July l, 2003

10.0 REGULATOR COMMENTS ON REMEDIAL ACTION

22. Letter from J. Kopotic (DOE-FSRD) to S. Ball (Massachusetts Department of Environmental Protection) and T. 0' Connell (Massachusetts Department of Public Health and Radiation Control Program), "Chapman Valve Site - Transmittal of Data and Remedial Action Approach" CCN 130172 and 131182, May 22, 1995 [See Reference No. 11].

23. Letter from A. Weinberg [Massachusetts Department of Environmental Protection (MDEP)] to A. Pantaleoni (Crane Co.), "Concurrence with DOE Remedial Action Plan," CCN 131136, June 15, 1995.

24. Letter from T. F. O'Connell (Massachusetts Department of Health) to J. Kopotic (DOE-FSRD), "Concurrence with DOE Remedial Action Plan," CCN 131136, June 14, 1995.

EXHIBIT III:

FEDERAL REGISTER DOCUMENTS

1.0 DOE STATEMENT OF CERTIFICATION

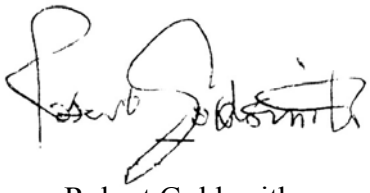
This section contains the statement of certification issued by the Department of Energy that the subject property is in compliance with radiological guidelines in effect at the conclusion of remedial action.

Statement of Certification: Chapman Valve Site
in Indian Orchard, Massachusetts

The DOE, the Oak Ridge Operations Office, the Office of Environmental Management, the Oak Ridge Reservation, the Remediation Management Group, and the U.S. DOE Office of Environmental Management (EM), Core Technical Group (EM-23), has reviewed and analyzed the radiological data obtained following remedial action at the Chapman Waive site in Indian Orchard, Massachusetts, (Deed Book 2891, Page 53, in the records of Hampden County, Massachusetts). Based on the analysis of all data collected, including post-remedial action surveys, DOE certifies that any residual contamination remaining onsite at the time remedial actions were completed falls within DOE radiological decontamination criteria and standards for use of the property without radiological restrictions. This certification of compliance provides assurance that reasonably foreseeable future use of the site will result in no radiological exposure above DOE radiological criteria and standards for protecting members of the general public and occupants of the property.

Property owned by: The Crane Company, 100 First Stamford Place, Stamford, Connecticut 06902.

Issued in Germantown, Maryland, on January 14, 2004.



Robert Goldsmith
Director, Core Technical Group
Environmental Cleanup and Acceleration
Office of Environmental Management

2.0 FEDERAL REGISTER NOTICE OF CERTIFICATION

This section contains a copy of the published Federal Register notice announcing the completion of remedial action, with accompanying statement of certification issued by the Department of Energy.

Advanced Scientific Computing Research for closely related research in computer science and/or applied mathematics. Applications may request support for up to three years, with out-year support contingent on the availability of funds and satisfactory progress. To support multi-disciplinary, multi-institutional efforts, annual funding levels of up to $1 million may be requested for the scientific application work and up to $200,000 per year for the Scientific Application Partnership Program work.

As required by the SC grant application guide, applicants must submit their budgets using the Budget Page (DOE Form 4620.1) with one Budget Page for each year of requested funding. The requested funding for the proposed work in computer science and applied mathematics should be included on a separate Budget Page. However, applicants are also requested to list the proposed computer science and applied mathematics costs separately in an appendix, as the Office of Advanced Scientific Computing Research may support this part of the work (up to about 20 percent of the total project cost). The Office of Fusion Energy Sciences expects to fund two or three centers, depending on the size of the awards.

Applications

Applications will be subjected to scientific merit review (peer review) and will be evaluated against the following criteria listed in descending order of importance as codified in 10 CFR part 605.10(4) *(http://www.science.doe.govl productionlgrantsl605index.html):*

1. Scientific and/or technical merit of the project;
2. Appropriateness of the proposed method or approach;
3. Competency of the applicant's personnel and adequacy of the proposed resources; and
4. Reasonableness and appropriateness of the proposed budget. The evaluation under the first criterion in 10 CFR part 605.10(4), Scientific and Technical Merit, will pay particular attention to:

(a) The importance of the proposed project to the mission of the Office of Fusion Energy Sciences;

(b) The potential of the proposed project to advance the state-of-the-art in computational modeling and simulation of plasma behavior; and

(c) The need for extraordinary computing resources to address problems of critical scientific importance to the fusion program and the demonstrated abilities of the applicants to use terascale computers.

The evaluation under item 2, Appropriateness of the Proposed Method or Approach, will also consider the following elements related to quality of planning and management:

(a) If the project involves more than one scientific code, how the use of multiple codes will contribute to a coherent set of scientific objectives that are more readily achieved through the use of multiple codes;

(b) Soundness of the plan for effective management of the project;

(c) Quality of plan for ensuring communication with math and computer science projects and with other relevant SciDAC projects;

(d) Viability of plan for verifying and validating the models developed, including close coupling with experiments for ultimate validation; and

(e) Quality and clarity of proposed work schedule and deliverables.

Note that external peer reviewers are selected with regard to both their scientific expertise and the absence of conflict-of-interest issues. Non-federal reviewers may be used, and submission of an application constitutes agreement that this is acceptable to the investigator(s) and the submitting institution.

General information about development and submission of applications, eligibility, limitations, evaluations and selection processes, and other policies and procedures may be found in the Application Guide for the Office of Science (SC) Financial Assistance Program and in 10 CFR part 605. Electronic access to SC's Financial Assistance Guide and required forms is made available via the Internet using the following Web site address: *http:// www.science. doe.gov/production/grants/grants.html.*

In addition, for this notice, project descriptions must be 25 pages or less, including tables and figures, but excluding attachments. The application must also contain an abstract or project summary on a separate page with the name of the principal investigator, mailing address, phone, FAX, and email listed. The application must also include letters of commitment from all non-funded collaborators (briefly describing the intended contribution of each to the research), and short curriculum vitae for the principal investigator and any co-PIs.

The Catalog of Federal Domestic Assistance Number for this program is 81.049, and the solicitation control number is ERFAP 10 CFR art 605.

Issued in Washington, DC on: January 14, 2004.

John A. Alleva,

Director. Grants Fr Contracts Division, Office of Science.

[FR Doc. 04-1201 Filed 1-20-04; 8:45 am]

BILLING CODE 6454-01-P

DEPARTMENT OF ENERGY

Certification of the Radiological Condition of the Chapman Valve in Indian Orchard, MA

AGENCY: U.S. Department of Energy.

ACTION: Notice of certification.

SUMMARY: The Department of Energy (DOE) has completed remedial actions to decontaminate the Chapman Valve site in Indian Orchard, Massachusetts. This property formerly was found to contain quantities of radioactive material from activities conducted for the Atomic Energy Commission's (AEC) Brookhaven National Laboratory (BNL) during the mid-1940s. Based on the analysis of all data collected, DOE has concluded that the property is in compliance with DOE radiological decontamination criteria and standards, and that no radiological restrictions on the use of the property are required

ADDRESSES: The certification docket is available at the following locations:

U.S. Department of Energy, Public Reading Room, Room 1E-190, Forrestal Building, 1000 Independence Avenue, SW., Washington, DC 20585;

U.S. Department of Energy, DOE Information Center, 475 Oak Ridge Turnpike, Oak Ridge, Tennessee 37831; Springfield Museum and Library, 220 State Street, Springfield, Massachusetts 01103.

FOR FURTHER INFORMATION, CONTACT: Donald Mackenzie, Health Physicist, U.S. Department of Energy, Core Technical Group, EM-23/Cloverleaf Building, 1000 Independence Avenue, SW., Washington, DC 20585-2040. Telephone Number: (301) 903-7426. Fax Number: (301) 903-2385.

SUPPLEMENTARY INFORMATION: The U.S. DOE, Oak Ridge Operations Office (OR), Office of Environmental Management, has conducted remedial action at the Chapman Valve site in Indian Orchard, Massachusetts, under the Formerly Utilized Sites Remedial Action Program (FUSRAP). The objective of the program is to identify and remediate, or otherwise control, sites where residual radioactive contamination remains from

activities carried out under contract to the Manhattan Engineer District (MED)/AEC during the early years of the nation's atomic energy program. In October 1997, the Energy and Water Appropriations Act, 1998 transferred responsibility for management of the FUSRAP program to the U.S. Army Corps of Engineers (U.S. ACE). Completion of the certification process was delayed pending preparation of a Memorandum of Understanding (MOU) between the DOE and the U.S. ACE with regard to completed, remediated sites such as the Chapman Valve property. The MOU between the U.S. DOE and the U.S. ACE regarding Program Administration and Execution of the FUSRAP program was signed by the parties in March 1999. Funding to proceed with the completion of DOE closure documentation for several FUSRAP sites, including the Chapman Valve site, was obtained from the U.S. ACE in late 2000. The closure documentation for these sites will document the cleanup and inform the public of their successful decontamination of radioactive contamination.

The Chapman Valve site was formerly owned and operated by the Chapman Valve Manufacturing Company. In 1948, the company set-aside approximately one-third of an area known as Department 40 in the western end of Building 23 for the machining of uranium rods for the AEC's BNL. Segregation of the area from other parts of the facility was achieved by installing a floor to ceiling wooden partition that was more than 50 feet high. Special modifications to the facility included building shields, quenching tanks, suction systems, cranes, and ductwork. Uranium operations were terminated on November 8, 1948. After the contract was completed, the company had in its possession over 27,000 pounds of metal scrap, oxides, and sweepings. This material was identified for removal several months after contract completion.

The Oak Ridge National Laboratory (ORNL) personnel indicated in a 1991 survey report that the residual uranium contamination found at the Chapman Valve site was typical of MEDIAEC operations. This survey indicated that the contamination was limited to the interior of the segregated area within Department 40 and included floors, walls, and overhead beams. Following a review of files, it was concluded there are no indications that work with uranium metal was conducted at the site after the AEC operations were terminated.

In November and December 1994, additional radiological surveys were performed to supplement and refine survey information. Characterization findings confirm the presence of contamination located predominantly in the western end of Building 23. In addition to confirming the ORNL survey results, these findings were in agreement with historical process information obtained during interviews conducted with a former Chapman Valve supervisor. Based on this characterization data, DOE conducted remedial action at the Chapman Valve site from July to September 1995.

Post-remedial action surveys conducted in 1995 have demonstrated, and the DOE has certified, that the subject property is in compliance with the DOE radiological decontamination criteria and standards in effect at the conclusion of remedial action. These standards are established to protect members of the general public and occupants of the site, and to ensure that reasonably foreseeable future use of the site will result in no radiological exposure above applicable guidelines. Accordingly, this property is released from the FUSRAP program. These findings are supported by the DOE's Certification Docket for the Remedial Action Performed at the Chapman Valve site in Indian Orchard, Massachusetts. The DOE makes no representation regarding the condition of the site as a result of activities conducted subsequent to DOE's post-remedial action surveys.

The Certification Docket will be available for review between 9 a.m. and 4 p.m., Monday through Friday (except Federal holidays), in the DOE Public Reading Room located in 1E-190 of the Forrestal Building, 1000 Independence Avenue, SW., Washington, DC. Copies of the Certification Docket will also be available in the DOE Public Reading Room, U.S. Department of Energy, Oak Ridge Operations Office, 200 Administration Road, Oak Ridge, Tennessee, and the Springfield Museum and Library, 200 State Street, Springfield, Massachusetts.

The DOE, through the Acting Office Director, Core Technical Group (EM-23), Deputy Assistant Secretary for Environmental Cleanup and Acceleration (EM-20), the Assistant Secretary for the Office Environmental Management (EM), has issued the following statement:

Statement of Certification: Chapman Valve Site in Indian Orchard, Massachusetts

The DOE, the Oak Ridge Operations Office, the Office of Environmental Management, the Oak Ridge Reservation, the Remediation Management Group, and the U.S. DOE Office of Environmental Management (EM), Core Technical Group (EM-23), has reviewed and analyzed the radiological data obtained following remedial action at the Chapman Valve site in Indian Orchard, Massachusetts, (Deed Book 2891, Page 53, in the records of Hampden County, Massachusetts). Based on the analysis of all data collected, including post- - remedial action surveys, DOE certifies that any residual contamination remaining onsite at the time remedial actions were completed falls within DOE radiological decontamination criteria and standards for use of the property without radiological restrictions. This certification of compliance provides assurance that reasonably foreseeable future use of the site will result in no radiological exposure above DOE radiological criteria and standards for protecting members of the general public and occupants of the property.

Property owned by: The Crane Company, 100 First Stamford Place, Stamford, Connecticut 06902.

Issued in Germantown, Maryland, on January 14, 2004.

Robert Goldsmith,

Director, Core Technical Group, Environmental Cleanup and Acceleration. Office of Environmental Management.

[FR Doc. 04-1203 Filed 1-20-04; 8:45 am]

BILLING CODE 6450-01-P

DEPARTMENT OF ENERGY

Energy Information Administration

Policy Statement; Disclosure Limitation Policy for Statistical Information Based on Petroleum Supply Reporting System Survey Data

AGENCY: Energy Information Administration (EIA), Department of Energy (DOE).

ACTION: Policy statement. Disclosure limitation policy for statistical information based on Petroleum Supply Reporting System survey data.

SUMMARY: The EIA is announcing its disclosure limitation policy for statistical information based on Petroleum Supply Reporting System (PSRS) survey data. Beginning with survey data for January 2004, EIA extends its 1986 policy of not applying disclosure limitation methods to statistics based on PSRS survey data to all PSRS survey information collected under a pledge of confidentiality. EIA will continue to protect information collected under a pledge of confidentiality by not publicly releasing

ORO-777

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Description of the Formerly Utilized Sites Remedial Action Program

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DESCRIPTION OF THE FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM

Notice: This document contains information of a preliminary nature; it is subject to revision or correction and therefore may not represent a final report.

U.S. DEPARTMENT OF ENERGY
Oak Ridge Operations
P.O. Box E
Oak Ridge, Tennessee 37830

September 1980

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DESCRIPTION OF THE FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM*

1.0 Introduction

The background and the results to date of the Department of Energy program to identify and evaluate the radiological conditions at sites formerly utilized by the Corps of Engineers' Manhattan Engineer District (MED) and the U.S. Atomic Energy Commission (AEC) are summarized in section 2.0. The sites of concern were federally, privately, and institutionally owned and were used primarily for research, processing, and storage of uranium and thorium ores, concentrates, or residues. Some sites were subsequently released for other purposes without radiological restriction. Surveys have been conducted since 1974 to document radiological conditions at such sites. Based on radiological surveys, sites are identified in this document that require, or are projected to require, remedial action to remove potential restrictions on the use of the property due to the presence of residual low-level radioactive contamination. Specific recommendations for each site will result from more detailed environmental and engineering surveys to be conducted at those sites and, if necessary, an environmental impact assessment or environmental impact statement will be prepared. Section 3.0 describes the current standards and guidelines now being used to conduct remedial actions. Current authority of the U.S. Department of Energy (DOE) to proceed with remedial actions and the new authority required are summarized in section 4.0. A plan to implement the Formerly Utilized Sites Remedial Action Program (FUSRAP) in accordance with the new authority is presented in section 5.0, including the objectives, scope, general approach, and a summary schedule. Key issues affecting schedule and cost are discussed in section 6.0.

2.0 Background

Historical Records Review

The original program for the development and use of atomic energy, established under the MED and later continued by the AEC, involved the development of technology and the production of nuclear materials for national defense and security. The program was conducted under very stringent security restrictions and, at contract termination of the MED/AEC activities, the sites involved were decontaminated according to the health and safety criteria and guidelines then in use and applied on a site-specific basis. However, radiological criteria for releasing these sites for unrestricted use have changed and some criteria are still being developed. Therefore, to define the radiological condition of these sites in light of the changing environmental criteria and standards, a records search was begun in 1974.

In many instances, documentation of the MED/AEC activities at these sites was destroyed in compliance with Government Records Management practices. Many of the radiological records covering the extent of cleanup actions are incomplete. Also, many of the sites have changed ownership and are presently used for other purposes. In some cases, buildings have been modified or the earlier MED/AEC facilities no longer exist.

*Much of the information presented in this document was extracted from a draft of "A Background Report for the Formerly Utilized MED/AEC Sites Remedial Action Program," prepared for the Environmental Control Technology Division, Assistant Secretary for Environment, U.S. Department of Energy, by the Aerospace Corporation, March 1980.

AEC/ERDA/DOE Site Survey Program

In early 1974, the AEC initiated a survey program to identify all formerly utilized sites involved with nuclear materials and to determine their radiological status. The responsibility for this survey was assigned to the Division of Operational Safety. At that time, all divisions and field offices of the AEC were required to search their files to identify any such former government-owned or leased sites and facilities that had been used in the research or production activities of the MED and the AEC. In addition, the files were searched for records identifying the radiological conditions at the termination of the MED/AEC activities and/or the transfer of custodial responsibility for such sites, the current radiological condition of the sites, and the land-use and ownership data. This effort identified many additional sites for which pertinent information was lacking or was insufficient to determine their radiological conditions.

On January 19, 1975, the AEC was abolished and its programmatic responsibilities transferred to the Energy Research and Development Administration (ERDA) which continued the activities of the survey program. Contacts were made with former and current owners and site visits were conducted under the direction of the ERDA field offices to determine the need for radiological surveys. If radiological surveys were determined to be necessary, the permission of the site owners was obtained and a press release was issued to inform the public of the survey work. Subsequent survey results were also issued in a public press release and were published in a radiological survey report that analyzed the significance of the findings with respect to the potential risks to the public health.

Pursuant to the DOE Organization Act of 1977, the functions and authority of the ERDA were transferred to the DOE. In the DOE, the Assistant Secretary for the Environment (ASEV) was assigned the responsibility for the site-survey program. The results of several site surveys clearly indicated that some remedial action would be needed, not only on the former sites, but also on adjacent or remote properties that had become contaminated from the original processing site. Due to the importance of this effort, the ASEV initiated the FUSRAP and drafted a generic plan to identify all formerly utilized sites and to resolve any site radiological problems. Using this generic plan as a guide, in mid-1979 responsibility for the FUSRAP activities was divided between the ASEV and the Assistant Secretary for Energy Technology (now Assistant Secretary for Nuclear Energy [ASNE]). The ASEV is responsible for identifying the sites, characterizing the radiological condition, determining the need for remedial action at the sites, and ultimately for certifying the post-remedial action radiological condition of the FUSRAP sites. The ASNE is responsible for implementing the required remedial actions, including suitable disposal or stabilization of residual material.

Overview of MED/AEC Activities

In 1942, under the jurisdiction of the U.S. Army, the MED was established as the agency responsible for the development of nuclear materials for national defense and security. The authority for process development, engineering design, procurement of materials, and site selection associated with the nuclear materials program was transferred to the MED from the Office of Scientific Research and Development, Department of the Army. The headquarters for the MED, originally established in New York, was transferred to Oak Ridge, Tennessee, in 1943.

On December 31, 1946, the MED was deactivated and its responsibilities were transferred to the newly constituted AEC. During the 1942 to 1946 time period, there

were more than 10 contractors and several hundred subcontractors involved in the production, research, and development operations. These contractors included indus trial concerns, universities, and other scientific organizations. In contrast to the highly centralized operation of the MED, the AEC decentralized and established five major centers of operation (New York City, New York; Santa Fe, New Mexico; Oak Ridge, Tennessee; Hanford, Washington; and Chicago, Illinois). The AEC continued the MED practice of contracting with industrial concerns and academic institutions to perform the actual operations.

The most readily available source of historical information on the early activities of the MED/AEC is A History of the United States Atomic Energy Commission, Volume I - The New World and Volume II - Forging the Atomic Shield. A synopsis of the procurement, storage, and processing of the raw materials containing uranium is presented here to give the reader a general overview of the MED/AEC activities.

Uranium Procurement. The MED relied on three sources of uranium during the war years. About two-thirds came from mines in the Belgian Congo, slightly more than one-sixth from mines near Great Bear Lake in Canada, and the remainder from American ores, which in reality were tailings from vanadium refinery operations.

African Sources. At the beginning of the nuclear program in the late 1930s and early 1940s, it was determined that, while there were significant quantities of uranium ore available in Czechoslovakia and Canada, the most important sources, by far, were in the mines of the Belgian Congo. The supplies of ore in the United States were not considered extensive and, with the growing interest in uranium, Germany ceased all sales of the Czechoslovakian ores. As a result of this, plus the German takeover of Belgium and the increased German activity in Africa, the United States, Great Britian, and Canada made an all-out effort to obtain as much of the Belgian Congo ore (pitchblende) as quickly as possible to guarantee adequate supplies of uranium for the war period. Through activities that began in September 1942, the United States was able to purchase all of the above-ground supplies of uranium ore from the Belgian Congo. This included 1,200 tons of ore (65 percent uranium) from African Metals' predecessor, Union Miniere, that had been imported to the United States in 1940 and stored in the Archer-Daniels Midland Company warehouse, Port Richmond, Staten Island, New York, and some 3,000 tons of similar ore still in the Congo. By the end of 1944, the U.S. Army had received approximately 3,700 tons of Congo ore.* The amount of ore being received far exceeded the processing capacity in North America at that time, and the ores had to be stored. The MED used three primary storage areas: Seneca Ordnance Depot, Romulus, New York; Clinton Engineer Works (now Oak Ridge National Laboratory), Clinton, Tennesssee; and Perry Warehouse (Middlesex Sampling Plant), Middlesex, New Jersey. The Perry Warehouse also became a sampling, weighing, and assaying facility.

The MED contracts with African Metals, Inc., involved only the recoverable uranium oxide (U_3O_8 black oxide**) in the ore. African Metals maintained ownership of the residue or tailings that contained radium and other precious metals. As a result, it was necessary for the MED to establish weighing and assaying operations. Initially, the weighing and assaying were performed at contractor facilities; however, in November 1943, the MED set up a separate sampling program at the Perry Warehouse.

*By the end of 1946, MED had contracted for approximately 3,800 tons of U_3O_8 from over 29,000 tons of African ore containing from 5 to 65 percent uranium oxide.

**The various steps of the uranium recovery and refining process produced various concentrations and compounds of uranium oxide, which were generally referred to by their color and chemical state.

The weighing and assaying of the ore samples were performed for the Federal Government by Lucius Pitkin, New York, New York; Frick Chemical Laboratory, Princeton University, Princeton, New Jersey; and the National Bureau of Standards (NBS), Washington, D.C. Weighing and assaying for African Metals, Inc., were performed by Ledoux and Company, New York, New York.

Following weighing and assaying, the ore was shipped to the various refineries to be processed to black oxide or sodium diuranate concentrates. Because the tailings were owned by African Metals, Inc., the MED was required to store the residues from these operations until they could be returned to the owner. These residues from ores containing greater than 10 percent U_3O_8 were stored at the Clinton Engineer Works or the Perry Warehouse before return shipment. Residues from ores containing less than 10 percent U_3O_8 were stored at the Lake Ontario Ordnance Works (LOOW). Some of this residue was returned to African Metals and some is still at U.S. storage sites.*

Canadian Sources. Negotiations to obtain Canadian ore were begun in 1942 with Eldorado Gold Mines, Ltd., (later Eldorado Mining and Refining, Ltd.). The Eldorado Gold Mines, Ltd., mined uranium ore at their Great Bear Lake mine and refined the Canadian ore at their facility at Port Hope, Ontario. By 1944, about 400 tons of the oxide had been produced and enough Canadian ore had been mined to produce an additional 500 tons of the oxide. By 1946, over 4,000 tons of ore concentrate containing over 1,100 tons of U_3O_8 in the form of black oxide had been delivered to the MED. Because the Canadian ore was processed to black oxide at the Eldorado facility and the entire concentrate was sold to the MED, no weighing and assaying program was set up for the Canadian ore.

Domestic Sources. Most of the uranium in the United States was in carnotite ores on the Colorado plateau, but the high-grade deposits had already been mined earlier primarily for the radium content. The heavy demand for vanadium during the war also created the potential for a practical source of uranium oxide as a by-product of the vanadium processing. However, the tailings from vanadium processing were of such low uranium content that it was necessary to concentrate them at or near the mine prior to their shipment to the processing facilities. The United States Vanadium Corporation's concentrated vanadium tailings were stockpiled at Uravan, Colorado, to produce a sludge containing 15 to 20 percent black uranium oxide. This sludge was transported directly to the Linde Refinery in Tonawanda, New York. The U.S. Vanadium Corporation also had a plant at Durango, Colorado, for processing vanadium tailings and sands to produce a sludge. The output from the Durango and Uravan facilities went to Grand Junction, Colorado,** for processing to "yellow cake" (10 to 15 percent U_3O_8) that, in turn, went to the Linde refinery at Tonawanda, New York.

Concurrent with the U.S. Vanadium Corporation operation, the Vanadium Corporation of America processed American ores for vanadium at its plants in Naturita, Colorado, and Monticello, Utah.** Most of the slimes (50 percent U_3O_8 by weight) from these plants went directly to Vitro Manufacturing Company, Canonsburg, Pennsylvania,

*Some of the African Metals residue that is still in the United States is currently stored at the Feed Materials Production Center, Fernald, Ohio.

**Uranium mills which produced concentrates for MED/AEC programs that are inactive are covered under the Uranium Mill Tailings Radiation Control Act of 1978.

for processing. A portion of the 50-percent slime tailings were sold to the government and processed at the Uravan facility. By the end of 1944, domestic ore production had yielded less than 800 tons of uranium oxide, and, by the end of 1946, over 1,300 tons of uranium oxide had been produced in various concentrations from the domestic sources.

Uranium Processing Operations and End Use. The initial refining operations consisted of mechanical grinding and crushing of the ores to a sandy material. Acid was used to dissolve and, hence, extract the uranium. The acid extract was treated with other chemicals to precipitate the majority of impurities, and the product was further treated to precipitate the uranium. A final roasting and drying operation produced a black oxide (U_3O_8) or sodium diuranate ($Na_2U_2O_7$) concentrate.

During World War II, the ores were refined to black oxides at the facilities of Linde and Eldorado. Vitro (at Canonsburg) refined the ores to produce sodium diuranate. Following the war, Mallinckrodt Chemical Co., Inc., also produced black oxide at its facilities in St. Louis, Missouri, and later at the AEC Weldon Spring Chemical Plant.

Black oxide and sodium diuranate were further refined to orange oxide (UO_3) at the Mallinckrodt Chemical Company plant, St. Louis, Missouri, and by E.I. du Pont de Nemours and Company, Deepwater, New Jersey.

At the du Pont plant, brown oxide (UO_2) was made from black oxide and from uranium peroxide ($UO_4 2H_2O$) obtained from uranium scrap processing. About one-half of the du Pont output was from scrap and by-product material. Brown oxide was also produced by Harshaw Chemical Company (Cleveland, Ohio), Linde, and Mallinckrodt. Brown and orange oxide were in turn refined into green salt (UF_4) by du Pont, Harshaw, Mallinckrodt, and Linde.*

Harshaw made uranium hexafluoride for the thermal diffusion and gaseous diffusion uranium-235 separation projects. The green salt was used mainly in metal manufacturing by du Pont; Mallinckrodt; Iowa State College (now University), Ames, Iowa; Westinghouse, Bloomfield, New Jersey; Brush Laboratories, Cleveland, Ohio; and Electromet, Niagara Falls, New York. Scrap metal recovery operations were conducted at Metal Hydrides, Inc., Beverly, Massachusetts, and Iowa State College.

Uranium metals in the form of powder were also produced directly from uranium oxides instead of green salt by Metal Hydrides. The metals manufactured by these various companies were then shipped to the Hanford Site at Richland, Washington, for use in plutonium production. The plutonium produced at Hanford was then shipped to Los Alamos for use in the weapons development program.

Quality control of various processes in the ore/metal production chain was performed by the University of Chicago, Metallurgy Laboratory, Chicago, Illinois; Princeton University, Princeton, New Jersey; Massachusetts Institute of Technology, Cambridge, Massachusetts; and the National Bureau of Standards, Washington, D.C.

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*Following the war and after the construction of the Weldon Spring Chemical Plant, much of the AEC uranium-conversion operations were centralized and transferred to Weldon Spring under Mallinckrodt and the Feed Materials Processing Center at Fernald, Ohio, under the National Lead Company of Ohio. The latter is currently the center for uranium-conversion operations.

Activities following World War II broadened in scope. The AEC entered into a number of research, development, and production contracts to recover uranium as by-products of certain industrial processes such as phosphoric acid production. In addition, contracts were terminated or established as product needs and research needs varied.

In addition to the actual contractor-owned facilities, a number of offsite storage locations were used such as landfills for disposal of low-level contaminated soil and waste from the uranium-ore-handling operations. Examples include the St. Louis Airport Storage Site, where residue from the Mallinckrodt AEC Operations were deposited; the former Haist property, Tonawanda, New York, where material from the Linde AEC operations was deposited; the Burrell Township-Pennsylvania Railroad Landfill, where Vitro Corporation deposited residues from Canonsburg; and the Middlesex Municipal Landfill, Middlesex, New Jersey, where residues were deposited during construction activities at the Middlesex Sampling Plant. Some private properties in Middlesex also became contaminated inadvertently as a result of radionuclide migration.

The companies and locations discussed in this report were identified during the records review of the MED history conducted under the FUSRAP activities.

Thorium Operations. Operations with thorium after the war were similar to the uranium operations, but were conducted on a smaller scale. The first major research for the MED on thorium was begun early in 1946 with the procurement of thorium salt for a research project at Iowa State College. The thorium salts were supplied by Lindsay Light and Chemical Company, which was the major supplier through most of the early years of the program.* Lindsay Light and Chemical Company first received thorium from Germany and later processed monazite ores from India and Brazil. In later years, processing of monazite and other ores for the AEC was accomplished by other industrial firms such as the Davison Chemical Division of the W. R. Grace Company, Curtis Bay, Maryland; Dow Chemical Company, Walnut Creek, California; and by Iowa State College. Extractive research, metal production and handling, and research and development for both uranium and thorium was conducted at a number of companies including Mallinckrodt, Simonds Saw and Steel, Lockport, New York; Sylvania Corning Nuclear Corporation, Bayside, New York; Battelle Columbus Division, Columbus, Ohio; Brush Beryllium Company, Cleveland, Ohio; and Horizons Metal Inc., Cleveland, Ohio.

The National Bureau of Standards was involved in quality control for the thorium programs, and the Middlesex Sampling Plant was used for storage of some thorium. A major objective of the DOE FUSRAP effort currently underway is to ensure that all of the thorium sites have been identified and surveyed for radiological conditions. More in-depth record searches and personal communications with former AEC employees are also being conducted.

3.0 Current Standards

Throughout this report and in the site summary reports in Appendix A, reference is made to "established standards" and current guidelines for contamination and exposure levels. These standards/guidelines are as follows:

*Lindsay Light and Chemical Company was using thorium for gas mantles, catalysts, and electron tube cathodes prior to nuclear applications. Remedial action activities at this site and associated properties are being undertaken by the State of Illinois and Kerr-McGee Chemical Corporation, with assistance from the Nuclear Regulatory Commission (NRC).

- Surface Contamination

  "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source or Special Nuclear Material," by the USNRC, November 1976.

  The NRC Decontamination Guidelines present alpha and beta-gamma limits for surface contamination for both fixed and transferable contamination, dependent on the mixture of nuclides present.

- Radon Daughter Products and External Gamma Radiation Exposure

  A regulation based on the Surgeon General's Guidelines, "Grand Junction Remedial Action Criteria," 41FR56, 777-56, 778, December 30, 1976.

  In 1972, Congress passed P.L. 92-314 that provided remedial action in the community of Grand Junction, Colorado. Regulations implementing that law were issued by the AEC, then ERDA, as 10CFR712. P.L. 92-314 was later extended by P.L. 95-236.

  In all cases, the most restrictive guideline (that for schools or dwellings) has been used. However, it should be noted that on several of the sites where the contamination is associated with an industrial building rather than with the soil, little likelihood exists of the site being used for these more restrictive purposes.

- Air and Water Concentrations

  10CFR20, Appendix B, Table II presents, by nuclide, concentration limits in both water and air for the general public. The value of the most restrictive form, either soluble or insoluble, has been used.

The EPA has proposed regulations for private uranium mill tailing sites: 40CFR192, "Interim Cleanup Standards" and "Final Cleanup Standards for Inactive Uranium Mill Tailing Sites," 45FR27366. These standards cover cleanup of open lands and contaminated buildings associated with these sites.

4.0 Legislative Authority

Current Authority

Pursuant to the First War Powers Act of 1941 and the Atomic Energy Acts of 1946 and 1954, as amended the MED and its successor, the AEC, conducted during the 1940s and 1950s a program involving research, development, processing, and production of uranium and thorium. This program also included the storage of radioactive ores and processing residues, e.g., mill tailings. Virtually all of this work was performed by private contractors for the government on land that was either federally, privately, or institutionally owned.

Due to the urgency and magnitude of the early nuclear materials programs and the limited knowledge available regarding the radioactive characteristics of uranium ore and residual material from its processing, many of these sites became contaminated with radioactivity as a result of work done for the government.

In several western states, uranium mill tailings (a waste product of the uranium mill processing operations that was not subject to regulation by the government) accumulated in large piles and contaminated private adjacent and vicinity properties by migration. In some instances, these tailings were also used as fill and construction material in various construction work in the communities. The presence of these tailings containing radium caused radon gas to collect in dwellings and in many cases produced unacceptable exposure to occupants. The government had no statutory authority to take remedial action; however, out of a sense of moral responsibility toward the affected homeowners, the Congress in 1972 passed P.L. 92-314 that provided for remedial action in the community of Grand Junction, Colorado. Regulations implementing that law were issued by the AEC and then by ERDA as 10 CFR 712. P.L. 92-314 was later extended by P.L. 95-236. Additional extensions of this program have been authorized and will be sought as needed in the annual DOE budget authorization and appropriation requests.

In 1978, Congress passed the Uranium Mill Tailings Radiation Control Act (P.L. 95-604) under which the DOE was authorized to enter into cooperative agreements with various states for undertaking remedial actions at certain designated inactive former uranium mill processing facilities in the United States. The scope of this Act was very narrowly drawn to cover, under section 101(6), the sites designated in the Act and any other former processing sites and contaminated nearby properties at which substantially all of the uranium was produced for sale to the United States Government. None of the FUSRAP sites could qualify under this definition because the uranium and thorium processed at these sites were generally owned by the government. Excluded from coverage under the Act are those sites owned or controlled as of January 1, 1978 or thereafter by a Federal agency, or under active NRC or Agreement-State license. The legislative history made it clear that this Act was not to set a precedent for the DOE to undertake other waste management remedial action programs. Pursuant to that Act, the EPA Administrator was authorized and directed to develop environmental and health standards for uranium mill tailings contamination covered by the Act.

The FUSRAP program formally began in 1974. Radiological surveys and other research work have been conducted by the AEC and its successors, the ERDA and the DOE, under the implied authority of the Atomic Energy Act of 1954, as amended. The intent of Congress, as expressed in the FY 1978 DOE Authorization Act was that, at the completion of this program, the DOE would seek additional legislative authority, pursuant to a Congressional review of findings, for the undertaking of any required remedial action work.

A survey of existing statutory authority shows that pursuant to the Atomic Energy Act of 1954, as amended, the AEC was directed to protect public health and safety during the research and production operations. In the case of those operations over which the government exercised ownership or control, the DOE's existing authority has been interpreted to include the implied authority to decontaminate such sites through remedial actions undertaken at the conclusion of contract work. Accordingly, the DOE has undertaken remedial action efforts at the Kellex site in Jersey City, New Jersey, and in Middlesex, New Jersey. However, the absence of sufficient contractual, property, or other historical records (as a result of records retention schedules and limitations) has prevented final determination of the extent of government involvement in, and implied remedial action authority over, many of the sites. In addition, explicit contractual language and/or notations in deeds under which the United States is relieved from all contractual liability raises the issue as to whether, without the proposed legislation, the government has any continuing financial or other responsibility with respect to these properties.

Existing statutory authority has been reviewed by the DOE, in addition to all available contract, property records and other files, to determine the extent to which the DOE could exercise its existing authority under the Atomic Energy Act of 1954, as amended, to perform remedial action work under the FUSRAP program. As part of this study, consideration was given to the extent to which the MED and the AEC would have been contractually responsible for the costs of decontamination, and whether the contractors and/or property transferees involved recognized the presence of the contamination when they closed out their contracts with the United States Government. This review has shown that authorization exists for remedial action at 10 sites.

Unlike the uranium mill tailings sites, none of the FUSRAP facilities were at any time licensed for conducting the MED/AEC activities because many were either in operation before licensing requirements were established or were excluded from the licensing requirements pursuant to Section 110 of the Atomic Energy Act of 1954, as amended. Three sites, Gilman Hall at the University of California, Berkeley, California; Linde Air Products at Tonawanda, New York; and the University of Chicago, are currently licensed under the NRC or the Agreement State provisions of the Atomic Energy Act of 1954, as amended, and are excluded from the FUSRAP remedial action because the NRC or the Agreement State has sufficient licensing authority to protect public health and safety.

Legislative authority will be required to clarify the DOE's authority for remedial action at 18 of the FUSRAP sites discussed in this report and for the location and acquisition of disposal sites.

The EPA is responsible for establishing radiological standards of general applicability for properties released for unrestricted use; the NRC has responsibility for establishing criteria and standards for restricted use sites that would be licensed. The NRC criteria would be basically modeled after 10CFR40 Appendix A, proposed regulations for licensed active uranium mill tailings sites.

New Authority Needed

Broader authority is needed to conduct remedial action at the formerly utilized MED/AEC sites that are determined by established criteria to pose a potential threat to the public or to the environment because of their radiological contamination. The new authority should include any location where the MED or the AEC activities resulted in residual contamination exceeding established standards, including associated properties that became contaminated from these activities. Sites that are licensed by the NRC or by an Agreement State under Section 274 of the Atomic Energy Act of 1954, as amended, should be excluded from the authorization.

The authority would not include sites currently owned or leased by the DOE since no clarification of authority is needed for these sites. However, new authority is needed for the DOE to perform remedial actions at three properties that were formerly owned or leased by the Federal Government. These properties* were transferred to the present owners by quitclaim deeds or other documents under which the present owners released the Federal government from all responsibility for claims relating to the presence of the residual radioactive material. These sites are being included in the scope of the FUSRAP in order to expedite cleanup and to provide for the long-term

*St. Louis Airport storage site, Palos Park Forest Preserve, and Ashland Oil Company.

Federal management at the site, or at new federally owned disposal sites. This approach is consistent with the recommendations of the Interagency Review Group on Nuclear Waste Management.

In addition to the formerly utilized MED/AEC sites, there are other contaminated sites that were used for processing and using of radium-containing ores. At some of these sites, work was performed for the Federal Government. Authority is needed to identify and conduct radiological surveys at all such sites known to contain radioactive material above background levels that resulted from the processing of uranium or thorium ores and/or their daughter products, including radium, for the purpose of informing Congress of the extent of contamination and of the estimated cost for remedial action.

Under the existing and proposed new authority, radiological conditions at the MED/AEC sites would be assessed, relative priorities established on the basis of the potential health hazard, and determination made to conduct remedial action if present site conditions or possible unrestricted future use would constitute a risk to the public. Restitution to the Federal Government for the costs of remedial action would be provided for if the identity of any person having legal responsibility to clean up a site could be determined. Currently, the DOE is contacting those parties it has reason to believe could be shown to be legally responsible for remedial action at a site, to secure their agreement to undertake clean-up operations, or for the reimbursement of expenses that may be incurred by the DOE for remedial actions.

For the states containing MED/AEC sites, the DOE Secretary would consult with the state to determine whether it is unreasonable to remove sufficient contaminated material from the site to release it for unrestricted use, or whether residual radioactive material could be stabilized onsite as a permanent disposition action. Initially, the DOE would acquire the MED/AEC sites for remedial action purposes and to minimize health effects or to prevent windfall profits. Any property acquired or dedicated for use as a permanent disposal site would be licensed by the NRC. Affected states in which radioactive contaminated sites are located would be responsible for locating suitable disposal sites for the residual radioactive material; initially, the DOE would acquire this property. The disposal sites could be transferred to the state by agreement to accept ownership and custodial responsibilities. The DOE would have authority to provide financial support to the state in carrying out the custodial responsibilities.

The EPA Administrator would be authorized, in consultation with the DOE Secretary, to develop health and environmental standards of general applicability for residual radioactive materials at formerly utilized sites that are to be released for unrestricted use. These general standards would supplement and be consistent with standards established by the Administrator under the Uranium Mill Tailings Radiation Control Act of 1978. Where such standards do not exist, the Administrator would be required to promulgate the needed standards within a specified time.

The DOE Secretary, in consultation with the EPA Administrator, could promulgate remedial action standards for each site at which the Secretary determines it is necessary to begin remedial action before the Administrator promulgates standards of general application.

The DOE has proposed legislation to provide the needed authority. This proposed legislation is under review by other Federal agencies and the Office of Management and Budget.

5.0 FUSRAP Program Description

Objectives of Remedial Action

The objectives of the FUSRAP are to:

- Identify former MED/AEC sites
- Characterize their radiological condition
- Decontaminate sites as required and pursuant to authorization and appropriation by Congress
- Develop acceptable disposal and stabilization sites in consultation with the affected states, and ultimately
- Certify the acceptability of the sites for future use.

The effort to accomplish the first two of these objectives has been initiated. The authority sought under the legislation proposed by the DOE is necessary in most cases to accomplish the remaining objectives.

Scope and Problem Definition

The scope of the FUSRAP program is confined to those MED/AEC sites that were formerly under contract to, or owned by, the government and were involved in the handling, processing, and storage of radioactive materials. The materials processed consisted primarily of pitchblende and carnotite ores, and other materials from which uranium and thorium were recovered as products. The products of the processing included uranium and thorium metals and compounds. Waste by-products were also produced that generally contained low levels of radioactivity due to residual quantities of uranium, thorium, and their radioactive decay products. In some cases, these contaminants have migrated offsite. Radium contamination is a major concern because it decays to a radioactive gas, radon, that diffuses into the air and can be inhaled. Furthermore, the radon decays to radioactive solid materials that can also be inhaled or ingested.

Also included in the sites discussed in this report are Palos Park, Illinois, where the remains of two research reactors are buried; Chupadera Mesa, New Mexico, which is near the location of the Trinity atom bomb test; and two other sites at Los Alamos, New Mexico, involved in the nuclear weapons development program. At the Palos Park site, the primary contaminant of concern appears to be tritium. At the sites involved in weapons development, plutonium and other nuclides such as uranium-235 and strontium-90 are of concern.

Approach to Remedial Action

Consistent with the objectives of the FUSRAP, sites are being identified by searching through the MED/AEC records and by publishing press releases asking for public assistance in identifying the sites. After a site has been identified, it is assigned to one of the DOE national laboratories whose responsibility is to assess the site's radiological condition. This is accomplished by performing a records search, reviewing old radiological survey documents, and performing radiological surveys as required. A series of engineering studies and environmental reports, including those prescribed by the National Environmental Policy Act (NEPA), will be prepared to evaluate remedial action alternatives. After the evaluation of the alternatives, appropriate measures (remedial actions) will be selected and implemented, and the resulting contaminated wastes will be disposed of in a manner that ensures public safety and compliance with

the provisions of the Atomic Energy Act of 1954, as amended, and related NRC or Agreement State licensing requirements. In some cases, the residual radioactivity will be stabilized onsite in accordance with the provisions of a license from the NRC or Agreement State. When a site is decontaminated sufficiently to comply with the EPA standards for unrestricted use, it will be certified for release by the DOE. During the course of the investigation, the public will be informed, through press releases for example, of the nature of the MED/AEC work done at the site, the contamination potential, survey results, and remedial actions undertaken. Detailed reports of the survey findings will also be published by the DOE and, upon request, will be available to the public for a nominal fee.

The approach to identification and eventual correction of radiological contamination at the MED/AEC sites or adjacent properties is dependent upon institutional issues which, in turn, impact the steps of the generic program plan for the FUSRAP.

Institutional Issues. Three paramount issues must be addressed and solutions defined before remedial actions as outlined in the generic FUSRAP plan can be implemented:

- Legislative authority must be established by which the Federal Government (DOE) can act to correct problems of radiological contamination at formerly utilized sites. Although the DOE has implied authority at some sites, a large number of sites will require additional legislative authority.

- Radiological criteria must be developed for use as guidelines to determine the extent of decontamination required at each site, to determine if a radiological problem exists, and to establish standards for unrestricted use.

- Disposal sites must be developed for ultimate disposal of contaminated material that is removed from the MED/AEC sites.

Sequence of Events Leading to Remedial Action. Although each formerly utilized site will have certain site-specific characteristics, a general sequence of events can be outlined leading to the ultimate program objective, which is to preclude any future radiological problems at formerly utilized sites from previous MED/AEC activities.

Figure 1 is a schematic presentation of the basic steps involved in the remedial action program. Step 2 determines which sites need remedial action. Sites needing remedial action must be addressed in each of the following steps. If no remedial action is necessary, only Steps 1, 2, and 8 are required. A brief discussion of each step follows.

Step 1, Site Identification - The overall objective of this step is to identify and locate all candidate sites and to determine if any actions are required under the FUSRAP.

The activities include a records search and review of information submitted by the public or industry in response to specific requests. When a site is identified as having been exposed to radioactive materials under the MED/AEC activities, a records search will be initiated to determine the radiological condition of the site. If there is adequate documentation that indicates the site is not contaminated, the site will be certified as clean and no further action will be required. If the documentation or records are inadequate or indicate the site may be contaminated, survey efforts to determine or verify the radiological condition of the site will be initiated. These activities will be performed by the ASEV.

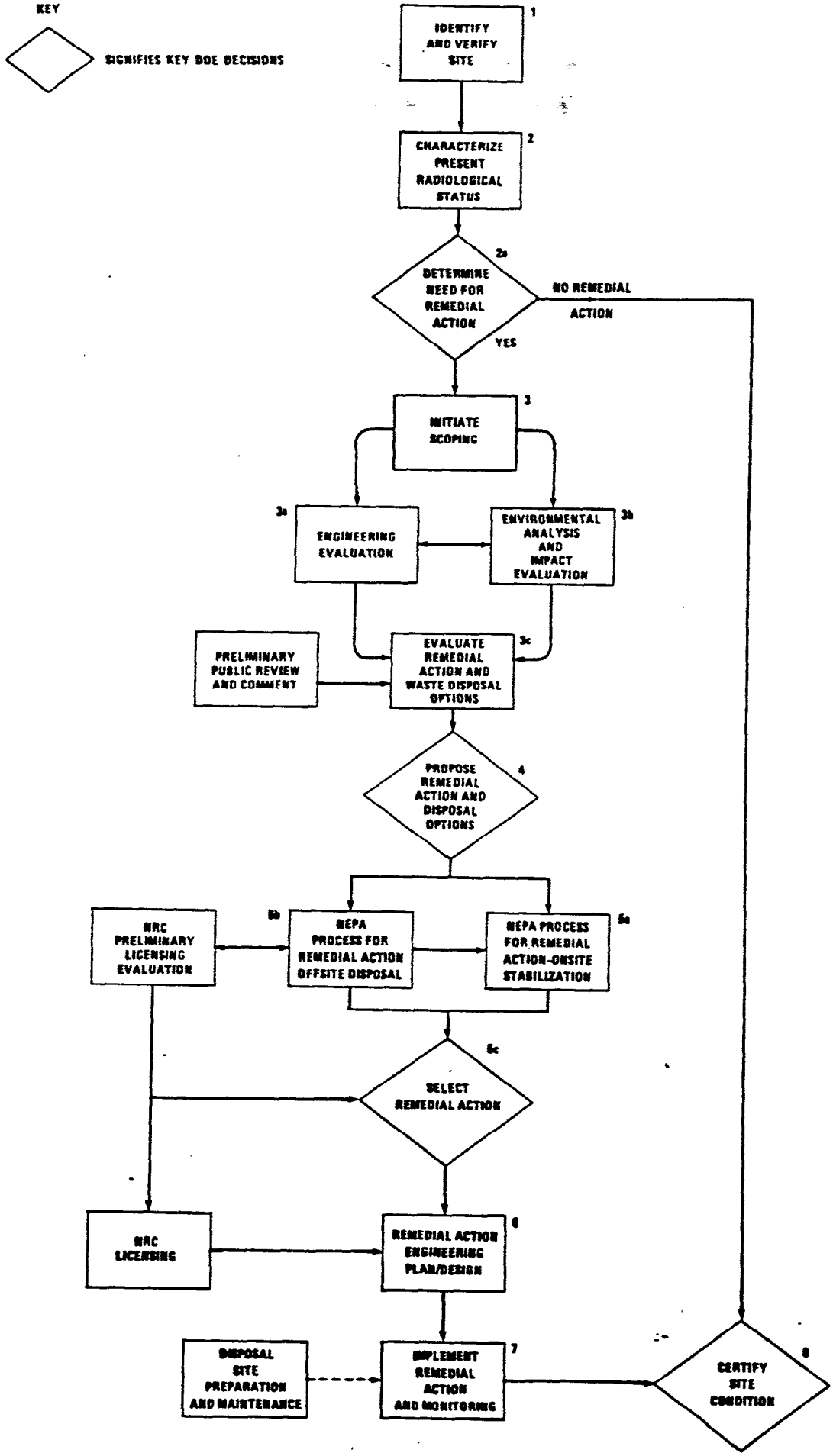


Figure 1. Formerly Utilized Sites — Basic Steps Involved in the Remedial Action Program

A large portion of this step in the FUSRAP is complete. An effort has also been initiated to identify the disposition of equipment that may have been removed from the FUSRAP sites and also to identify the subcontractors to the MED/AEC contractors. One concern is the location of material and equipment that was removed as part of the earlier AEC decontamination efforts.

Step 2, Radiological Survey - The purpose of this step is to characterize the current radiological condition of those sites determined in the preceding step to require a radiological survey. A plan for the radiological survey of a specific site will be prepared, taking into account the past and current activities at the site and associated radioactive material and potential contamination. The extent of the effort associated with a specific site survey will depend on the data available. In some cases, earlier survey reports exist and only supplemental information is required to characterize the site; in other cases, no data are available and a radiological survey is required. The elements that make up the complete radiological survey include the following:

- Measurements of fixed and transferable alpha and beta-gamma radiation on buildings and equipment surfaces
- Gamma-ray exposure rates
- Beta-gamma exposure rates
- Alpha exposure rates
- Radionuclide contamination in surface water and groundwater
- Radionuclide contamination in building drains and associated components
- Radionuclide contamination in underground drains and surface drainageways
- Surface and subsurface deposits of radioactive material
- Radionuclide concentrations in air
- Radionuclide concentrations in vegetation samples.

These activities will be performed by the ASEV.

In order to place all measurements and results in the proper perspective with the surrounding area, measurements of a similar nature will be performed in areas not affected by the former MED/AEC activities. These results will be used to represent the natural background radiation of the area. Aerial radiometric surveys will also be performed in support of the radiological assessment, independent of the ground-level radiological survey. The most important result from this effort will be the identification of any unknown offsite contamination. If the aerial survey indicates the presence of contamination not previously detected, the new area will be surveyed from the ground.

When the field work is complete, a survey report that characterizes the radiological condition of the site will be prepared. The report or report supplement will also include, for contaminated sites, an evaluation of radiation exposures to man from known radiation exposure pathways at the site. This evaluation will outline the levels of radioactivity and extent to which humans could be exposed in the course of normal site activity. These levels will be compared to levels of exposure received from normal background sources of radiation to place the exposure in perspective. The evaluations will be prepared on the basis of the conditions at the site during the radiological survey. In cases where the possibility of radiation exposure above background levels is identified, either summaries or the complete report will be submitted to appropriate state regulatory authorities, the EPA, and the NRC.

Upon public release of the report, or before, meetings will be held with affected property owners and concerned agencies to explain the results of the survey and the future DOE plans for action. Press releases will also be used to inform the public and provide an accurate basis for understanding the results of the radiological survey.

Step 2a, Determine the Need for Remedial Action - The radiological status report will be reviewed and will provide the basis for a determination by the DOE as to whether remedial action is required to remove or reduce residual radioactive materials to levels that conform to the applicable EPA, NRC, or DOE standards, including those to be developed pursuant to the proposed legislation. This determination will be performed by the ASEV and provided to the ASNE, who will be responsible for accomplishing the remedial action.

Step 3, Initiate Scoping - The purpose of this step is to begin the process of identifying the specific alternative remedial actions to be examined and, as appropriate, the candidate disposal sites. This step will involve interactions with the affected state and local authorities, the EPA, the NRC, and other appropriate agencies. The principle issues to be examined will be identified, and the responsibilities, schedule, and appropriate interfaces for conducting the necessary studies will be agreed upon. A key output is for the state to identify candidate disposal sites for subsequent study during the engineering and environmental evaluation. To obtain this information, the DOE would work with the states and support screening studies. Two disposal options will generally be evaluated: a permanent disposal site within the state where the wastes are generated, and a regional disposal site for remedial action wastes from states within the region. Regional sites that could satisfy the needs of several states is a preferred option to minimize the number of disposal sites.

Step 3a, Engineering Evaluation - Engineering evaluations will be required only for those sites for which radioactivity is found to exceed the established health and safety guidelines (e.g., see section 3.0) and/or the standards to be developed. The engineering evaluation will include assessment of existing conditions for the site as well as surrounding properties. The scope of the effort will include the following:

- Verification of property ownership
- Preparation of descriptive maps and site plans
- Analysis of radiological surveys to determine decontamination requirements and identify and collect any supplemental data needed for a sound engineering evaluation of remedial action options
- Performance of an engineering assessment of the decontamination or demolition of structures
- Engineering evaluation of removal, transport, interim storage, and permanent disposal options for contaminated soil, structures, debris, and other materials
- Evaluation of suitable means of stabilizing residual radioactivity, where appropriate, including investigation of pertinent aspects of site geology, hydrology, and meteorology
- Analysis of alternative remedial action options including preliminary project plans for the remedial action and disposal sites, specifications, and cost estimates
- Preparation of summary reports.

Step 3b, Environmental Analysis - The objective of the environmental analysis is to provide an environmental evaluation of the remedial action options covered by the engineering evaluation. The analysis will discuss the environmental impacts of the

present condition of the site, stabilization of the material onsite and/or decontamination of the site, and removal of the material to a temporary storage or to a disposal site. This analysis will provide a basis for determining whether a major Federal action is involved that may require the preparation of an environmental impact assessment or impact statement conforming to the requirements of the NEPA. Environmental analysis and comments on the analysis will be used as input to support decisions regarding the need for the NEPA process. The analysis will include a review of the impacts of the options during and after any remedial action and will cover the full scope of environmental concerns as well as radiological effects.

Step 3c, Evaluate Remedial Action and Waste Disposal Options - The engineering evaluation and environmental analysis produced in Steps 3a and 3b above will be evaluated by the DOE to identify the preferred option and reasonable alternatives. In this step, the DOE will advise the appropriate Federal, state, local agencies, and the public of the results of the preliminary engineering evaluation, the environmental analysis, and the DOE conclusions regarding the preferred option and reasonable alternatives. The DOE will seek their preliminary reviews and comments.

The risks, benefits, and costs of each remedial action and disposal option will be considered in the selection of the proposed remedial action. Factors affecting the remedial action, including environmental issues, technical issues, and public opinion, will be considered in the risk, benefit, and cost analyses. In selecting or proposing remedial action, emphasis will be given to determining the most practical and expedient means to eliminate or limit exposure to the public. If it is determined that material must be moved and no permanent disposal site is available at the time of the implementation of an action, the alternative of moving the contaminated material and stabilizing it at an interim storage site located at or near the contaminated site will be examined. It is assumed that the DOE will have ownership and maintenance responsibilities for all stabilized sites, interim storage sites, and permanent disposal sites except where the affected states agree to accept ownership and custodial responsibilities. It is also assumed that the stabilized sites, temporary storage sites, and the permanent disposal sites will be licensed by the NRC and will meet the relevant criteria of the proposed NRC regulations (basically modeled after 10 CFR 40 Appendix A). On the basis of this interagency and public review, the DOE will develop its proposals for remedial action and waste disposal options.

Step 4, Propose Remedial Action and Disposal Options - The remedial action and disposal option proposed by the DOE Secretary, and the reasonable alternatives will be identified and documented for the conduct of the NEPA process in Steps 5a, 5b, and 5c.

Step 5a, NEPA Process for Remedial Action - Onsite Stabilization - When the remedial action is proposed, the available data will be reviewed to determine if the proposed action is a major Federal action that will have a significant impact on the environment and what NEPA documentation is required. This review will also ensure that the data collected in the environmental analysis cover all environmental issues.

If required, the data developed during the environmental analysis step, along with any additional data required, will be used in the preparation of an environmental impact assessment (EIA) or an environmental impact statement (EIS). The NEPA documentation will be prepared as outlined in the CEQ NEPA Regulations (Title 40 CFR, Parts 1500-1508), the DOE NEPA guidelines (45 FR 20,594-20,701, March 28, 1980), and the DOE Order 5440.1. As noted in Figure 1 and discussed below, the NRC licensing process will be initiated in parallel with this step.

Step 5b, NEPA Process for Remedial Action - Offsite Disposal Options - In this step, the MED/AEC site and the candidate disposal sites that were identified in Step 3 by the affected state in consultation with the DOE, will be evaluated in parallel through the NEPA process to provide the basis for selecting the disposal site. The NEPA process will be conducted as outlined in the CEQ NEPA Regulations (Step 5a). As noted in Figure 1 and discussed below, the NRC licensing process will be initiated in parallel with this step.

Step 5c, Selected Remedial Action - At the conclusion of the NEPA process for both onsite remedial action or offsite disposal, the DOE will issue a Record of Decision announcing the selected remedial action and a decision as to how the radioactive materials will be permanently diposed.

The selection of the disposal site option will take into consideration the preliminary NRC licensing evaluation of the site, as appropriate.

Step 6, Remedial Action Engineering Plan - An engineering plan for the proposed action will be prepared, containing detailed plans and specifications for implementation of the selected remedial action alternative including, as appropriate, at the disposal site. The engineering plan will present detailed cost estimates, work plans, and schedules that define the engineering aspects of the remedial action and will be used to contract for the remedial action.

During this step, a license application for either stabilizing onsite or for offsite disposal will be prepared and submitted to the NRC.

Step 7, Implement Remedial Action and Monitoring - The remedial action contractor will conduct the action in accordance with the contract and as outlined in the engineering plan. Part of this step, where appropriate, will be the preparation of a disposal site. It will also include initiation of the operation, surveillance, and/or maintenance step that will continue as long as the site is used as a repository for these wastes. Independent monitoring by the DOE-ASEV will be conducted during the remedial action, and periodic status reports will be prepared.

Step 8, Certify Site Condition - During and upon completion of the remedial action, radiological surveys will be performed by the DOE-ASEV to verify the effectiveness of the remedial action, and the radiological condition of the site requiring remedial action will be documented. If the surveys verify that the levels of residual radioactive materials meet the established standards for unrestricted use, the site will be released for use without restrictions. If the surveys do not verify that the residual radioactivity meets the levels within the standards for unrestricted use, then further remedial action measures will be prescribed.

To assure control and enforcement of restrictions on "stabilized" sites, ownership by the Federal Government or the state will be required and the sites will be licensed by the NRC or the state. Disposal sites will be treated in a similar fashion. Such controls may permit some beneficial land use, such as making the area into a park where no permanent structures may be constructed, or possibly continuing the use of the site for other regulated nuclear activities. In any case, upon completion of the remedial action, a final report will be prepared documenting the entire remedial action effort and the radiological condition of the site. The final report will also note the quantity of material removed from the site and its disposition. The final report and all supporting documentation will be stored in permanent Federal Government

archives and copies or summary material will be placed in the records of appropriate local and state agencies and recorders' offices.

Status of Sites

As a result of the DOE efforts to identify the former MED/AEC sites, investigations to determine the radiological status of over 70 sites were or are being completed. Based on data collected to date, the DOE has determined that 18 sites will require some form of remedial action (as identified in Table 1) and 13 other sites are likely to require remedial action by the DOE.

Table 2 lists the 31 sites being considered and the current status of remedial action as of January 31, 1980. Figure 2 shows the location of these 31 sites. Radiological surveys of uniform character have been conducted at 20 sites, of which 19 reports have been issued in draft and 13 in final form. The remaining 11 sites have been surveyed with less rigor and will require more detailed surveys that are scheduled to be undertaken. Conceptual engineering evaluations have been initiated at five sites with final reports completed for two of those sites. Detailed engineering plans have been initiated at two sites. Remedial action has begun at a number of sites where there is existing DOE authority to conduct such actions. Implied authority for the undertaking of remedial action exists at 13 sites and must be clarified at 18 sites.

Appendix A to this document provides brief information summaries for each site.

6.0 Estimated Costs for Remedial Action Program

Preliminary cost estimates have been developed for remedial action for each MED/AEC site* and are summarized in Table 3, excluding those sites that are licensed by the NRC or Agreement States (Gilman Hall, University of Chicago, and Linde). These estimates are considered to be the upper bound of costs as explained below. Estimated costs for the remedial action program by work phase and by fiscal year are presented in Figure 3. Estimated costs of remedial action by site and by state are presented in Figure 4. The basis for the estimates are decontamination and restoration to unconditional public use using containers for waste transport, rather than bulk carriers and transportation of 500 miles to regional disposal sites.

Key Issues Affecting Costs and Schedule. Major factors influencing the cost of remedial action at the MED/AEC sites are:

- The option chosen for remedial action, either removal of contamination and restoration for unrestricted use by the public or permanent stabilization of existing contamination on the formerly utilized site to minimize exposure of the public with appropriate controls

- Criteria and standards for decontamination or stabilization

*"Formerly Utilized Sites Remedial Action Program - Preliminary Cost Estimates" prepared for USDOE Oak Ridge Operations Office Technical Services Division by Ford, Bacon & Davis Utah Inc., October 1979; and radiological survey, environmental monitoring, and certification cost estimates from the ASEV.

Table 1

MED/AEC SITES FOR WHICH A DETERMINATION HAS BEEN MADE THAT REMEDIAL ACTION IS REQUIRED*

Site	Health Priority
Ashland Oil Company, Tonawanda, New York	TBD
Bayo Canyon Area, Los Alamos, New Mexico	L
Clecon Metals, Inc., Cleveland, Ohio	M/H
Gilman Hall, University of California, Berkeley, California**	L
Conserv Inc., Nichols, Florida	L/M
E. I. du Pont de Nemours and Company, Deepwater, New Jersey	L
Gardinier, Inc., Tampa, Florida	M
Guterl Special Steel Corporation, Lockport, New York	L
Kellex Research Facility, Jersey City, New Jersey	H
Lake Ontario Ordnance Works Associated Properties, Lewiston, New York	TBD
Linde Air Products, Tonawanda, New York**	L
Mallinckrodt, Inc., St. Louis, Missouri	H
Middlesex Municipal Landfill, Middlesex, New Jersey	L
Middlesex Sampling Plant, Middlesex and Piscataway, New Jersey	H
Palos Park Forest Preserve, Cook County, Illinois	M
St. Louis Airport, St. Louis, Missouri	TBD
Seaway Industrial Park, Tonawanda, New York	TBD
Seneca Army Depot, Romulus, New York	L

Key: L = Low
M = Medium
H = High
TBD = To be determined

*Based upon DOE determinations completed through March 1980; determinations on 13 additional sites are in progress.

**Licensed by Agreement State provisions of the Atomic Energy Act of 1954, as amended, and excluded from FUSRAP; these licenses provide for site decontamination.

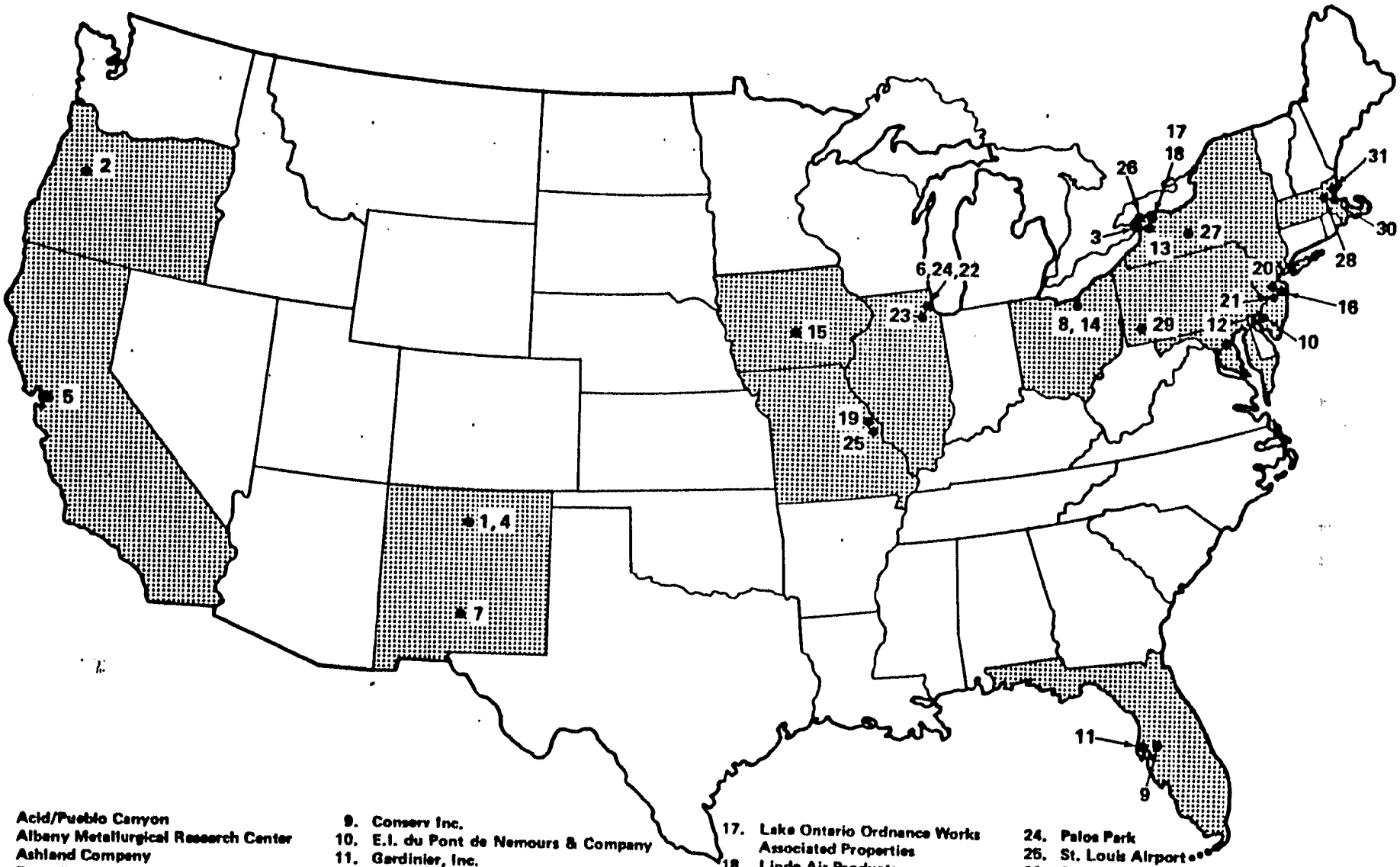
Table 2
Status of Remedial Action at MED/AEC Sites

Name	State	Radiological Survey	Report Draft	Report Final	Title I Eng'r'g Initiated	Title I Eng'r'g Draft	Title I Eng'r'g Final	Title II Eng'r'g Initiated	Title II Eng'r'g Draft	Title II Eng'r'g Final	EIS	Remedial Action Req'd	Remedial Action Underway	Remedial Action Authority Req'd	Remedial Action Authority Exists **
1 Acid Pueblo Canyon	NM				●										●
2 Albany Metallurgical Research Center	OR														●
3 Ashland Oil Company	NY	●	●	●								●		●	
4 Bayo Canyon	NM	●	●	●	●							●			●
5 Gilman Hall, University of California	CA	●										●			*
6 University of Chicago	IL														*
7 Chupadera Mesa	NM				●							✓		✓	●
8 Clecon Metals, Inc.	OH	●	●	●								●		●	
9 Conserv Inc.	FL	●	●	●								●		●	
10 E.I. du Pont de Nemours & Company	NJ	●	●	●								●			●
11 Gardinier, Inc.	FL	●	●									●		●	
12 W. R. Grace & Company	MD													***	
13 Guteri Steel Corp.	NY	●	●	●								●		●	
14 Harshaw Chemical Company	OH													● ✓	
15 Iowa State University	IA													●	
16 Kellex	NJ	●	●					●				●	●		●
17 Lake Ontario Ordnance Works Associated Properties	NY											●			●
18 Linde Air Products	NY	●	●	●								●			*
19 Mallinckrodt, Inc.	MO	●	●									●		●	
20 Middlesex Landfill	NJ	●	●		●	●	●					●			●
21 Middlesex Sampling Plant	NJ	●	●	●	●	●	●	●				●			●
22 National Guard Armory	IL	●	●											●	
23 Olin Chemical Company	IL	●	●	●										●	
24 Palos Park	IL	●	●	●								●		●	
25 St. Louis Airport	MO	●	●	●								●		●	
26 Seaway Industrial Park	NY	●	●	●								●		●	
27 Seneca Army Depot	NY	●	●	●								●			●
28 Shpack Landfill	MA													***	
29 Universal Cyclops, Inc.	PA	●	●											●	
30 Ventron Corporation	MA													●	
31 Watertown Arsenal	MA													●	
Totals		20	19	13	5	2	2	2				18	1	18	13

* These sites have been deleted from the FUSRAP because the NRC or Agreements States have sufficient licensing authority to protect public health and safety.

** Remedial Action authority exists where such action is required to protect public health and safety.

*** A determination as to whether additional authority is required to implement remedial action is currently underway.



1. Acid/Pueblo Canyon
2. Albany Metallurgical Research Center
3. Ashland Company
4. Bayo Canyon
5. University of California
6. University of Chicago
7. Chupadera Mesa
8. Clecon Metals, Inc.
9. Conserv Inc.
10. E.I. du Pont de Nemours & Company
11. Gardinier, Inc.
12. W. R. Grace & Company
13. Guteri Steel Corp.
14. Harshaw Chemical Company
15. Iowa State University
16. Kellex
17. Lake Ontario Ordnance Works Associated Properties
18. Linde Air Products
19. Mallinckrodt, Inc.
20. Middlesex Landfill
21. Middlesex Sampling Plant
22. National Guard Armory
23. Olin Chemical Company
24. Palos Park
25. St. Louis Airport
26. Seaway Industrial Park
27. Seneca Army Depot
28. Shpak Landfill
29. Universal Cyclops, Inc.
30. Ventron Corporation
31. Watertown Arsenal

Figure 2 Location of Sites Requiring or that May Require Remedial Action

Table 3
ESTIMATES OF REMEDIAL ACTION COSTS BY MED/AEC SITE*

Acid/Pueblo Canyon Area, Los Alamos, New Mexico	$ 1,900,000
Albany Metallurgical Research Center, Albany, Oregon	3,000,000
Ashland Oil Company, Tonawanda, New York	29,000,000
Bayo Canyon Area, Los Alamos, New Mexico	2,800,000
Chupadera Mesa Area, White Sands Missile Range, New Mexico	180,000
Clecon Metals, Inc., Cleveland, Ohio	2,400,000
Conserv Inc., Nichols, Florida	660,000
E. I. du Pont de Nemours and Company, Deepwater, New Jersey	3,000,000
Gardinier, Inc., Tampa, Florida	2,300,000
W. R. Grace & Company, Curtis Bay, Maryland	17,000,000
Guterl Special Steel Corporation, Lockport, New York	1,100,000
Harshaw Chemical Company, Cleveland, Ohio	9,000,000
Iowa State University, Ames, Iowa	570,000
Kellex Research Facility, Jersey City, New Jersey	1,400,000
Lake Ontario Ordnance Works Associated Properties, Lewiston, New York	3,000,000
Mallinckrodt, Inc., St. Louis, Missouri	26,000,000
Middlesex Municipal Landfill, Middlesex, New Jersey	50,000,000
Middlesex Sampling Plant, Middlesex and Piscataway, New Jersey	48,000,000
National Guard Armory, Chicago, Illinois	710,000
Olin Corporation, Joliet, Illinois	680,000
Palos Park Forest Preserve, Cook County, Illinois	7,100,000
St. Louis Airport, St. Louis, Missouri	98,000,000
Seaway Industrial Park, Tonawanda, New York	24,000,000
Seneca Army Depot, Romulus, New York	860,000
Shpack Landfill, Norton, Massachusetts	2,200,000
Universal Cyclops, Inc., Aliquippa, Pennsylvania	1,000,000
Ventron Corporation, Beverly, Massachusetts	880,000
Watertown Arsenal, Watertown, Massachusetts	630,000
	$338,000,000

*Upper boundary of costs for removal and disposal option.

Figure 3 Work Schedule and Funding Requirements for Remedial Action at MED/AEC Sites

Work Activities	Work Schedule by Fiscal Year										Estimated Costs
	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	
1. Radiological Characterization, Environmental Monitoring, Certification & Overview											$ 22,700,000
2. Engineering Analyses, Design, and Support											12,200,000
3. Environmental Analyses, Assessments, and Support											13,100,000
4. Decontamination and Residue Retrieval											49,000,000
5. Containerization of Residues											60,900,000
6. Transportation of Residues											48,000,000
7. Disposal of Residues											91,000,000
8. Contingency (15% of Sum of Items 2-7)											41,100,000
Total											$338,000,000

NOTE: Estimate based on retrieving containerizing, transporting, and disposal of an estimated 500,000 cubic yards of soil and rubble at the following average unit costs ($/cubic yard) in FY 1981 dollars:

	$/Cubic Yard
Contaminated residue retrieval	112
Containerization of residues	138
Transportation of residues (500 miles to regional disposal site)	110
Disposal of residues	208

Estimates of Annual Budget Authorization Requests in FY 1981 Dollars

Fiscal Year	Amount	Fiscal Year	Amount
1980	$ 9,400,000	1985	$ 46,200,000
1981	13,800,000	1986	48,000,000
1982	21,300,000	1987	50,800,000
1983	32,600,000	1988	42,600,000
1984	38,000,000	1989	35,300,000
			$338,000,000

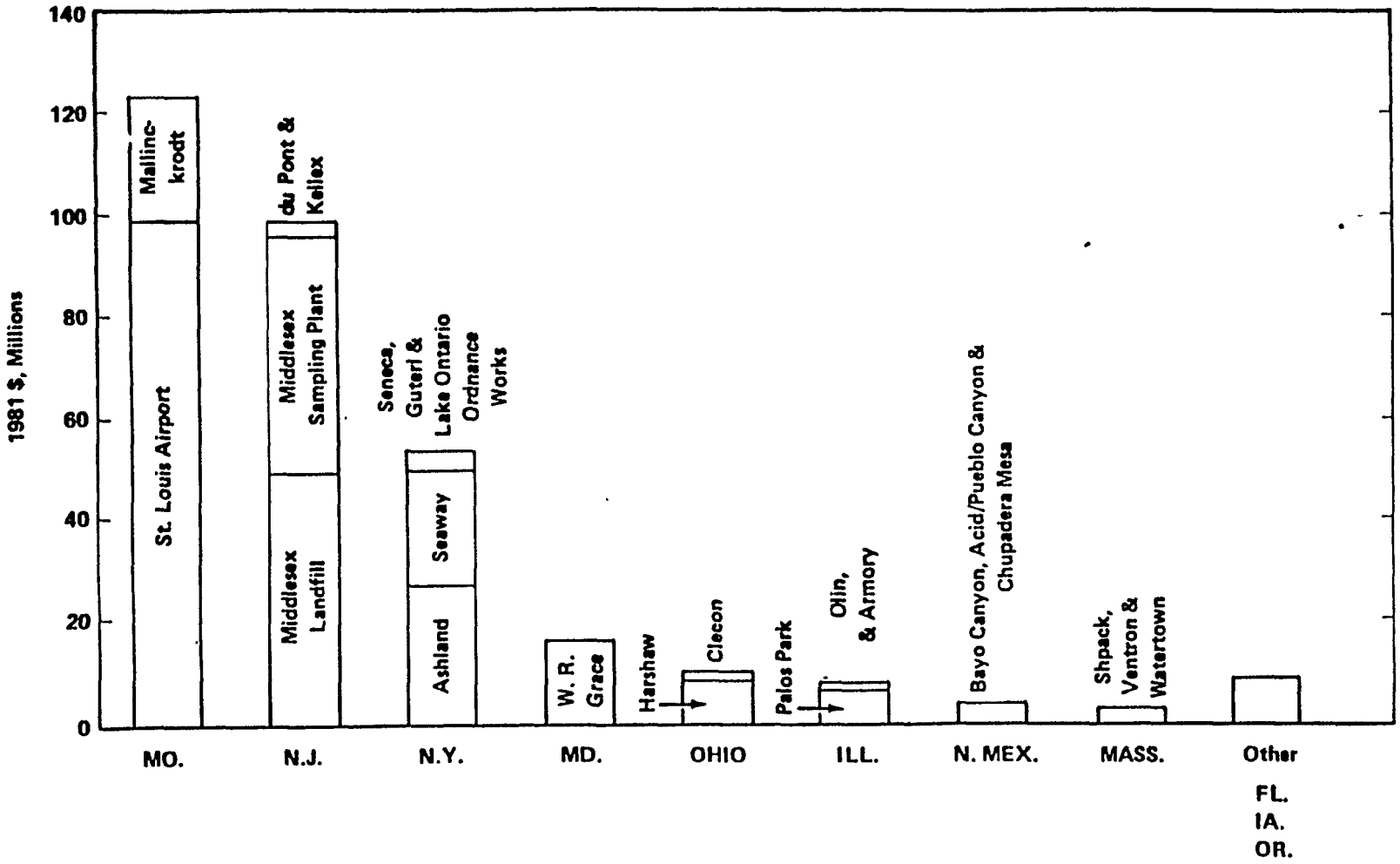


Figure 4. FUSRAP Costs by State and Site

- The method of packaging of materials for transport generated by decontamination, generally, either containerized or bulk

- Location of disposal site, either in-state or regional

- Type of disposal-site ownership (based on either government financing or commercial rates)

Remedial Action Options. Options available for remedial action at a contaminated site are either removal of contamination and restoration of the site to permit unrestricted public use, or permanent stabilization of the radioactive material on the remedial action site and restoration for restricted use. Because of the long time period required to locate and develop a disposal site, temporary remedial actions may be taken to reduce health impacts. Stabilization involves fixing of the contamination on the soil or structures such that transport offsite through such mechanisms as erosion, leaching into water supplies and aquifers, or through up-take in the biosphere does not occur and will not occur in the long term. Criteria and standards for stabilized sites will meet the intent of those criteria and standards used for the disposal sites, e.g., 10 CFR 40 Appendix A, the criteria proposed by NRC for privately owned mill tailing sites. Institutional controls have to be imposed at the stabilized site to prevent disturbance of the buried material and its subsequent release. Removal of contamination from structures, dismantling and removal of structures, and removal of soil and other contaminated material, followed by site restoration for unrestricted use by the public, is the most extensive remedial action that can be taken at a site. The costs for permanent stabilization might be a factor of 5 to 10 less than for decontamination and removal. For the purposes of providing a bounding cost of the proposed legislation, cost estimates were based upon decontamination of all the 29 MED/AEC sites and restoration for unlimited public use.

Criteria and Standards for Remedial Actions. The basis of the cost estimates provided for remedial action assumes contamination would be reduced to 5 picocuries of radium-226 per gram of soil (or comparable levels for other radionuclides), which is in the range of 2 to 10 times that of naturally occurring radium levels in the soil. If a lower value of acceptable contamination were to be imposed, substantially higher costs may result. For stabilized sites, another factor affecting cost is the depth of ground cover material that will be required by the NRC. In this cost estimate, no sites were considered for stabilization. Because the stabilization and disposal sites will be licensed by the NRC, the final criteria and standards established by the NRC will impact costs. The NRC has proposed criteria for licensed uranium mill tailings sites (10 CFR 40, Appendix A) and is developing criteria for large-volume, low-activity waste that are expected to be generally consistent with the mill tailings criteria. These criteria may be applied to the formerly utilized sites that are stabilized and to the disposal sites. In addition, the EPA has issued interim and proposed final criteria for remedial action at inactive mill tailings sites.

Method of Packaging. The packaging of contaminated material generated in the remedial action of decontaminating the MED/AEC sites can be accomplished either by use of containers such as 55-gallon drums, or bulk transporters such as large-volume trucks or railroad cars. The relative costs for the handling and transport of small containers is three to four times greater for the small containers versus bulk shipment. For the purposes of the proposed legislation, cost estimates were based upon containerization of waste residues.

Location of Disposal Sites. Transportation to a site for disposal of the contaminated material removed from the MED/AEC sites may be a significant factor in the cost of remedial action. The major factor in cost is the distance for transport of either containerized material or bulk quantities via truck or rail. Depending upon the location of the sites requiring decontamination and restoration, a suitable regional disposal site may be found that could satisfy the needs of more than one state. Cooperative efforts between states will be encouraged to jointly solve this common problem. The DOE will cooperate and support the states in this site selection activity. Cost estimates were based upon transportation costs associated with shipment of 500 miles to a regional disposal site.

Type of Disposal Sites. Sites for disposal of residues contaminated from the former MED/AEC use may be federally owned or state-owned. To ensure long-term institutional control of the disposal site, privately owned sites are not acceptable. This approach is consistent with that used in the Uranium Mill Tailings Radiation Control Act of 1978, and the criteria proposed by the NRC for the privately owned uranium mill tailings sites. Restriction of access to the site, and monitoring and surveillance requirements, will require administrative control that can be accomplished by either Federal or state ownership and custody of the site. Costs of operation of a disposal site for contaminated residues must reflect the quantities of wastes to be handled and the time period of active and passive controls.

These costs will be affected by whether the site is a single-use site or a multiple-use site.

APPENDIX A

MED/AEC SITE SUMMARY REPORTS

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SITE SUMMARY REPORTS

INTRODUCTION

The information contained in the following MED/AEC site summary reports represents the current knowledge of radiological conditions at, and former government use of, each site. In some cases, additional work necessary for complete characterization of a site is underway or planned.

Throughout the summary reports, reference is made to "current guidelines" for contamination and exposure levels. The guidelines discussed in section 3.0 Appendix A provides brief information on each site as follows:

- Owner history - from the MED/AEC period to the present
- Site location
- Site utilization during the MED/AEC period
- Use of site since the MED/AEC period
- Radiological history - results of surveys conducted and relative contamination levels
- Remedial action options and costs
- Project status - current status of surveys, engineering studies, recommendations for remedial action, and existing or implied authority for future remedial action.

ACID/PUEBLO CANYON AREA
LOS ALAMOS, NEW MEXICO

OWNER HISTORY

1943-1967: U.S. Government
1967-Present: Los Alamos County and U.S. Government (upper Canyon)

SITE LOCATION

Acid and Pueblo Canyons are located adjacent to the townsite of Los Alamos in north central New Mexico, about 25 miles northwest of Santa Fe. These canyons are two of many canyons cut into the Pajarito Plateau. Acid Canyon is a tributary of Pueblo Canyon.

MED/AEC SITE USE

These deep canyons were the discharge area for untreated radioactive liquid wastes between 1943 and 1951 resulting from research and processing at the Los Alamos Scientific Laboratory. Starting in 1951, treated radioactive effluents were discharged into the canyon from a liquid-waste-treatment facility which operated until 1964.

POST MED/AEC SITE USE

The area is unrestricted to public access and is used on a limited basis for recreational purposes.

RADIOLOGICAL HISTORY

Plutonium, americium, and fission products were discharged into the canyons in liquid effluents during the years 1943 to 1964. The first survey of Acid Canyon, for purposes of cleanup, was made on August 31, 1965. On October 4, 1966, work commenced on removing the waste-treatment-facility structures. Five-hundred truckloads of demolition debris and dirt from this location were removed. Ninety-four loads of debris from Acid Canyon were placed in a solid-waste disposal area within the currently operational Los Alamos Scientific Laboratory site. This decontamination activity included the removal of all drain pipes, wires, rocks, tuff, and other debris found contaminated in Acid and Pueblo Canyons. This work was completed in 1967, and it was reported that a small amount of contamination remains in inaccessible places.

In November 1973, it was reported that plutonium concentrations in filtered surface waters in Acid Canyon and the adjacent portions of Pueblo Canyon generally averaged about 20 picocuries/liter. A limited number of samples of the alluvium taken in 1970 indicated plutonium concentrations of 27 picocuries/gram in lower Acid Canyon, 4.6 picocuries/gram in Pueblo Canyon 1 mile below the Acid Canyon outlet, and 1.1 picocuries/gram 2 miles below Acid Canyon.

Some radiological and environmental surveillance evaluations have been completed and documented for Pueblo Canyon. Several hundred soil and sediment samples were collected for the present detailed radiological survey during 1977. Data show some limited areas in the canyons that exceed the EPA-proposed soil screening guides for plutonium concentrations. Measurements of penetrating radiation showed no areas that exceed radiation protection standards.

REMEDIAL ACTION OPTIONS AND COSTS

Some form of remedial action may be required and could include stabilization and/or decontamination by excavation of the cliff face, outfall area, cliff base and channel, and the Acid Canyon stream bed. Seventeen-hundred cubic yards of contaminated material would be produced. The estimated cost is $1,900,000.

PROJECT STATUS

Following the completion of the radiological survey report, the Assistant Secretary for Environment will determine whether the site requires remedial action. Work has been initiated on an Engineering Evaluation Report-Title I. Authority to implement a remedial action exists under the Atomic Energy Act of 1954, as amended.

ALBANY METALLURGICAL RESEARCH CENTER
ALBANY, OREGON

OWNER HISTORY

The site has been and is currently owned by the Bureau of Mines, U.S. Department of the Interior.

SITE LOCATION

The site is located in Albany, Oregon, approximately 23 miles south of Salem. Eight buildings and their surroundings were used for former MED/AEC activities.

MED/AEC SITE USE

From 1954 to 1971, the Albany Metallurgical Research Center was engaged in metallurgical operations involving thorium. Operations included reduction, melting, machining, welding, and alloying. Research on alloys of uranium and thorium started in 1955 and continued to 1978.

POST MED/AEC SITE USE

Research involving uranium and thorium was suspended in 1978. Onsite areas that contain contaminated soils have been fenced to restrict access. None of the buildings are currently used for uranium or thorium alloy research.

RADIOLOGICAL HISTORY

At the time that the original AEC contract was terminated (approximately 1960), these buildings were decontaminated according to the general guidelines provided by the AEC to the Bureau of Mines. These guidelines were not as specific as later guidelines, and there is no record that the final decontamination was documented. Contaminated materials, equipment, or wastes generated under the AEC contracts were removed from the site for disposal.

The Argonne National Laboratory (ANL) conducted a radiological survey of these buildings and grounds in 1978 and found contamination that exceeded current guidelines for unrestricted use still existed on surfaces and that some areas of soil were contaminated with uranium and thorium. As an interim measure, the Bureau of Mines has fenced in areas of contaminated soil to restrict access. Some additional survey work, including an aerial radiometric survey, was conducted in 1979, and some subsurface investigations are scheduled for 1980. ANL is preparing a radiological survey report to document all survey activities.

No significant public health impact exists due to restricted use of the contaminated areas; however, potential health impacts could result if usage was changed. Interim access control measures have been employed.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of contaminated soils, decontamination of buildings and removal of structural elements and plumbing. Thirty-seven-hundred cubic yards of contaminated material could be produced. The estimated cost for remedial action is $3,000,000.

PROJECT STATUS

A radiological survey has been completed and a final report is in preparation. Upon completion of this report, the Assistant Secretary for Environment will determine whether remedial action is required. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

ASHLAND OIL COMPANY
TONAWANDA, NEW YORK

OWNER HISTORY

1943-1944:	E. Haist et al. - leased by MED
1944-1960:	U.S. Government
1960-Present:	Ashland Oil Company

SITE LOCATION

The 10-acre site is located in a large industrial area in Tonawanda, New York. It is adjacent to the Seaway Industrial Park, another formerly utilized MED/AEC site.

MED/AEC SITE USE

From 1943 to 1946, the site was used for disposal of uranium-processing residues from the Linde Air Products Division-Union Carbide Corporation ore refinery operations. Eight-thousand tons of residue containing approximately 0.54 percent uranium were spread over two-thirds of the site to a depth of 1 to 5 feet.

POST MED/AEC SITE USE

In 1974, 6,000 cubic yards of residue were removed by Ashland and transported to the adjacent Seaway Industrial Park. The site was developed as an oil storage site at that time.

RADIOLOGICAL HISTORY

A radiological survey was conducted in 1958. Following this survey, the property was released for unrestricted use without removal of the residues. A detailed survey was conducted under the FUSRAP during July and August 1976. An aerial survey was conducted in September 1979.

Results of the 1976 survey indicated that external gamma radiation exceeded applicable guidelines over fairly large areas of the site. However, the results indicated that the residues on the site "do not pose an immediate health hazard, assuming that residues remain in place and that the site continues to be used in the manner in which it is presently used." The radon daughter concentration in the onsite building is close to background level, and only small quantities of radium or uranium are carried from the site in surface runoff. Because the property is located in an industrial area, the population density surrounding the site is very low, and thus there are few people at risk. If the site use were changed and buildings constructed onsite, there could be an increase in exposure and a potential health hazard could result.

REMEDIAL ACTION OPTION AND COSTS

Remedial action is indicated and could involve removal of approximately 48,000 cubic yards of residues and contaminated soil. The estimated cost for this remedial action is $29,000,000.

PROJECT STATUS

A radiological survey was completed in August 1976; a final report was issued in May 1978. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority to implement remedial action will be required.

BAYO CANYON AREA
LOS ALAMOS, NEW MEXICO

OWNER HISTORY

1944-1967: U.S. Government
1967-Present: Los Alamos County

SITE LOCATION

Bayo Canyon is located adjacent to the townsite of Los Alamos in north central New Mexico, about 25 miles northwest of Santa Fe. Bayo Canyon is one of many canyons cut into the Pajarito Plateau.

MED/AEC SITE USE

Experiments with high explosives were conducted in Bayo Canyon during the period 1944 through 1961. The explosive test assemblies included natural and depleted uranium and lanthanum-140, which was used as a tracer. Strontium-90 was also present as a contaminant of the lanthanum-140. The site facilities include radiochemistry laboratories, radioactive liquid-waste disposal facilities, and solid-waste disposal facilities.

POST MED/AEC SITE USE

The site was decommissioned in 1963. Since 1967 the canyon has been used exclusively for recreational purposes, including picnicking, trail riding, hiking, wood cutting, and pinon nut gathering. Proposed uses include residential and light commercial development.

RADIOLOGICAL HISTORY

From 1949 through 1969, 1.355 curies of natural uranium, 1.218 curies of depleted uranium, and between 30 and 40 curies of strontium-90 were dispersed into the surface environment of the Bayo Canyon area. An additional 85 to 120 curies of strontium-90 were deposited in waste-handling facilities and some fraction migrated into the subsurface environment. Most of the activity was associated with debris that was removed in 1963, leaving a comparatively small amount of radioactivity at the surface of the site and in subsurface layers of soil. A radiological survey was conducted under the FUSRAP in 1977.

The results of this survey show that exposure of current nearby residents to airborne strontium-90 and uranium is no different than that of other northern New Mexico residents. However, dose estimates for construction workers if the area were to be developed indicate exposure levels at less than 1.5 percent of DOE guidelines. The estimated exposure of residents in the developed area would be, at most, 3 percent of DOE guidelines. Individuals presently using the area for recreational purposes receive somewhat lower exposures because of the shorter exposure period and minimal interaction with disturbed soil.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could take the form of stabilization of dispersed radioactivity with restrictive control over change in site use or decontamination by excavation of soil to remove radioactivity. If decontamination is performed, 3,500 cubic yards of contaminated material will be produced. The estimated cost to perform this remedial action is $2,800,000.

PROJECT STATUS

A radiological survey was completed in 1977; the final report was issued in June 1979. The Assistant Secretary for Environment has determined that the site will require remedial action. Preparation of an Engineering Evaluation Report-Title I, has been initiated. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

GILMAN HALL
UNIVERSITY OF CALIFORNIA
BERKELEY, CALIFORNIA

OWNER HISTORY

University of California

SITE LOCATION

The site is located on the Berkeley Campus of the University of California and consists of the third floor and basement of Gilman Hall.

MED/AEC SITE USE

Laboratory facilities in Gilman Hall were used in support of the Manhattan Project and/or early AEC activities. It is believed that weapons-grade plutonium was involved.

POST MED/AEC SITE USE

A preliminary radiological survey was completed by Lawrence Livermore Laboratory and a letter report issued in 1976. The survey was designed to document alpha contamination. However, evidence of significant cesium-137 was also found.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could take either one of two forms. The area could be left as is but placed under control, which would require that any future renovation and/or demolition work be performed under contamination removal and control procedures. This may require a license.

Alternatively, the area would be decontaminated by stripping away floor tile, sand blasting concrete surfaces, and removing piping. Thirty cubic yards of contaminated material would be produced. Estimated cost for this remedial action is $483,000.

PROJECT STATUS

A preliminary radiological survey was conducted in 1976. A detailed survey will be initiated soon. The Assistant Secretary for Environment has determined that remedial action is required. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

UNIVERSITY OF CHICAGO
CHICAGO, ILLINOIS

OWNER HISTORY

The site is owned by the University of Chicago.

SITE LOCATION

The University of Chicago buildings associated with the MED work were the New Chemistry Lab and Annex, West Stands, Ryerson Physical Lab, Eckhart Hall, Kent Chemistry Lab, Jones Lab, Ricketts Lab, and an area known as Animal Quarters. A comprehensive information search could not verify the location or even the existence of the Animal Quarters.

MED/AEC SITE USE

The University was the site of the first successful nuclear pile and it conducted associated research required for the production of plutonium and ultimately the atomic bomb. Research was conducted under the MED and the AEC during the 1940s and 1950s.

POST MED/AEC SITE USE

The New Chemistry Lab and Annex, the West Stands, and Ricketts Lab have been torn down. The remaining buildings are currently in use as offices, laboratories, and classrooms. Some of the laboratories are still being used for nuclear research and are under license by the NRC.

RADIOLOGICAL HISTORY

References indicate that all of the buildings were decontaminated prior to release; however, some documentation is missing and some was inadvertently destroyed. Radiological surveys were performed during the period September 1976 to September 1977 under the FUSRAP.

Results of the 1976-1977 surveys indicate that contamination is widespread throughout the laboratories but at fairly low levels except for isolated small areas. Analysis of potential exposure conditions indicate that persons will not receive exposures exceeding current guidelines under present usage. However, remodeling or demolition activities could free fixed contamination resulting in a potential health hazard. Soil samples indicate contamination is confined to the buildings.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve decontamination of the buildings involved. Seventy-five cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $630,000.

PROJECT STATUS

A radiological survey was completed in September 1977; a draft report has been issued for review. Upon issuance of the final report, the Assistant Secretary for Environment will make a determination as to whether remedial action is required. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

However, as the University campus is under license by the NRC, this site would not be decontaminated under the FUSRAP program since the NRC has sufficient licensing authority to protect public health and safety.

CHUPADERA MESA AREA
WHITE SANDS MISSILE RANGE, NEW MEXICO

OWNER HISTORY

The site was and continues to be private lands with multiple ownership.

SITE LOCATION

The site is located approximately 70 miles southeast of Albuquerque, New Mexico, and immediately north of the White Sands Missile Range.

MED/AEC SITE USE

The site area received fallout from an atomic bomb test at Trinity site in 1945.

POST MED/AEC SITE USE

Chupadera Mesa is extensively used as grazing land. In the northern area, the land is used primarily for growing alfalfa and assorted row crops.

RADIOLOGICAL HISTORY

The University of California, Los Angeles, conducted the first contamination survey in the 1947 to 1950 period. Thousands of soil and biological samples were obtained. Subsequently, in the 1972 to 1976 period, the Los Alamos Scientific Laboratory (LASL) collected similar samples. In 1977, LASL collected additional data around Trinity ground zero and the outlying fallout zones. The existing data are being evaluated and a radiological survey report is currently being prepared.

REMEDIAL ACTION OPTIONS AND COSTS

It is expected that some stabilization of contamination may be required. The estimated cost is $180,000.

PROJECT STATUS

Following the completion of the radiological survey report, the Assistant Secretary for Environment will determine whether the site requires remedial action. Work on an Engineering Evaluation Report-Title I has been initiated. Authority to implement a remedial action exists under the Atomic Energy Act of 1954, as amended.

CLECON METALS, INC.
CLEVELAND, OHIO

OWNER HISTORY

MED/AEC utilization period:	Horizons, Inc.
Present:	Clecon Metals, Inc.

SITE LOCATION

The site, encompassing approximately 3.5 acres, is located within Cleveland, Ohio, in a primarily industrial area which is sparsely populated. Two of three buildings on the site were used for processing radioactive materials.

MED/AEC SITE USE

During the 1940s and 1950s, two buildings at the Horizons metal-handling facility were used for the production of granular thorium metal. The feed material, thorium nitrate tetrahydrate, was processed through a number of steps and ultimately converted to thorium metal by use of an electrolytic process.

POST MED/AEC SITE USE

The plant site is currently used for the production of gaskets and for the lamination of various materials. The buildings were formerly used for processing radioactive materials, for receiving and storing nonradioactive materials, and for office space. Approximately 60 workers use these buildings.

RADIOLOGICAL HISTORY

In December 1954, the Health and Safety Laboratory performed an air hygiene survey that revealed airborne concentrations of thorium in both buildings to be 18 to 377 times greater than the applicable guideline. A subsequent survey indicated that the contamination was either removed or covered due to construction modifications made since the thorium operations. A radiological survey was conducted under the FUSRAP during February and March 1977.

Results of the 1977 survey indicate alpha, beta, and gamma levels in excess of current guidelines in several areas of both buildings. Contamination is located mainly in storage areas, drains and under floors. Exposure is limited to a few persons for short time periods. If use of buildings changes, doses of 0.2 to 0.4 rem/year could occur.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated, and could include decontamination of building surfaces, removal of some structural elements, removal of portions of the pumping system, and excavation of soil. An estimated 800 cubic yards of contaminated material would be produced. The estimated cost for remedial action is $2,400,000.

PROJECT STATUS

A radiological survey was conducted in February and March 1977. The final report was issued in February 1979. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority for the ASNE to implement remedial action is required.

CONSERV INC.
NICHOLS, FLORIDA

OWNER HISTORY

1952-1960:	Virginia-Carolina Chemical Corporation
1960- :	Unidentified - changed ownership 3 times
Present:	Conserv Inc.

SITE LOCATION

The site is located at Nichols, Florida, approximately 22 miles east of Tampa. The area involved with radioactive materials is approximately 0.5 acres.

MED/AEC SITE USE

Starting in 1952, a pilot plant was operated for the recovery of uranium from wet-process-produced phosphoric acid. This plant was disassembled in 1960. Location of equipment, tanks, piping, and building materials is unknown.

POST MED/AEC SITE USE

1961-1968:	Phosphoric acid and other phosphate product production
1969-1973:	Plant shut down
1974-Present:	Phosphoric acid and other phosphate product production. The site of the former recovery plant is currently used for storage and contains a building that houses a maintenance shop, lunchroom, tool storage cage, and a small office. This building is built on the concrete pad of the former recovery plant.

RADIOLOGICAL HISTORY

A preliminary radiological survey, conducted in April 1977, indicated alpha, beta, and gamma contamination of the concrete pad of the former recovery plant and uranium-238 and radium-226 contamination of nearby soil. Soon after the survey, the plant operator removed approximately 4 cubic yards of contaminated soil. The soil was buried in an inactive gypsum pile located about 2,600 feet from original site and covered with 2 to 3 feet of gypsum and soil. A detailed radiological survey was conducted under the FUSRAP during December 1977.

Results of the December 1977 survey indicate contamination is primarily located in the soil around the concrete pad, on the pad outside the building, and in the area where contaminated soil was dumped. It should be noted that present site activities dealing with phosphate product production contribute significantly to elevated radiation levels at the plant site. In many areas of the plant site, the levels are unrelated to the former MED/AEC activities. No significant health hazard currently exists, principally because of infrequent occupancy. However, if the site use were changed to crop production or if a new building were constructed over the areas of higher contamination, exposures exceeding the guidelines could result.

REMEDIAL ACTION OPTIONS AND COST

Remedial action is indicated and could involve excavation of contaminated soils near the concrete pad and in the area of dumping of previously excavated soil. Cleaning and/or removal of the concrete pad may be required. One-hundred-thirty cubic yards

of contaminated material would be produced. The estimated cost for this remedial action is $660,000.

PROJECT STATUS

A radiological survey was completed during December 1977; the final report was issued in February 1979. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority to implement remedial action will be required.

E. I. du PONT de NEMOURS AND COMPANY - CHAMBERS WORKS
DEEPWATER, NEW JERSEY

OWNER HISTORY

The site is owned and operated by the E. I. du Pont de Nemours Company.

SITE LOCATION

The 700-acre Chambers Works site is located adjacent to the residential communities of Deepwater, Pennsville, and Penns Grove, New Jersey. Within this site, operations involving MED/AEC activities were confined to four locations. These were three buildings and a radioactive material burial facility.

MED/AEC SITE USE

The du Pont operations for the MED included development of a process for converting uranium oxide to uranium tetrafluoride, production of uranium peroxide from the MED scraps, production of uranium tetrafluoride, uranium metal, uranium hexafluoride, and various related research activities. Such activities took place during the period 1942 through 1947. Decontamination and radiological survey activities took place during 1948. The last portion of the site used for the MED was released to du Pont in December 1948.

POST MED/AEC SITE USE

Of the three buildings involved in the MED activities, two have been demolished and one is still in use as a warehouse. A parking lot has been constructed on the site of one of the demolished buildings and a new building constructed at the site of the other. The radioactive material burial facility, which is approved by the State of New Jersey, possibly contains a few pieces of equipment from the demolished buildings.

RADIOLOGICAL HISTORY

In 1948, all contaminated equipment was removed from the site. Building decontamination, conducted under the direction of the AEC, included sandblasting, vacuuming, and washing of all building surfaces. A radiation survey was made by the Health Division of the AEC and the buildings were subseqently released to du Pont. A radiological survey was conducted under the FUSRAP during March 1977.

Results of the 1977 survey indicate that elevated concentrations of uranium were found in residues from the operations building and in some surface and subsurface soil samples. Alpha and beta-gamma contamination levels in some areas of the operations buildings were above the limits of current Federal guidelines. Under current conditions of site use, this contamination does not cause employees working at the site to receive radiation exposures appreciably different from those due to background radiation. However, under different conditions of use (i.e., use of contaminated soils for growing crops or actions which involve agitation or abrasion of dry contaminated surfaces), potential radiation exposures to employees and the public could result.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve decontamination of building surfaces and excavation of soil. Twenty-seven-hundred cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $3,000,000.

PROJECT STATUS

A radiological survey was completed in March 1977; the final report was issued in December 1978. The Assistant Secretary for Environment has determined that the site will require remedial action. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

GARDINIER INCORPORATED
TAMPA, FLORIDA

OWNER HISTORY

1951-1962:	Tennessee Corporation, U.S. Phosphoric Products Division
1963-1973:	Cities Service Company
1974-Present:	Gardinier, Incorporated

SITE LOCATION

The formerly utilized site, consisting of approximately 1.5 acres, is located within the Gardinier phosphoric acid production plant boundaries in Tampa, Florida.

MED/AEC SITE USE

During the period 1951 to 1960, Tennessee Corporation extracted uranium from phosphoric acid. This process consisted of (1) pretreatment of wet-process phosphoric acid, (2) solvent extraction of uranium, (3) precipitation of the uranium product, (4) drying and crushing, and (5) handling, packaging, and shipping. Pilot operations were carried out from 1951 through 1954 and the process plant was operated from 1956 through 1960.

POST MED/AEC SITE USE

A three-story building which housed the process plant is currently used as a workshop, lunchroom, office space, and as a storage area for equipment remaining from the uranium-recovery operations. A former pilot plant building is currently used as office space. Approximately 30 employees use these buildings. A new uranium recovery pilot operation is conducted on the site, which operation is currently licensed by the State of Florida. This license does not cover the MED/AEC material.

RADIOLOGICAL HISTORY

A radiation survey was conducted under the FUSRAP by Oak Ridge National Laboratory during December 1977. Some contaminated equipment was removed following the survey and transported to a licensed site.

Results of the 1977 survey indicate only slight contamination of the former pilot plant building, significant contamination of the former process building, and significant contamination of adjacent outdoor areas. Various measurements of alpha, beta, and gamma activity exceed current guidelines throughout the former process building. Highest levels of contamination were found on the second floor and are associated with stored equipment which was used in the uranium recovery process. External gamma levels measured outdoors also exceed guidelines and appear to be associated with radium-226, which has plated out in buried pipes and vessels.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve removal of stored equipment, excavation of soil and buried pipes and tanks, and decontamination of structures. Two-thousand cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $2,300,000.

PROJECT STATUS

A radiological survey was performed in December 1977; a draft of the final report is currently under review. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority is needed for the implementation of remedial action.

W. R. GRACE & COMPANY
CURTIS BAY, MARYLAND

OWNER HISTORY

This was and continues to be private land under the ownership of W. R. Grace & Company.

SITE LOCATION

The site consists of 4 acres of land at the Davison Division of W. R. Grace & Company at Curtis Bay, Maryland.

MED/AEC SITE USE

In late 1956 and early 1957, W. R. Grace assumed the license and contract of Rare Earths, Inc., to process, transfer, and use the radioactive material thorium. The thorium was shipped to Davison as a component of monazite sand. Title to the monazite and the thorium remained with the government during the performance of the work. The monazite sand was processed to remove the thorium which was shipped to GSA. Residue from the process was collected in dumpsters and emptied in a designated area of the onsite dump. The processing plant was never completed and the project was abandoned in 1957.

POST MED/AEC SITE USE

The site is presently unoccupied, untraversed, remote, and within the fenced enclosure surrounding the entire plant but not separately enclosed.

RADIOLOGICAL HISTORY

Radiation Management Corporation conducted a survey in 1978 to measure external radiation levels and investigate the possible migration of radioactive material from the deposit site.

It is estimated that the total volume of waste material possibly contaminated with monazite residue is 504,000 cubic feet in one location and 200,000 cubic feet in a second. There is no apparent indication of migration from the burial area. It is unclear whether or not the waste material exceeds 0.05 percent ThO_2. Surface radiation levels ranged from background levels to 17 mr/hr. Analysis of plant material indicated no detectable thorium daughter products. Core samples indicated thorium concentrations of 6.2 ± 0.9 pCi/gm at a depth of 5 feet and 97 ± 10 pCi/gm at 15 feet. The results assumed thorium in equilibrium with its daughters. Institutional control measures have been instituted to limit access to the disposal site.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of contaminated soils and restoration. An estimated 26,000 cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $17,000,000.

PROJECT STATUS

A detailed radiological survey is scheduled for 1980. Upon completion of this survey, the Assistant Secretary for Environment will determine if remedial action is required. Determination of whether additional authority is required to implement remedial action is currently underway.

GUTERL SPECIAL STEEL CORPORATION
LOCKPORT, NEW YORK

OWNER HISTORY

MED/AEC utilization period:	Simonds Saw & Steel Company
Present:	Guterl Special Steel Corporation, Simonds Steel Division

SITE LOCATION

The plant site is located in an industrial area of Lockport, New York. The formerly utilized site consists of the rolling mill building, the forging shop building, and the area immediately surrounding these buildings. The area involved is approximately 4 acres.

MED/AEC SITE USE

1948-1956	Rolling mill operations of uranium and thorium metal; operations included weighing, heating, rolling, shearing, and quenching.

POST MED/AEC SITE USE

1957-Present:	Rolling mill operations of nonradioactive metals; approximately 50 persons currently work in the buildings formerly involved with radioactive materials.

RADIOLOGICAL HISTORY

During all operations from 1948 through 1956, the AEC was responsible for radiological monitoring and safety. Residue from the operation was returned to the AEC or National Lead of Ohio. Protective measures included the use of hoods and dust-collection equipment over the 16-inch rolling mill stands and pans in the mill pits to collect material. A radiological survey performed during November 1958 indicated highest radiation levels in the quench tank area. Decontamination was performed and consisted of removing the quench tank, covering this area with steel plate, and washing and vacuuming other areas. A resurvey was conducted in December 1958 to verify decontamination actions. A radiological survey was conducted under the FUSRAP during October 1976.

Results of the 1976 survey indicate that only small accessible areas of contamination in the rolling mill building exceed present exposure guidelines. Other areas, particularly the former quench tank, have significantly high contamination levels but do not presently contribute greatly to exposure because of inplace shielding in the form of steel plates. Under current conditions of site use, this contamination does not cause employees working at the site to receive radiation exposure appreciably different than those due to background. However, under different conditions of site use (i.e., removal of steel plates, disturbance of soil or soil floors in buildings), potential exposure to employees and the public could result.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve excavation of outdoor soil, indoor soil floors, removal of some equipment, and cleaning of structures. Three-hundred-fifty cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $1,100,000.

PROJECT STATUS

A radiological survey was completed during October 1976; the final report was issued in November 1979. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority to implement remedial action will be required.

HARSHAW CHEMICAL COMPANY
CLEVELAND, OHIO

OWNER HISTORY

The site has been and is currently owned by Harshaw Chemical Company.

SITE LOCATION

The site is located within Cleveland, Ohio, and consists of three buildings and surrounding areas.

MED/AEC SITE USE

In September 1942, the MED contracted with Harshaw for the production of green salt (UF_4). This work was a continuation of smaller scale work performed for the Office of Scientific Research and Development. In 1943, Harshaw also began production of uranium hexafluoride, an operation that was substantially expanded in 1947. Another MED/AEC contract involved the production of uranium tetrachloride and uranium oxyfluoride. Building G1 (Plant C) was used for the UF_6 production and the foundry building was used for the UF_4 production. Analytical work was performed in building K1. Equipment and material from the MED/AEC operations was apparently stored in those and other buildings at the site. In 1960, the facility was released to the Harshaw Chemical Company from AEC control.

POST MED/AEC SITE USE

Building G1 is presently being used primarily as a storage warehouse, but it does contain some chemical production operations including the drying of fluorspar. The building is normally occupied by fewer than 10 people and contains a locker room area on the second floor which is used by employees working at another building on the Harshaw site. Additional personnel are present only during use of the locker room and transfer of material in and out of storage. A 60- by 200-foot addition was constructed on the north side of the building after the MED/AEC use of the facility was terminated. This addition is used for storing fluorspar.

RADIOLOGICAL HISTORY

This site was visited by the AEC personnel on October 27 and 28, 1953, to survey the equipment and buildings for contamination and to provide the necessary actions prior to the return of the building to the contractor. A meeting with representatives from the Harshaw Chemical Company was held, and a decontamination program was agreed to. The actions taken as a result of this visit are unknown.

Another survey was conducted on November 21, 1957, by the Research and Development Division, Oak Ridge, Tennessee. The purpose of this survey was to locate any areas where residual contamination was of such magnitude that it might represent a potential radiation or contamination control problem that would require the imposition of restrictions on the use of the building. At the time of this survey, all equipment had been removed except for the Rockwell furnace, two denitration pots, and some process vessels in the recovery area. The report of this survey identified contaminated areas with recommended methods for decontamination. A supplemental agreement assigned the responsibility to the contractor for decontaminating all equipment transferred to it and for decontaminating its own premises used in the performance of the contract. Further, the decontamination effort was to be

accomplished in accordance with the recommendations contained in the report of survey. The building was released from further AEC control in 1960.

A radiation survey of the building at Harshaw was performed in May 1976 by the Chicago Operations Office to identify previously utilized MED/AEC sites. During this survey, three soil samples were taken in the area adjacent to the building. These soil samples showed readings greater than normally expected. A draft of the radiation survey report was furnished to the Harshaw Chemical Company on July 8, 1976. The results of the survey showed residual contamination remained at the building.

Soil corings were taken by the Argonne National Laboratory at selected locations around the Harshaw complex on November 10, 1976. A draft of this soil survey report was transmitted to the DOE Headquarters with a recommendation that the survey be extended. The DOE Headquarters concurred with the recommendations, and additional survey work was accomplished between August and September 1979, including an annual radiometric survey. Preliminary results indicate that there is general deposition of contamination throughout the site and it may extend beyond the Harshaw site boundary.

Based on the completed preliminary surveys, the contamination is at an acceptable level and does not represent a hazard to Harshaw personnel. However, if modifications, remodeling, cleanup, or other structural changes were to be undertaken, radioactive material now fixed in the structure could be released and lead to airborne contamination. Harshaw has indicated that they would contact the DOE prior to any such actions. Likewise, no health hazard is envisioned from the contaminated soil in its present status.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of soil, decontamination of the building, and excavation of a portion of the Cuyahoga River. Ninety-two-hundred cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $9,000,000.

PROJECT STATUS

Upon completion of the currently initiated radiological survey, the Assistant Secretary for Environment will determine whether remedial action will be required. Additional authority to implement remedial action is required.

IOWA STATE UNIVERSITY
AMES, IOWA

OWNER HISTORY

The site has been and is currently owned by Iowa State University. Additional areas that have become contaminated by activities at the University site are owned by the Municipality of Ames, Iowa.

SITE LOCATION

Four buildings on the University campus at Ames were used for the MED/AEC activities. Three additional areas have become involved because of disposal of contaminated sewage sludge. The areas are the Ames Iowa Municipal Airport, the Grand Avenue underpass, and the Ames Municipal Cemetery.

MED/AEC SITE USE

Early MED/AEC activities were concerned with metallurgical research, fundamental chemical and analytical research, and the development of processes to produce pure uranium and other materials. During the 1942 period, the small-scale production in the physical chemistry laboratory furnished about 2 tons of uranium for use as heart metal in the first chain-reacting pile in Chicago. About 2 million pounds of virgin uranium were produced up to January 1, 1945, at which time production at Ames was discontinued. A recovery process developed at Ames resulted in the recovery of over 600,000 pounds of metal from scrap supplied by all of the MED sites. This operation was discontinued in December 1945. In 1947, the project at Ames was declared a major research facility and a program to produce thorium metal was initiated. Prior to 1947, approximately 4,500 pounds of thorium had been produced. Approximately 65 tons were produced in total.

RADIOLOGICAL HISTORY

Between July 1951 and August 1952, filtrates containing thorium and mesothorium were released into the sewage lines. Water-removal operations at the Water Pollution Control Plant produced a dry sludge cake that contained much of the released thorium and mesothorium (less than 1 curie). This sludge cake was collected and held at the west end of the drying beds at the Water Pollution Control Plant. In accordance with AEC recommendations, the sewage sludge cake containing mesothorium was placed on the City of Ames Municipal Airport grass runway, the Municipal Cemetery, and the grass areas of the Grand Avenue underpass.

An initial radiation survey was conducted on May 12, 1976, at the Municipal Airport of Ames, the Municipal Cemetery, the Grand Avenue underpass, and the site of buildings on the Iowa State University campus. Based on preliminary results of this survey and subsequent surveys, minor contamination of some land does exist. The Municipal Cemetery and the Grand Avenue underpass show no significant contamination. There was no discernible radiation different from the background level at the sites of Chemistry Annexes I and II. A single area in a taxi strip at the Municipal Airport shows some thorium contamination. The area west of the sludge beds at the Water Pollution Control Plant shows thorium contamination in a "ditch" area (approximately 6 times background) and a more generalized area (up to 2 times background).

None of the areas surveyed have contamination that will have a significant impact on the health of the public under current site usage. However, change of site usage could result in undesirable exposure.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of contaminated soils and decontamination of building floors and surfaces. Sixty cubic yards of contaminated material would be produced. The estimated cost is $570,000.

PROJECT STATUS

A complete radiological survey was completed in FY 1980 and a report is in preparation. Upon completion of the report, a determination will be made as to whether remedial action is required. Additional authority to implement remedial action is required.

KELLEX RESEARCH FACILITY
JERSEY CITY, NEW JERSEY

OWNER HISTORY

1942-1951:	Kellex Corporation
1951- :	Vitro Corporation of America
Current:	Delco-Levco and Pierpont Associates

SITE LOCATION

The Kellex research facility activities were conducted in one building located on the site of the M. W. Kellogg Company property in Jersey City, New Jersey.

MED/AEC SITE USE

The Kellex Corporation was established by the M. W. Kellogg Company in 1943 in order to design and construct the first gaseous diffusion plant for uranium enrichment. The work continued to July 1952 and included research and development of purex reprocessing for spent fuel and component testing with uranium hexafluoride.

POST MED/AEC SITE USE

The Kellex buildings were demolished around 1953 and only the concrete slab floor remains. The original area of the Kellogg facilities has been subdivided and is currently being developed as commercial properties. A supermarket and other stores have been constructed on part of the property. The location of the former Kellex building is presently unused and is owned by Pierpont Associates.

RADIOLOGICAL HISTORY

In 1953, the Vitro Corporation of America prepared a contamination status report that detailed the findings of a radiation survey of the former Kellex building. This report indicated that most external gamma radiation readings were less than 100 microroentgens per hour, and no transferable alpha or beta-gamma contamination was observed in any of the accessible areas.

Representatives from Oak Ridge Operations and ORNL conducted a site visit and exploratory survey of the Kellex site on October 21, 1976. The survey revealed gamma ray readings in the 5- to 6-microroentgen per hour range (background). However, due to the size of the property and uncertainty as to the exact location and extent of Kellex operations, it was decided that a formal survey should be conducted. A radiological survey was conducted under the FUSRAP by ORNL during March 1977.

Results of the 1977 radiological survey indicate that the radiation and radioactive levels were indistinguishable from background levels with the exception of a few isolated and well defined spots on or near the site of the former Kellex Laboratory.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action was indicated and work was started on the site in July 1979. During the remedial action, additional contamination was discovered and the decontamination effort extended to cover the additional areas. This additional work has since been suspended in order to evaluate results in the context of the criteria appropriate to the intended use of the site. The estimated cost for remedial action is $1,400,000.

PROJECT STATUS

A radiological survey was completed in March 1977; a draft of the final report, dated September 1977, has been prepared. The Assistant Secretary for Environment has determined that remedial action is required. Remedial action is underway. Authority for completing the remedial action exists under the Atomic Energy Act of 1954, as amended.

LAKE ONTARIO ORDNANCE WORKS ASSOCIATED PROPERTIES
LEWISTON, NEW YORK

OWNER HISTORY

1944-1955:	U.S. Government
1955-Present:	Private

In 1948, the AEC acquired approximately 1,511 acres of the former Lake Ontario Ordnance Works (LOOW) from the Army. In 1955, the AEC declared 1,298 acres excess and, as of 1968, this acreage had been acquired by the town of Lewiston (89 acres), Fort Conti Corporation (642 acres), Mr. M. W. Frank (199 acres), Niagara Mohawk Power Company (5 acres), The Somerset Group, Inc. (133 acres), and the Air Force (230 acres). In 1975, the ERDA declared a 22-acre sewage plant excess and transferred this plot to the town of Lewiston, New York, leaving 191 acres under DOE control.

SITE LOCATION

The DOE storage site currently consists of 191 acres and is located about 3 miles southeast of Youngstown, 3 miles northeast of Lewiston, and 7 miles north of the City of Niagara Falls in the County of Niagara Falls, New York. However, that portion of LOOW that was declared excess by the AEC and contains residual radioactive material above background, is considered the FUSRAP site.

MED/AEC SITE USE

This site was a portion of the former LOOW and was first used by the MED in 1944 for the storage of radioactive low-grade pitchblende residues from the nearby Tonawanda refinery. Following World War II, contaminated materials from wartime plants and some post-wartime operations were stored at the site. After April 1, 1949, part of the high-grade pitchblende residues from the St. Louis refinery were stored at the site in drums, and subsequently transferred to the 165-foot high concrete silo. In the early 1950s, the site was used as an interim storage site for incoming and outgoing uranium billets. In addition, radioactive materials from the University of Rochester and Knolls Atomic Power Laboratory (KAPL) were transferred to this storage site. The KAPL wastes were later transferred to the Oak Ridge National Laboratory burial grounds.

In about 1953, the AEC operated a boron isotope separation plant at the site. The plant was placed on standby in 1958 and was restarted in 1964 and again put on standby in July 1974.

POST MED/AEC SITE USE

The DOE site is currently dormant and the National Lead Company of Ohio (NLO) is under contract to act as caretaker. The 191 acres of this site that remain under DOE control constitute a DOE Surplus Facility. However, in 1958, at the termination of ore procurement contracts, 25-year-storage lease agreements were negotiated with African Metals Corporation (Afrimet), the U.S. subsidiary of Union Miniere du Haut Katanga of Brussels, Belgium (owner and supplier of Belgian Congo ore), for the storage of its residues in four concrete structures on the site. Approximately 60 percent (12,000 tons) of the radioactive residues stored at the site belong to Afrimet. These storage lease agreements expire on July 1, 1983.

RADIOLOGICAL HISTORY

In October 1970 and June 1971, radioactive surveys of the 1,298 acres formerly held by the AEC showed that about 6.5 acres exceeded the AEC criteria of 50 microroentgen per hour including background. Decontamination was carried out in 1972 and involved the removal of about 15,000 to 20,000 cubic yards of radioactive soil and debris. This contaminated material was piled on the remaining 191-acre AEC site. A final radiation survey conducted in June 1972 indicated that only a few portions of the central drainage and Sixmile Creek exceeded the 50 microroentgen per hour criteria, and beta-gamma levels measured at contact were less than 0.2 mrad/hr.

For a number of years, NLO has periodically sampled and analyzed the groundwaters and surface waters on and around the site. No significant radioactivity has been found in surface waters, and radium-226 and uranium concentrations in well samples are substantially below levels specified in guidelines for water in uncontrolled areas. In August 1978, the DOE Environmental Measurements Laboratory began offsite radon monitoring, both indoors and outdoors, to supplement the site fence-line monitoring conducted by NLO. To date, the average concentrations in residences neighboring the DOE site are within the range of indoor concentrations found in New York City and its suburbs.

REMEDIAL ACTION OPTIONS AND COSTS

The DOE is evaluating a number of options for long-term disposition of the residue at this site. In the interim, temporary remedial measures to minimize emanation of radon from the residues are being instituted and the monitoring program is being expanded. Further remedial action may be required. Preliminary estimates of cost are approximately $3,000,000.

PROJECT STATUS

A detailed radiological survey under the FUSRAP is underway. Remedial action to remove residual contamination from drainage areas and steps to prevent further offsite transport will be initiated during FY 1980. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

LINDE AIR PRODUCTS DIVISION
TONAWANDA, NEW YORK

OWNER HISTORY

Union Carbide Corporation - Linde Air Products Division

SITE LOCATION

The site, which contains approximately 55 acres, is located in a partially industrialized area of Tonawanda, New York. Five buildings on this site were involved in the MED activities.

MED/AEC SITE USE

The Linde Division was under contract with the MED to perform uranium separations during the period from 1942 through approximately 1948. Uranium oxide (UO_2) was produced from ores received from Colorado and the Belgian Congo and then converted to uranium tetrafluoride. All buildings involved in the MED activities were transferred back to Linde Division in 1953.

POST MED/AEC SITE USE

Four of the five buildings involved are presently being used for either warehousing, fabrication facilities, research laboratories, or offices. Approximately 50 employees utilize these four buildings. The fifth building is presently not being used.

RADIOLOGICAL HISTORY

A radiation survey was conducted by the AEC Health and Safety Division-NYO in November 1952 to determine disposition of equipment used in the uranium operations. All equipment was removed and decontamination took place in 1953. A radiological survey was conducted under the FUSRAP during October and November 1976. As a result of findings of this survey, Linde applied for and received an amendment to its New York State license to include the contaminated building.

REMEDIAL ACTION OPTIONS AND COST

Remedial action is indicated and could involve extensive decontamination of buildings, excavation of soils under building floors and outdoors, and cleanup of streams and ditches onsite. Fifty-thousand cubic yards of contaminated material would be produced. Estimated cost for this remedial action is $35,000,000.

PROJECT STATUS

A radiological survey was completed during October and November 1976. The final report was issued in May 1978. The Assistant Secretary for Environment has determined that the site will require remedial action. However, additional radiological work is required to develop engineering plans. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

MALLINCKRODT, INC.
ST. LOUIS, MISSOURI

OWNER HISTORY

The site has been and is currently owned and operated by Mallinckrodt, Inc., formerly named Mallinckrodt Chemical Works.

SITE LOCATION

Mallinckrodt leased portions of two locations in St. Louis at Broadway Street and at Destrehan Street to the MED/AEC for the processing of uranium concentrate. About 20 existing buildings on the Mallinckrodt property at Broadway and Destrehan, plus their surroundings, were subject to radiological contamination.

MED/AEC SITE USE

In April 1942, Mallinckrodt Chemical Works was requested by the Army to set up an industrial-scale process to produce uranium dioxide and uranium trioxide. Mallinckrodt had the processing system operating by early summer 1942 and provided uranium compounds and uranium metal for use in the research, development, and production programs of the AEC. Work also included (1) production of uranium tetrafluoride (UF_4), (2) production of uranium derby metal (vacuum recast of purified ingot metal), (3) machining of uranium metal rods for reactor fuel slugs, (4) reversion of uranium tetrafluoride to UO_2 or U_3O_8, (5) recovery of scrap uranium metal, (6) production of UO_2F_2, (7) extraction and concentration of thorium-230 from pitchblende raffinate, and (8) experimental processing of very low enrichment UF_4. From 1942 through 1945, uranium processing was done exclusively at the Broadway Street location. Some uranium metallurgical research continued through 1956. From 1945 to 1957, uranium ore or concentrate was processed in buildings at the Destrehan Street location. In 1957, all operations at Destrehan were terminated.

POST MED/AEC SITE USE

Since 1962, the site has been used for various commercial chemical production operations. Some of the original buildings have been torn down, some are being used as warehouses, and new buildings have been constructed. Columbian-Tantalum ore and potassium compounds are stored onsite.

RADIOLOGICAL HISTORY

From 1948 to 1950, the main plant property was decontaminated and final contamination surveys were performed. In 1951, the main plant property was returned to Mallinckrodt for unrestricted use. Between 1957 and 1962, the Destrehan and Broadway Street properties were also decontaminated, surveyed, and released for unrestricted use. In the process, some of the buildings were removed to the AEC waste disposal sites. Contaminated earth was also removed and backfilled. Early in the program, decontamination procedures were supervised by the New York Operations Office of AEC and later by the Oak Ridge Operations Office. The AEC decontamination activities did not reduce radioactivity levels to background but reduced them only to the prevailing acceptable levels at that time. A new radiological survey of the former uranium processing areas was conducted under the FUSRAP during the summer of 1977.

Results of the 1977 survey indicate alpha and beta-gamma contamination levels inside and outside some of the buildings were above limits set by current Federal guidelines

concerning the release of property for unrestricted use. Elevated external gamma radiation levels were measured at some outdoor locations and in some of the buildings. Quantities of uranium in an amount that may require licensing were found in soil at some places, and the concentration of uranium in one water sample taken from an old waste pit was in excess of Federal water quality standards stated in 10 CFR 20. Radon and radon daughter concentrations in three buildings were in excess of current Federal guidelines for nonoccupational radiation exposure.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve extensive excavation of contaminated soil and decontamination of buildings including removal of structural elements. Forty-nine-thousand cubic yards of contaminated material would be produced. Estimated cost for this remedial action is $26,000,000.

PROJECT STATUS

A radiological survey was completed in 1977, a draft report has been completed, and the final report is being prepared. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority is needed to implement remedial action.

MIDDLESEX MUNICIPAL LANDFILL
MIDDLESEX, NEW JERSEY

OWNER HISTORY

Pre 1961: Borough of Middlesex
Post 1961: Borough of Middlesex and Middlesex Presbyterian Church (5 acres)

SITE LOCATION

The site is located in the Borough of Middlesex, New Jersey, approximately 35 miles northeast of Trenton. The contaminated area covers about 3 acres.

MED/AEC SITE USE

This area is a former landfill for the Borough of Middlesex. The landfill was used by the Middlesex Sampling plant for disposal of nonradioactive wastes. However, during the operation of the sampling plant, some contaminated wastes were shipped to the landfill. There is no documented material to indicate when the contamination of the landfill occurred; however, a review of operating files from 1946 to 1966 indicates that the most probable time frame was between November 1947 and October 1948. Construction of a drainage ditch and paved storage area took place during this period. It is believed that the material deposited at the landfill may have resulted from this construction effort.

POST MED/AEC SITE USE

The contaminated area is currently undeveloped and not used for any activity.

RADIOLOGICAL HISTORY

In May 1960, during a local civil defense (CD) exercise, CD monitors detected elevated radiation levels in the landfill. The matter came to public attention and received newspaper coverage. The AEC noted the issue and upon reviewing its past local activities concluded that AEC operations were the source. Upon analytical confirmation of the presence of pitchblende, a further survey of the area was made. Readings taken at that time confirmed gamma radiation levels 20 to 50 times background over a fairly consolidated area of less than one-half acre.

Following meetings with local officials in November 1960 to discuss the significance of survey findings and to offer remedial assistance, the AEC removed the part of the material nearest the surface (about 650 cubic yards). The area was covered with about 2 feet of clean dirt sufficient to shield surface radiation levels to about 50 microroentgens per hr at 1 meter. The contaminated soil was removed to the AEC New Brunswick Laboratory site. Upon receiving assurance by the AEC that no health hazard existed, Borough officials agreed that the situation was satisfactory. No official record of the residual contamination exists in available Borough records. On January 30, 1974, another meeting was held with Borough officials to request permission to resurvey the involved area to permit re-evaluation of current conditions. Location of the suspect area was confirmed by survey data; it was in the area of the boundary between the church and Borough properties. The Oak Ridge National Laboratory has conducted additional survey and assessment work during 1978. During the period May 20-27, 1978, EG&G (a DOE contractor) performed an aerial survey of Middlesex. The survey produced no new conclusions related to the landfill.

As a result of the survey findings, the following conclusions were made:

- The contaminated area in its present configuration and use presents no significant radiation exposure potential to the public. This should be the case as long as the area is undisturbed by excavation or the construction of habitable enclosures.

- The exposure of individuals at or exceeding guide levels cannot be convincingly dismissed as a credible possibility under circumstances which could exist if the area were developed in the future with residences or other habitable structures.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated. In April 1978, an engineering evaluation and environmental analysis was completed of options for various remedial actions at this site. The options range from stabilization of the material onsite to removal of all material to background radiation levels and backfilling to present condition with clean fill. Based upon the engineering evaluation of the site, it is estimated that the original 6,500 cubic yards of contaminated soil from the sampling plant have now been dispersed with other soil and landfill debris. The contaminated portion involves a volume of between 34,000 to 69,000 cubic yards of soil. There has been additional sanitary landfill activity since the radioactivity was dispersed in the landfill. An estimated 16,000 to 21,000 cubic yards of nonradioactive soil and debris currently cover the contaminated soils. The estimated cost for the removal and backfill remedial action is $50,000,000.

PROJECT STATUS

Radiological surveys have been completed. An engineering evaluation report was issued in April 1979 and an environmental analysis was issued in July 1979. The Assistant Secretary for Environment has determined that the site will require remedial action. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

MIDDLESEX SAMPLING PLANT
MIDDLESEX, NEW JERSEY

OWNER HISTORY

1943-1950: American Marietta Company
1950-Present: U.S. Government

SITE LOCATION

The site is located in Middlesex, New Jersey, and contains six buildings on 9.6 acres. Some portions of the adjacent and nearby properties, especially along the south border, have significantly contaminated soil. Two nonadjacent private properties have also been identified as having contaminated soil from the Middlesex Sampling Plant: the Our Lady of Mount Virgin Catholic Church at 650 Harris Avenue, Middlesex, New Jersey, and the private residence at 432 Williams Street, Piscataway, New Jersey.

MED/AEC SITE USE

This facility, also known as Perry's Warehouse, was used for the sampling, weighing, assaying, and storage of uranium and thorium ores. The uranium sampling operations were conducted between November 1943 and February 1955. The bulk of the Belgian Congo uranium ores and other uranium ores used by the United States were handled at this site. The residue from the processing of these ores was temporarily stored at Middlesex prior to its return to the vendor. There are indications that the site was also used as an interim holding site for disposition of various research-related and decontamination wastes. Following the termination of the uranium-sampling operations, the primary AEC activities at the plant involved the sampling and storage of thorium materials and residue. All AEC activities at the site terminated in September 1967 with the conclusion of the decontamination of the site and certification of the site for unrestricted disposal.

POST MED/AEC SITE USE

The site was used by the U.S. Marine Corps for their 6th Motor Transport Battalion reserve training from 1969 to approximately 1975. The site is presently in the custodial care of the DOE. Access is restricted by a 7-foot-high chain-link fence.

RADIOLOGICAL HISTORY

Prior to 1967, the AEC contracted Isotopes, Inc., to decontaminate the site. The AEC Health and Nuclear Safety Branch performed a follow-up survey and additional decontamination. Upon completion of this decontamination on September 2, 1967, Oak Ridge Operations certified the site for unrestricted disposal. Decontamination required sandblasting, vacuuming, detergent and acid washing, concrete chipping, equipment removal, and in cases of severe contamination, building member removal. Waste was transported by rail to a Nuclear Fuel Services licensed burial site at West Valley, New York. A radiological survey was completed under the FUSRAP in May 1976.

Results of the 1976 survey indicate surface contamination levels on the former plant site exceed the Nuclear Regulatory Commission guidelines, and radon concentration levels exceed the nonoccupational maximum permissible concentration (10 CFR 20) in some structures. These results indicate the possible need for extensive radon and radon daughter measurements in structures both onsite and offsite over periods as recommended in 10 CFR 712 for structures in Grand Junction, Colorado. As a result

of an aerial survey conducted by EG&G for the DOE between May 20 and May 27, 1976, and followup ground surveys by ORNL, two additional properties were identified that were contaminated by material handled at the Sampling Plant.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve excavation of soil at the site and adjacent and nearby properties, and removal of buildings and equipment from the sampling plant site. The DOE has proposed a two-stage remedial action at this site and is in the process of obtaining local government and owner approval. The plan would entail the cleanup of all offsite contaminated property and interim storage of the contaminated material onsite until a disposal site is identified at which time the entire site would be decontaminated. Seventy-seven-hundred cubic yards of contaminated materials would be produced. Estimated cost for this remedial action is $48,000,000.

PROJECT STATUS

A radiological survey was performed in May 1976. The final report was issued in November 1977. Additional offsite survey work is being conducted. The Assistant Secretary for Environment has determined that remedial action is required. An engineering evaluation report (Title I) and an environmental analysis report were issued in July 1979. The DOE has drafted preliminary remedial action plans that schedule the remedial action to begin in FY 1980 and a cooperative agreement between the DOE, the Borough of Middlesex, and the State of New Jersey was signed in December 1979. In addition, the NEPA process has been completed for remedial actions at the Williams Street and Catholic Church properties and proposed remedial actions have been approved (September 1979). Authority exists for implementation under the Atomic Energy Act of 1954, as amended.

NATIONAL GUARD ARMORY
CHICAGO, ILLINOIS

OWNER HISTORY

The property is owned by the State of Illinois.

SITE LOCATION

The armory is located at 52nd Street and Cottage Grove Avenue, Chicago, Illinois.

MED/AEC SITE USE

During the MED/AEC era, uranium was apparently used at the site and it is believed that some type of uranium processing was performed. Personnel recall that the grandstand surrounding the arena was used for storage of radioactive materials. The use of the arena may have involved the chemical processing and metal casting of uranium. Use of the facility was terminated in 1951.

POST MED/AEC SITE USE

Contaminated dirt from the arena was removed and at a later date additional dirt removed and replaced with a concrete pad. It is currently in use as offices, classrooms, and as storage and garage areas.

RADIOLOGICAL HISTORY

A survey was conducted under the FUSRAP during September and October 1978. Surface contamination was found in 10 of over 160 rooms in the armory. Contamination was generally in small localized spots except for Room 1 where it was widespread. The highest alpha contamination was $5x10^4$ dis/min/$100cm^2$ and the highest beta-gamma was $3.5x10^3$ dis/min/$100cm^2$. Contamination was also observed in catch basins in a number of rooms. Air samples indicated radon concentrations below maximum permissable concentration for uncontrolled areas. Analyses of soil samples indicated results within the range of concentrations found in background samples.

Direct instrument and smear surveys indicate some contamination is still present within the building. All of the contamination in Room 1 exceeds guidelines for unrestricted use. Contamination in two catch basins in Room 1 exceeds guidelines. Seven other locations throughout the building exceed guidelines. Radon concentrations in air samples were normal and soil sample analyses showed no elevated readings above background levels in soils. Other radioactive items such as radium dials were also noted.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be indicated and could involve decontamination of building surfaces and excavation of floor areas. Twenty-five cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $710,000.

PROJECT STATUS

A radiological survey was completed in October 1978. Draft survey reports have been completed and final reports are being prepared. The Assistant Secretary for Environment will make a determination of need following the final report. Authority to implement remedial action will be required.

OLIN CORPORATION
JOLIET, ILLINOIS

OWNER HISTORY

The site was originally owned by Blockson Chemical Company, which was sold in 1955 to Olin Mathieson Chemical Corporation, the present owner.

SITE LOCATION

The site consists of a single building used for a pilot plant operation in Joliet, Illinois.

MED/AEC SITE USE

The site was used during the period of 1951 to 1962 to conduct a development program for the extraction of uranium from phosphoric acid.

POST MED/AEC SITE USE

The building (site) is presently being used to process phosphoric acid which contains elevated levels of natural uranium.

RADIOLOGICAL HISTORY

The work at the site included operation of a small pilot plant for the extraction of uranium from phosphoric acid. A radiological survey for the FUSRAP was conducted from March to November 1978. A draft of the final report has been prepared and is undergoing review.

Natural uranium contamination was found on the floors, overhead beams, and in the tanks and equipment where chemicals were processed. Small areas exceed applicable guidelines. Some contamination of the roof was found in which radium-226 was identified. In some places contamination is easily removed. The extent to which the contamination is due to the MED/AEC work because of the present operation is not known. Radon concentrations in air samples were normal. Results of analyses of soil samples taken about the grounds adjacent to the buildings showed no elevated readings above natural background in the soil.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and would involve decontamination of building surfaces and equipment. Three-hundred cubic yards of contaminated material might be produced. Estimated cost for this remedial action is $680,000.

PROJECT STATUS

Upon completion of the radiological survey report, the Assistant Secretary for the Environment will determine whether the site requires remedial action. Authority to implement a remedial action will be required.

PALOS PARK FOREST PRESERVE
COOK COUNTY, ILLINOIS

OWNER HISTORY

1942-1956:	Leased by the U.S. Army Corp of Engineers from Cook County Forest Preserve District
1956-Present:	Cook County Forest Preserve District

SITE LOCATION

The park preserve is located in Cook County, approximately 5 miles east of Lemont, Illinois. Within the park preserve, 20 acres were used for the MED/AEC activities.

MED/AEC SITE USE

The site contained two nuclear reactors and associated buildings and laboratories and a radioactive waste burial facility. The first successful nuclear reactor, CP-1 at the University of Chicago, was rebuilt as CP-2 at the site. The first heavy-water cooled and moderated reactor, CP-3 (designated CP-3' when rebuilt) was also at the site. Among the programs carried out at this site during and after World War II were fission product separations, reactor physics, tritium recovery from irradiated lithium, and studies of the metabolic effects of radionuclides on laboratory animals.

POST MED/AEC SITE USE

The site is currently utilized as part of the entire park forest preserve for recreational activities.

RADIOLOGICAL HISTORY

In 1956, the Federal Government returned all of the 20 acres to the Forest Preserve District. Before that time, the research reactors were decommissioned, radioactive materials were removed from the site and remaining radioactive components, including the reactor vessel, were encased in concrete and buried onsite. The empty buildings were surveyed, decontaminated if necessary, and demolished. The waste burial site was decommissioned by digging 8-foot-deep trenches around the perimeter and filling them with concrete. A 1-foot-thick concrete pad was poured over the top. The plot was then covered with soil and seeded. By the summer of 1956, decommissioning was complete, and the area was surveyed with state-of-the-art portable survey equipment. No detectable surface contamination was found. A limited environmental monitoring program was begun at the Palos site in 1954, continuing about every other year until 1975.

An extensive radiological survey was conducted under the FUSRAP during 1977 which showed that tritium was migrating from the former waste burial site.

Results of the 1977 survey indicate that the only significant pathway for exposure to the public is tritiated water moving from the former waste burial site to a dolomite aquifer and being consumed by individuals using the picnic wells on the preserve. The possible dose to people from this pathway is estimated to be 0.7 mrem/year.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve excavation of contaminated material and restoration. Estimated cost for this remedial action is $7,100,000.

PROJECT STATUS

A radiological survey was completed during 1977 and the final report was issued in April 1978. The Assistant Secretary for Environment has determined that the site will require remedial action. Both an environmental analysis report and an engineering evaluation report-Title I have been completed and were issued in September 1979. Additional authority is required to implement remedial action.

ST. LOUIS AIRPORT
ST. LOUIS, MISSOURI

OWNER HISTORY

1946-1973: U.S. Government
1973-Present: City of St. Louis, Airport Authority

SITE LOCATION

The storage site is a 21.7-acre tract located adjacent to the north boundary of the Lambert-St. Louis International Airport. The site is approximately 15 miles northwest of St. Louis.

MED/AEC SITE USE

The site was used for storage of residues and contaminated scrap and equipment generated by the Mallinckrodt Chemical Corporation, Destrehan Street Plant uranium-processing operations during the period 1946 to 1953. Various residues were stored above ground and in the open, above ground in steel drums, and below ground in an open concrete pit. Contaminated scrap and equipment were buried and later covered with clean fill. During 1966 and 1967, all residues were removed from the site.

POST MED/AEC SITE USE

The site has remained unused since 1967 with access controlled by the airport manager. Decontamination activities have taken place during 1969. Proposals have been made by the NRC to relocate contaminated material from the formerly licensed Latty Avenue site in Hazelwood, Missouri; and the St. Louis Airport Authority has recommended development of the site as a driver-training course for the police academy.

RADIOLOGICAL HISTORY

Wastes generated from uranium processing and other activities between 1947 and 1967 were stored onsite. In addition, 60 truck loads of contaminated scrap metal and a contaminated vehicle were buried onsite. During 1966 and 1967, most of the stored residues were sold and removed from the site. All onsite structures were razed and buried onsite. Contaminated soil in the residue storage area was removed and 1 to 3 feet of clean fill spread over the site. A radiological survey for the FUSRAP was conducted in August and November of 1978. Present access to the site is limited and it is used to receive clean rocks and fill.

Contamination of the site is due to buried deposits of naturally occurring radionuclides, namely uranium-238, radium-226, and thorium-230. Average concentrations of radon and radon daughters in air were well below guideline values for the general public. Surface radiation guidelines are exceeded at 10 onsite locations and 2 offsite locations in a ditch on the site side of an adjacent road north of the site. Soil along the northern fence has been disturbed by burrowing animals and eroded by water drainage. This contamination is the cause of the elevated surface beta-gamma and external gamma radiation exposures found in these ditches. The guidelines for external gamma exposure would be exceeded at five locations at the site if the area were frequently occupied. Currently, access to the site is limited.

REMEDIAL ACTION OPTIONS AND COSTS

Two remedial action options have been proposed. The first is stabilization and control for which a cost estimate ranging from 1.5 to 3 million dollars has been developed. The second is removal of 180,000 cubic yards of the contaminated material and restoration of the site at an estimated cost of $98,000,000.

PROJECT STATUS

A radiological survey was conducted in August and November 1978; the final report was issued in September 1979. An environmental impact analysis was issued in July 1979 addressing proposed and alternative actions. No Title I design has been done. Additional authority for the Assistant Secretary for Nuclear Energy to implement remedial action is required.

SEAWAY INDUSTRIAL PARK
TONAWANDA, NEW YORK

OWNER HISTORY

Seaway Industrial Park Development Company, Inc.

SITE LOCATION

The site, covering 100 acres, is located in Tonawanda, New York, adjacent to the Niagara River. It is primarily used as a landfill. Approximately 13 acres of the landfill has been used for storage of radioactive materials. It is adjacent to the Ashland Oil Company property, another formerly utilized MED/AEC site.

SITE USE

In 1974, approximately 6,000 cubic yards of uranium-processing residue, comprised essentially of low-grade uranium ore tailings, were excavated from the adjacent Ashland Oil, Inc., property and dumped onto three areas of the landfill.

RADIOLOGICAL HISTORY

Since their initial transport to the site, the residues have been somewhat scattered and mixed with clean soil by earthmoving and spreading associated with the landfill operation. A radiological survey was conducted under the FUSRAP during August 1976. The survey indicated that radioactive material is being transported off-site by surface runoff. An aerial survey was conducted in September 1979.

Results of the 1976 survey indicate external gamma, radon, and radon daughter levels exceed guideline values over small areas of the landfill. However, these levels do not present a health hazard under the current site use because of low exposure time to landfill workers in the vicinity of the residues.

Potential health hazards could result from either conversion of the site use by construction of buildings or from use of residues for fill at another site or as a construction material. If a building were constructed in certain portions of the site, radon daughter levels of 0.15 or higher could develop in the building.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve excavation of the residues from the site, including a stream and drainage ditch. Thirty-nine-thousand cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $24,000,000.

PROJECT STATUS

A radiological survey was completed in August 1976; the final report was issued in May 1978. The Assistant Secretary for Environment has determined that the site will require remedial action. Additional authority for the Assistant Secretary for Nuclear Energy to implement remedial action is required.

SENECA ARMY DEPOT
ROMULUS, NEW YORK

OWNER HISTORY

The site is owned and operated by the U.S. Army.

SITE LOCATION

The depot consists of approximately 10,000 acres, of which approximately 20 acres were involved in the MED activities. This area consists of 11 munitions bunkers and surrounding areas over which material was transported.

MED/AEC SITE USE

About 2,000 barrels of pitchblende ore were stored in 11 munitions bunkers during a short period in the 1940s.

POST MED/AEC SITE USE

Upon removal of the ore, the bunkers reverted back to storage sites for ammunition and have continued in this function since that time.

RADIOLOGICAL HISTORY

Since the original short-term storage of uranium ore in munitions bunkers, some contamination of the interior surfaces of at least eight bunkers has been present. A radiological survey was conducted under the FUSRAP during September 1976. The survey indicated that the interior surfaces of at least eight of the bunkers have been contaminated with uranium ore and as a consequence, natural uranium and its daughters, including radium-226, may be found on these surfaces and on outdoor surfaces near the entrances to these bunkers.

Results of the 1976 survey indicate that the interior surfaces of at least eight of the bunkers were contaminated with uranium ore. Direct alpha readings exceeded the maximum guideline in some areas of each of the eight bunkers and transferable alpha exceeded the maximum guideline in six. Transferable beta contamination in excess of the guidelines was found in one area of the floor of one bunker. Radon daughter concentrations exceed 0.03WL in six bunkers but all were less than 0.048WL. External gamma radiation levels at one meter were below guideline values. The only contaminated soil was found near the surface in small areas near bunker entrances. No health hazard exists because of the very low occupancy time of the bunkers.

Potential health hazards could result from exposure to radon and radon daughters concentrations in the bunkers if occupancy times were to increase. While no crops are currently grown on site, use of the contaminated soil for such a purpose could produce additional human exposure.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action is indicated and could involve thoroughly cleaning all floors, walls, ceilings, vents, and drains. Contaminated soil outside the bunkers could be excavated. Four-hundred cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $860,000.

PROJECT STATUS

A radiological status survey was completed during September 1976; the final report was issued in February 1979. The Assistant Secretary for Environment has determined that the site will require remedial action. Authority to implement remedial action exists under the Atomic Energy Act of 1954, as amended.

SHPACK LANDFILL
NORTON, MASSACHUSETTS

OWNER HISTORY

The property is presently owned by Mrs. Isadore Shpack and had been owned by the Shpack family before the suspected date of contamination.

SITE LOCATION

The site is located in Norton, Massachusetts, near the common corporate boundary of Norton and Attleboro. Norton is approximately 15 miles northeast of Providence, Rhode Island. The area of concern comprises approximately 5 acres.

MED/AEC SITE USE

The Shpack Landfill was a private landfill that received "industrial" wastes from local operations. A NRC investigation determined that the former M&C Nuclear, Inc., Attleboro, Massachusetts (merged with Texas Instruments, Inc., in 1959) had used the Shpack Landfill area for the disposal of trash and other material, including burning zirconium ashes, associated with nuclear fuel operations conducted at the facility from 1957 to 1966. The NRC investigation concluded that it is possible that the aforementioned facility was the source of the major portion of the radioactive material.

POST MED/AEC SITE USE

The landfill is now closed and the area is undeveloped. The surface presently contains metal, brick, concrete, blocks, iron drums, plastics, and miscellaneous debris. The area is poorly drained and covered with water part of the year.

RADIOLOGICAL HISTORY

On September 22, 1978, the NRC Region I Office was contacted by a concerned citizen who had identified elevated (above background) radiation levels at the Shpack Landfill site. A special investigation by the NRC from October through December 1978 verified the presence of radioactivity above background levels at the Shpack Landfill. Gross alpha measurements of well water from the Shpack residence were found to be within EPA Drinking Water Standards. An independent study conducted by Brown University students produced results which were orders of magnitude higher than the gross alpha measurements of the NRC study and far in excess of EPA standards. The NRC, in conjunction with the State of Massachusetts, collected a number of additional water samples and had them analyzed at a number of independent laboratories. The results verified that well water in the area was not affected as all well samples were below EPA standards. As a result, the NRC determined contamination at the landfill posed no immediate hazard to human health but potential for exposure did exist. Representatives from the DOE and ORNL visited the site and performed a preliminary ground survey and EG&G, Inc., performed an aerial radiological survey. The ground survey (July 24, 1979) concluded that the site was contaminated with uranium- and radium-bearing materials and that the uranium was primarily depleted uranium. A full radiological survey was recommended. The aerial survey (August 8 and 9, 1979) did not detect any radiation levels significantly above those due to natural background.

Results of studies completed to date indicate that the current use of the landfill does not pose an immediate hazard to human health but potential for exposure does exist.

REMEDIAL ACTION OPTIONS AND COST

Remedial action may be required and could include excavation of contaminated soil. A preliminary estimate indicated that approximately 4,500 cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $2,200,000.

PROJECT STATUS

The DOE has asked ORNL to develop and implement a survey plan for the Shpack landfill site. Upon completion of these efforts, a determination will be made by the Assistant Secretary for Environment as to whether remedial action is required. A determination as to whether additional authority is required to implement remedial action is currently underway.

UNIVERSAL CYCLOPS, INC.
ALIQUIPPA, PENNSYLVANIA

OWNER HISTORY

1942-1955:	Vulcan Crucible Steel Company
1955-1960:	Vulcan Crucible Steel of H. K. Porter
1960-1966:	Vulcan-Kidd Steel of H. K. Porter
1966-Present:	Vulcan Cyclops, Inc.

SITE LOCATION

The site is located in Aliquippa, Pennsylvania, and consists of one building and surrounding areas.

MED/AEC SITE USE

Uranium billets were received, rolled into rods, boxed, and shipped out. This site consisted of a rolling mill, two furnaces for heating, and cutting and extruding equipment. The finished rods were stored in boxcars after being transferred to the receiving and shipping room for weighing. The building is one story over 30 feet high with part concrete, part dirt, and part metal floor.

POST MED/AEC SITE USE

Portions of the building are presently leased to Heritage Box Company and Precision-Kidd for use as storage areas.

RADIOLOGICAL HISTORY

During February 1949, dust samples at the mill were collected by representatives of the New York Operations Office-AEC. From data obtained from these samples, it was apparent that the entire group of employees was exposed to concentrations of alpha-emitting dust that were above the preferred level. Recommended corrective actions were provided to the Vulcan Crucible Steel Company. A follow-up survey was made and required decontamination and equipment disposition defined. Decontamination was completed by March 1950. A radiological survey was conducted under the FUSRAP during May 1978.

Results of the 1978 survey indicate some contamination is still present in the building. Floor areas and overhead beams showed transferable natural uranium contamination. Radon concentrations in air were normal. Only one soil sample contained elevated levels of uranium. Current use of the building does not present a health hazard. However, cleaning or demolition of the building could cause significant exposure.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of a small amount of soil and decontamination of one building. Fifty-five cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $1,000,000.

PROJECT STATUS

A radiological survey was completed in May 1978. A draft report has been issued and is undergoing review. Upon issuance of the final report, a determination will be made by the Assistant Secretary for Environment as to whether remedial action is required. Additional authority to implement remedial action is required.

VENTRON CORPORATION
BEVERLY, MASSACHUSETTS

OWNER HISTORY

1942-1965:	Metal Hydrides Corporation
1965-1976:	Ventron Corporation
1976-Present:	Thiokol Corporation

SITE LOCATION

The site is located in Beverly, Massachusetts, approximately 15 miles northeast of Boston. Three buildings were used for MED/AEC-related work.

MED/AEC SITE USE

From 1942 to 1948, Metal Hydrides Corporation was under contract to the MED and the AEC for conversion of uranium oxide to uranium metal powder, using calcium hydride. The method was proven at Metal Hydrides Corporation earlier in 1941. As better methods for production of uranium metal were developed, Metal Hydrides Corporation shifted their operations toward recovering uranium scrap and turnings from the slug fabrication plant at Hanford. Two wooden buildings that contained the foundry facilities were demolished some time between 1948 and 1950. Two other buildings have been erected at these locations. The remaining original building contained furnace and leaching facilities, a mixing room, a drying room, and analytical laboratories.

RADIOLOGICAL HISTORY

A radiation survey conducted in 1948 listed as contaminated the two foundry buildings and various pieces of equipment. As a result of that survey, it was recommended that painted surfaces be cleaned by sandblasting and contaminated concrete floor and platform materials be removed.

A visit to the site for exploratory measurements was made in January 1977 by Oak Ridge Operations and ORNL personnel. It was determined, based on the results of the exploratory measurements, that a complete radiological survey of the entire site should be performed.

Based on the 1977 exploratory measurements, soil and building contamination above background levels exist at the site. The degree and extent of the contamination will be determined from a complete radiological survey.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of contaminated soil and decontamination of building floors and surfaces. A preliminary estimate indicated that 100 cubic yards of contaminated material would be produced. The estimated cost for this remedial action is $880,000.

PROJECT STATUS

A radiological survey is scheduled. Upon completion of the survey, a determination will be made by the Assistant Secretary for Environment as to whether remedial action is required. Additional authority to implement remedial action is required.

WATERTOWN ARSENAL
WATERTOWN, MASSACHUSETTS

OWNER HISTORY

1946-1967: U.S. Government
1967-Present: Watertown Redevelopment Corporation

SITE LOCATION

The site is located adjacent to the current boundary of the Watertown Arsenal in Watertown, Massachusetts, approximately 5 miles west of Boston. Only one building has been confirmed as being utilized for the the AEC activities; however, several additional buildings may have been.

MED/AEC SITE USE

The Massachusetts Institute of Technology (MIT) operated a laboratory and a uranium ore testing facility for the AEC in a now-demolished building at the Watertown Arsenal. A modified ion exchange technique for production of U_3O_8, which employed a fluidized bed system, was developed at this site. Initial research on African ores was conducted at MIT in Cambridge. The activity was transferred to the Watertown Arsenal (building 421) in 1946. MIT conducted the research activities until 1950 at which time American Cyanamid took responsibility for the functions of the site. In 1953, the AEC activities at Watertown Arsenal, building 421, were transferred to a new facility.

POST MED/AEC SITE USE

The site has been transferred to the Watertown Redevelopment Corporation and is presently unused. Only the concrete pad of building 421 remains. Operations involving uranium are continuing in other areas of the arsenal.

RADIOLOGICAL HISTORY

The AEC Chicago Operations and Argonne National Laboratory (ANL) completed a comprehensive radiological survey of the portion of the arsenal (building 421 and surrounding area) used for the AEC activities. Direct instrument surveys of the pad of building 421 and south wall of building 331 (nearest building to the pad) identified three small spots on the pad that exceed the proposed ANSI standard No. N13.12. Smears indicated that the contamination was fixed and the analysis of one sample identified the contamination to be from natural uranium. Other direct instrument measurements taken showed no readings above natural background. Analyses of soil samples, water samples, and measurements of radon in the air gave no indication of radiation above natural background.

During the ANL radiological assessment of the building 421 site, it was discovered that several additional buildings and facilities were involved in uranium operations during the MED/AEC era. This included buildings 34 and 41, which have been razed. Both building sites are within the confines of the arsenal area, though they have been turned over to the Watertown Redevelopment Corporation. There is no evidence of a radiological survey being performed for these two buildings. In addition, there is an area on the north side of Arsenal Street that had been used for uranium storage and as a burn area. A survey was made in this area by Watertown Arsenal Radiation Safety personnel in 1973. Their investigation revealed a significant amount of contamination on the pad and a need for a more comprehensive survey of the area. The DOE plans to

survey the area north of Arsenal Street and the pads of buildings 34 and 41 during 1980. These areas were used by the Army for uranium storage and as a burn area.

Based on the preliminary surveys, the contamination is at an acceptable level and does not represent a hazard to the general population. However, if site use is changed, there is a potential for excessive exposure.

REMEDIAL ACTION OPTIONS AND COSTS

Remedial action may be required and could involve excavation of soil and decontamination of the concrete pad. Two-hundred-sixty cubic yards of contaminated material would be produced. Estimated cost for this remedial action is $630,000.

PROJECT STATUS

Additional radiological survey work is scheduled for FY 1980. Upon completion of this survey, the Assistant Secretary for Environment will determine whether remedial action is required. Additional authority to implement remedial action is required.

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Department of Energy
Oak Ridge Operations
P. O. Box E
Oak Ridge, Tennessee 37831

March 24, 1986

Mr. Joseph F. Nemec
Program Manager - FUSRAP
Bechtel National, Inc.
P.O. Box 350
Oak Ridge, TN 37831

Dear Mr. Nemec:

FUSRAP PROTOCOLS

Enclosed for your information and use is one copy each of the current revisions of the FUSRAP summary protocol, the FUSRAP designation/elimination protocol, and the FUSRAP verification and certification protocol. These documents, in combination with the latest revision of the Energy Systems Acquisition Project Plan for FUSRAP, detail procedures, requirements, and responsibilities for each phase of the remedial action program effort.

If there are any questions, please call me.

Sincerely,

E. L. Keller

E. L. Keller, Director
Technical Services Division

CE-53:Keller

Enclosures:
As stated
cc w/encls.:

P. Merry-Libby, ANL
W. Latham, AD-421

35692

FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM

SUMMARY PROTOCOL
IDENTIFICATION - CHARACTERIZATION - DESIGNATION - REMEDIAL ACTION - CERTIFICATION

JANUARY 1986

U.S. DEPARTMENT OF ENERGY

OFFICE OF NUCLEAR ENERGY
DIVISION OF FACILITY AND SITE
DECOMMISSIONING PROJECTS

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SUMMARY PROTOCOL
IDENTIFICATION - DESIGNATION
REMEDIAL ACTION - CERTIFICATION

INTRODUCTION

This summary protocol describes those activities necessary for accomplishing the Formerly Utilized Sites Remedial Action Program objective, which is to ensure that sites formerly used by the Manhattan Engineer District and the Atomic Energy Commission are not contaminated with radioactive residues that may present a radiological hazard to the general public. This summary protocol is presented in four phases: Preliminary Analyses (identifying potentially contaminated sites), Radiological Evaluation and Designation (evaluating the radiological condition of the site and determining if remedial action is needed), Engineering and Remedial Action* (site characterization and planning, selecting, engineering, and implementing the action), and Certification of Site Conditions (verifying site conditions and archiving the records that document the results of remedial action). Additional guidance is provided on the first two phases and the fourth phase respectively in two supplements to this protocol entitled FUSRAP Designation/Elimination Protocol (Supplement No. 1) and the FUSRAP Verification and Certification Protocol (Supplement No. 2). Additional details regarding implementation of the third phase of the program are provided in the report Energy Systems Acquisition Project Plan-FUSRAP (Revision 1)" April 1985, and subsequent revisions.

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*Remedial action may involve decontamination or stabilization and restricted use through institutional control or physical modifications.

Appendix A is a flow diagram with decision points and assignment of responsibilities for specific program activities. All phases except the Engineering and Remedial Action Phase are outlined in some detail and covered in the enclosed flow charts. Only a brief discussion of the Engineering and Remedial Action Phase is contained in this protocol (see "Energy Systems Acquisition Project Plan--Formerly Utilized Sites Remedial Action Program, Revision 1," Steps 3 through 7, April 1985).

This protocol places the primary emphasis on contaminated sites or potentially contaminated sites for which there is existing authority that will permit DOE to perform remedial action at the site. However, the section on the first phase of this protocol also discusses the actions taken with regard to sites for which DOE is unable to establish remedial action authority. In the interest of efficiency and economy of operation, this protocol limits the amount of radiological survey data collected during the first two phases of the protocol to the minimum needed to determine if a site should be included in the program or eliminated from it. Any additional radiological data needed for project engineering will be accomplished during the engineering and remedial action phase of the operation. Similar guidance is provided for engineering of the remedial action to ensure that the magnitude and cost of the engineering, planning, and environmental reviews do not exceed the worth or the beneficial effect of the action. Throughout this process, the professional judgment of the radiological survey personnel and the engineering and project management personnel is utilized, with guidance from the DOE Division of Facility and Site Decommissioning Projects (DFSD) to determine the level of survey, engineering, and/or environmental work required to achieve the associated goals.

In order to ensure that any remedial action completed is preformed to comply with and meet appropriate standards and guidelines, the last phase, Certification Phase, includes a verification activity. The

goal of this phase is also to ensure through proper documentation that each remedial action is adequately documented and archived so that a permanent record of its final radiological condition will always be available.

SUMMARY PROTOCOL

The following narrative was prepared, along with Figure I--Preliminary Analyses, Figure II--Radiological Evaluation and Designation and Figure III--Engineering and Remedial Action and Certification of Site Condition (attached), to describe DOE protocols for determining if a site warrants consideration for remedial action. The narrative is subdivided to follow these figures. As can be noted in Figures I, II, and III, the decision point that is the transition from one phase to the next is repeated on these figures but is discussed in the narrative in the earlier of the two phases.

PRELIMINARY ANALYSES PHASE

During this phase of the program, sites are identified and evaluated to determine if they can be designated (included in) or eliminated from the remedial action program, or if a radiological survey of the site is required to more clearly define the radiological condition of the site to support this decision. This phase has five steps that include two decision points. This phase of the program is conducted by DOE-DFSD with assistance from a technical support contractor, a radiological survey contractor, and an aerial survey contractor as appropriate.

Step 1 - Data Collection and Site Identification

During this step, information sources are identified and investigated by the DOE-DFSD Technical Support Contractor. These sources include input from individuals or organizations and historical

FIGURE I

SUMMARY FLOW SHEET FOR PRELIMINARY ANALYSES PHASE OF FUSRAP REMEDIAL ACTION PROTOCOL

STEP 1 — DATA COLLECTION AND SITE IDENTIFICATION

IDENTIFY RECORDS SOURCES

USE INPUT FROM THE GENERAL PUBLIC AND OTHER SOURCES

CONDUCT RECORDS SEARCHES

- Assemble and Maintain a List of Sites
- Assemble Available
- Assemble Operating Records
- Assemble Radiological Data of Health and Safety Studies
- Assemble Ownership Data
- Identify Licenses
- Interview Knowledgeable

STEP Z — HISTORICAL DATA ANALYSIS

DETERMINE POTENTIAL FOR CONTAMINATION

CONTAMINATION POTENTIAL SIGNIFICANT IF

- Large Operation (Quantity and Time)
- History of Burial
- Radiological Data Suggest Contamination
- Input from Cognizant Individuals Suggest Contamination

CONTAMINATION POSSIBLE IF

- Historical Data Indicates AEC Operations Could Have Resulted in Contamination or that Onsite Residue may be Present
- Little or no Information is Available to Indicate Site Clean-Up Completed

CONTAMINATION POTENTIAL LOW IF

- Only Small Quantities of Material Were Handled
- Site Operated for a Short Time
- Previous Surveys Adequately Demonstrate Clean-up

DETERMINE DOE AUTHORITY FOR REMEDIAL ACTION

DOE HAS AUTHORITY IF

- Contractual Records Indicate
- An Imminent Health Hazard May Exist (Must Be Confirmed By Site Visit and/or Survey)*
- Authority has been Provided by Legislation*

DOE AUTHORITY IS NEGATED IF

- Site is Currently Licensed to Conduct the Same Activities Conducted under the AEC/MED Contracts
- Site is Clearly under the Jurisdiction of Another Remedial Action Program (NRC Formerly Licensed Sites, Mill Tailings Act, Super Fund)

STEP 3 — DECISION POINT: DOE Division of Facility and Site Decommissioning Projects (DFSD) Determines Need for Additional Investigations

CATEGORIZE SITES

SCHEDULE SITE VISITS FOR SITES WHERE

- DOE has Authority for Remedial Action and Potential for Contamination is Significant
- DOE has Authority for Remedial Action and Contamination is Possible
- An Imminent Hazard May Exist

SCHEDULE WIDE-AREA SURVEYS (AERIAL OR SCANNING) IF

- Records or Survey Data Indicate Presence of Offsite Contamination and
- Contamination is such that it can be Identified by Wide-Area Surveys

HOLD SITES FOR FUTURE CONSIDERATION IF

- Contamination is Possible
- Insufficient Data Collected to Date to Establish Authority or
- Sites not Covered Under Another Program

(These Sites are Periodically Reviewed for Reclassification and/or Inclusion Through Legislation)

TERMINATE ACTIVITIES IF

- Potential for Contamination is Low
- DOE has no Authority for Remedial Action
- Site is Covered by Other Program or
- Site is Licensed

STEP 4 — INITIAL INVESTIGATIONS

COMPLETE REQUIRED SITE VISITS

- Determine Site Use
- Identify if Imminent Hazard Exists
- Conduct Screening Survey
- Prepare Preliminary Survey Report

Offsite Contamination

IMPLEMENT MOBILE GAMMA SCANS OR AERIAL SURVEYS

- Analyze Data
- Determine if Offsite Contamination is Present

No Contamination

Document Findings

STEP 5 — DECISION POINT: DOE Division of Facility and Site Decommissioning Projects (DFSD) Determines Need for Survey Data or Remedial Action and Priority

INCLUDE IN REMEDIAL ACTION PROGRAM IF

- Imminent Hazard Exists or
- Records Indicate Significant Potential for Contamination, Survey Data Indicates Radiation Clearly Above Guidelines, and DOE has Authority

INCLUDE IN RADIOLOGICAL SURVEY PROGRAM AND SCHEDULES

First Priority if DOE has Authority and (1) Survey Data Indicates Contamination is Possible and Records Suggest Potential for Significant Contamination or (2) Survey Data Identifies Radiation Clearly Above Background and Records Indicate Contamination could be due to MED AEC Work

Second Priority if DOE has Authority and Survey Data Indicates Contamination may be Present and Records Indicate Contamination due to MED AEC Work may be Possible

Third Priority if Records have not Clearly Established DOE Authority but Survey Data Identifies Radiation Levels Clearly Above Background and Records Indicate Potential for Contamination is Significant and may be due to MED AEC Work

TERMINATE ACTIVITIES IF SCREENING SURVEY IS SUFFICIENT TO CERTIFY CONDITION OF SITE IS SUITABLE FOR UNRESTRICTED USE, IS OTHERWISE LICENSED OR CONTROLLED BY APPROPRIATE RESTRICTIONS OR IF NO DOE AUTHORITY FOR REMEDIAL ACTION EXISTS.

records. While input from individuals and organizations is actively sought and has provided much useful data, MED/AEC operating records provide, by far, the more usable data. Records associated with MED and AEC operations stored at various DOE and contractor records centers, the National and Regional Archives, and other agency records centers (such as NRC license records) located throughout the country, are scanned to determine if they are pertinent to the FUSRAP investigations. Records groups identified as possible sources of data are reviewed and available contracts, operating records, and records of previous radiological surveys are assembled. The level or detail of the reviews for specific groups of records depends on the importance of the records to the program. The more likely that new or additional data will be found in a specific set or group of records the more detailed the review of the records will be. Information from these sources is used to develop a list of potential FUSRAP sites that is updated as new data is collected. Ownership data are collected, wherever possible, especially for those sites determined to be highly probable candidates for FUSRAP.

In some cases, copies of pertinent materials are made and maintained for the record; in other cases, the location and a general description of the records are recorded. A data management system is utilized to keep track of records reviewed, identified, and collected.

Step 2 - Historical Data Analysis

During this step, site-specific data collected during records searches and investigations are reviewed and analyzed by the contractor to determine the potential for contamination and DOE authority to conduct remedial action at the site. Potential for contamination is considered significant if the records indicated that: (1) the MED/AEC onsite operations were large, that is conducted over many years and/or the contractor processed large quantities of material; (2) the site had a history of onsite burial of radioactive

material; or (3) radiological data suggests the site is contaminated and/or input from cognizant individuals suggests that the site is contaminated. Contamination is considered possible if the historical data indicates AEC operations could have resulted in the site being contaminated and there is little or no data to indicate the site was ever decontaminated. Potential for contamination is considered low or improbable if only small quantities of radioactive materials were handled, work on the site for MED/AEC for a very short period of time, and/or previous surveys adequately demonstrate decontamination was accomplished. Experience suggests that, for the most part, the potential for contamination is somewhat proportional to the quantities of data or records identified for a specific site, i.e. the more material processed at a site the more records were generated during shipping, billing, processing, etc. As a result, unless there is evidence to suggest otherwise, if only small amounts of information can be identified on a specific site, it is normally assumed that the site only operated for a short period of time or used small quantities of active material.

Generally, only sites in the first two categories will be considered for radiological survey or the remedial action program. Those sites having low potential for contamination will normally be eliminated from the program.

The contractor will also review and analyze the records and assemble materials that provide information regarding DOE authority for remedial action. The contractor will interface with DOE General Counsel to obtain guidance regarding pertinent material needed to determine if authority exists and will provide available records to the General Counsel's office to obtain preliminary findings to be used in the contractor's recommendation for inclusion. The recommendation report will include a brief description of the former activities conducted at the site and those data used as a basis for the recommendations provided in the report. Those recommendations or

findings of the contractor will indicate the potential for residual radioactive material being found at the site and if DOE has existing authority to conduct remedial action at the site. Sites for which there is potential for contamination but no DOE authority has been established are handled in several ways or categories. The first category of sites are those for which it is clear that DOE has no existing authority or that it is unlikely that additional records review will identify any information to provide such authority. The states and or other Federal agencies, as appropriate, are provided information on the sites in this category so that they can take appropriate actions. These sites are eliminated from FUSRAP. The other group includes those sites for which continuing records reviews may provide additional data on which to base an authority determination. Sites in this category are held until there is sufficient data to provide authority or until the likelihood of identifying additional pertinent records is sufficiently low that the site is placed in the first group. The contractor will also search records to determine if a needed action should be covered by programs other than FUSRAP.

Step 3 - Decision Point: DOE Division of Facility and Site Decommissioning Projects (DFSD) Determines Need for Additional Investigation

During this step, DOE-DFSD staff utilize the information assembled and developed by the Technical Support Contractor to determine if the site should be visited and a preliminary onsite survey and/or mobile gamma scan or aerial survey conducted, if activities regarding the site should be terminated, or if the site should be held for future consideration.

Site visits and preliminary surveys will be conducted at sites that could be contaminated with material from MED/AEC operations and for which DOE has authority to conduct remedial action if it is determined to be necessary and/or where an imminent hazard may exist.

Wide area surveys (aerial or mobile gamma scans) will be conducted at sites where records or survey data indicate offsite areas may have been affected and the potential contamination is such that wide area surveys will detect it. Sites are handled as discussed above if contamination is possible but DOE has no authority for remedial action.

DOE may terminate investigations and close files on a site if the potential for contamination is low or the site is clearly under the jurisdiction of a program other than FUSRAP. Similarly, if the site is currently licensed for the same activities conducted under MED/AEC and contamination resulting from licensed work is indistinguishable from that caused by MED/AEC, DOE activities relating to the site will be terminated.

If during this step DOE determines that initial radiological investigations are required, the Technical Support Contractor is tasked to identify the current site owner and a site contact if the information is not already available. DOE selects and assigns a survey contractor(s) to conduct the required onsite investigations, then notifies the owner and makes arrangements for site visits. For sites in the Hold for Future Consideration or Terminate Activity categories, no owner contact will be needed unless the owner was previously made aware of the investigations. Sites in the Hold for Future Considerations category will be assessed as more data are available and recategorized as appropriate.

Step 4 - Initial Radiological Investigations

This step involves site visits and wide area surveys at the sites identified in Step 3 that require additional investigation. These activities are necessary to assemble data required to include or eliminate the site from the program or to determine the need for a more comprehensive radiological evaluation of the site, and to

determine if there is offsite contamination. Site visits are conducted to determine current site use, to determine if an imminent hazard exists, to obtain a preliminary assessment of the radiological condition of the site, and collect data that will be used by DOE to determine if the site can be eliminated from or included in the program without implementing a more comprehensive survey.

The site visit is a multipurpose operation conducted by the assigned survey contractor and, in some cases, a DOE representative. During this visit, the owners or lessees are provided a brief description of the program and the purpose of the investigation. The survey team determines the current use of the site and any expected changes in use. A cursory walk over survey is performed to aid DOE in determining if further activity is needed at the site to ensure that the health and safety of the public is protected, and to ensure that there is no imminent hazard resulting from former MED/AEC operations. The cursory survey may involve gamma, alpha, and/or beta-gamma measurements and some air, water, or soil sampling if felt necessary by onsite survey personnel. The survey contractor should collect sufficient data to provide descriptions of the facility's physical and radiological condition to support a survey plan (if DOE determines that a radiological evaluation survey is needed) or a designation for remedial action (if it is appropriate). This effort should be limited to 1 day or less if possible. Following the visit, the survey contractor will be responsible for providing a draft preliminary survey report to DOE within 1 month (unless otherwise directed) after the visit. The report should contain the contractor's suggestions regarding need for additional surveys.

For those areas determined to need wide area surveying to determine if offsite surveys are needed, two types of surveys may be utilized, aerial and mobile gamma scanning. The aerial survey is conducted using a helicopter or fixed wing aircraft and covers very large areas and identifies the general area(s) of contamination. The

gamma scan is a mobile-based survey conducted along streets, alleys, and other accessible roadways throughout the area. Individual properties having radiological anomalies can be identified using mobile gamma scanning techniques. Following completion of wide area surveys, the survey contractor will prepare a report providing the results of the survey and recommendations concerning the potential for offsite contamination. If there is no indication of offsite contamination, the aerial and/or mobile gamma survey reports may suffice to document the findings and offsite survey efforts will be terminated. If the wide area surveys provide positive indications of the presence of offsite contamination potentially due to DOE predecessor activities, DOE will determine if further radiological characterization is required, or if the area can be designated on the basis of wide area survey data alone. Where additional offsite investigations are required the survey contractor or technical assistance contractor, as appropriate, will be tasked by DOE to identify owners of the properties involved. DOE will notify the owner of the findings and proposed actions if necessary.

Step 5 - Decision Point: DOE Division of Facility and Site Decommissioning (DFSD) Projects Determines Need for Survey Data or Remedial Action

Upon receipt of the site visit and preliminary survey report, DOE reviews the report and recommendations, and, giving due consideration to those data provided by the records searches, will categorize each site either for inclusion in the radiological survey program, or direct inclusion in the remedial action program, or elimination from the program.

Sites will be included for remedial action if DOE has authority for remedial action and data indicate that the potential for contamination is significant and the preliminary survey demonstrates that the contamination is clearly above guidelines. In this case, any additional survey work will be performed during the engineering phase of the task.

If DOE-DFSD determines the site visit and preliminary survey results, along with the historical data are sufficient to verify that the radiological condition of the site is within appropriate guidelines or that the site conditions are controlled by license or appropriate restrictions, the site is eliminated from the program. Sites in this category are processed for elimination and the findings that the radiological condition of the site is acceptable for unrestricted use or, as necessary, for controlled use, are documented and archived.

Sites that can neither be included or eliminated from the remedial action program are scheduled for preinclusion site radiological evaluation surveys to better characterize their radiological condition. When DOE-DFSD assigns a radiological survey contractor to complete the survey, DOE-DFSD will provide the contractor a survey priority for the subject site. Three categories are proposed for assigning survey priorities to sites. First priority sites (those to be scheduled for survey first) are sites for which DOE has authority (through the Atomic Energy Act or Congressional mandate) for remedial action and:

- Preliminary survey data indicate that the site may be contaminated and records suggest the potential for contamination from MED/AEC operations is significant; or
- Survey data identify radiation clearly above background and records indicate it resulted from MED/AEC operations.

Second priority is assigned to sites for which DOE has authority and preliminary survey data indicate contamination is related to MED/AEC work and may be present in quantities that can exceed guidelines.

Third priority is assigned to those sites where that the preliminary data indicate radiation levels are clearly above background; but it is not clear from the data collected that the

radioactivity is from former MED/AEC operations; that is, DOE authority to conduct remedial action is not clear cut. Surveys at third priority sites will be conducted to confirm authority as well as to determine the need for remedial action. If authority is confirmed, the site will be forwarded to the next appropriate step. If the site is contaminated and authority is not confirmed, DOE activities will be terminated, and the appropriate State or Federal agency having jurisdiction will be notified.

RADIOLOGICAL EVALUATION AND DESIGNATION PHASE

The purpose of this phase is to further evaluate the radiological conditions of the site by more comprehensive surveys, to compare the conditions to applicable guidelines and standards, to determine the potential for exposure and, ultimately, to determine if there is a need for remedial action.

During this phase, the radiological surveys are conducted at sites where those data collected during the Preliminary Analysis Phase are not sufficient to include or eliminate sites from the program. As with previous activities, every effort is made to conduct only as much survey work as is necessary to obtain sufficient data to make a designation determination. Determining the extent of survey activity is the responsibility of the radiological survey team leader. In addition, an engineering contractor representative(s) may work with the survey contractor(s) both before and during the survey(s) to ensure the data collected will be of use for engineering work that may be needed. In some cases, where agreed upon between DOE-DFSD and the DOE Oak Ridge Operations Office Technical Services Division (OR-TSD), the comprehensive survey will be thorough enough to provide the basis for the engineering bid request for remedial action.

The radiological evaluation and designation phase of the program contains two steps: the Radiological Evaluation Survey for

Designation and the Decision Point (see Figure II, Step 1 and Step 2). However, the radiological evaluation survey is further divided into two subelements.

Step 1 - Radiological Evaluation Survey for Designation

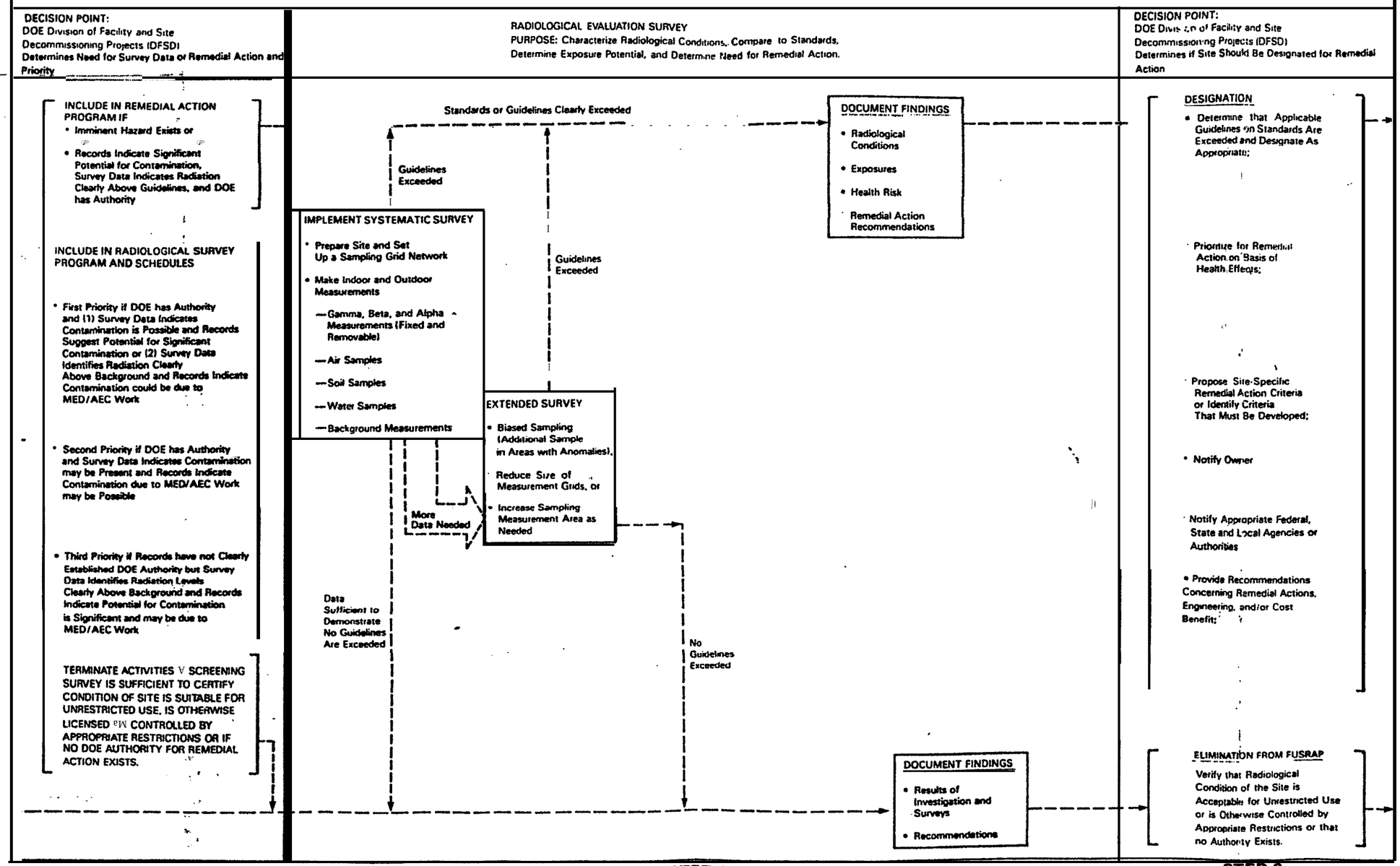
The radiological evaluation survey is subdivided into (1) Systematic and Extended Survey, the onsite survey effort; and (2) Document Findings, the report preparation effort. The onsite survey effort is organized in stages that increase in complexity as they proceed from left to right on the flow chart (Figure II). Each stage represents a part of the survey program and, if conducted, are conducted as part of the same onsite survey. The radiological survey team leader is responsible for the decision to implement more comprehensive stages of the survey activity. This responsibility includes the decision to conduct the extended survey (i.e., biased measurements) in selected areas of the site or to remove minor contamination as part of the survey.

Systematic and Extended Survey. The systematic stage of the survey is, as its name implies, a radiological survey involving systematic and preplanned sampling and direct radiation measurements over a predesigned grid network. These surveys may be of structures or outside areas. The measurements taken can include:

- Gamma, beta, and alpha scans and grid point measurements (fixed and removable); (grounds, buildings, and/or equipment)
- Air samples and analyses (Grab samples);
- Soil samples and analyses; (surface and subsurface)
- Water samples and analyses; (surface and ground water)and
- Background measurements.

FIGURE II

SUMMARY FLOW SHEET FOR RADIOLOGICAL EVALUATION AND DESIGNATION PHASE OF THE FUSRAP REMEDIAL ACTION PROTOCOL



While the survey may include all or any combination of these measurements, it will primarily be the judgment of the radiological survey team leader to determine which and how many measurements are needed. The survey team leader will interact with the engineering contractor representative* as required in planning the survey and will provide a survey plan to DOE-DFSD prior to the survey. This plan will document the measurements to be performed during the systematic survey and briefly indicate under what conditions the extended effort (biased sampling) will be completed. Whenever possible, survey results will be forwarded for final analysis and recommendations as to inclusion or elimination based on the results of the systematic stage of the survey. This decision will be based on or guided by pre-established criteria approved by DOE-DFSD (Appendix B). For isotopes other than radium-226 and thorium isotopes, the soil concentration limits must be calculated (Appendix B). This calculation is done by the radiological support contractor with the assistance of the criteria development contractor (ANL). At some future time, EPA is expected to issue guidelines or standards for residual radioactive materials in the environment. These guidelines will be applied as appropriate.

Where systematic surveys do not provide sufficient data to support this decision, based on indicated action levels, the survey will be extended. The decision whether or not to subject the property to more comprehensive data collection (biased sampling) is made in the field by the radiological survey team leader. These judgments by the radiological survey team leader are important to the success of this approach to the survey process and require the presence of a well-qualified survey team leader.

*Engineering contractor is the Formerly Utilized Sites Remedial Action Program Management Contractor (PMC).

As indicated, the survey is extended to include more detailed measurement techniques only when the systematic effort cannot provide sufficient data to determine if the site exceeds applicable guidelines. The extended survey may include:

- Additional gamma and beta-gamma measurements over a smaller grid to more clearly identify the extent of the contamination;
- Alpha measurements (fixed and removable) of floors and walls and, in some cases, ceilings to define contamination in or on building materials to provide information regarding surface contamination;
- Sampling of building material to assist in defining the source of the contamination and in determining if it is derived from MED/AEC activities;
- Radon and radon daughter monitoring or sampling for other radionuclides in the air over several days to determine if action levels are exceeded;
- Additional soil sampling and subsurface sampling in areas where anomalies may exist;
- Surface and ground water sampling on and/or off the site; and
- Air sampling on and off the site.

It is essential that the extended survey be detailed enough to determine if the condition of the site can be certified to meet guidelines or if the site must be included in the remedial action program.

Document Findings. If, after the evaluation survey the survey contractor believes the site radiological conditions meet established criteria for the site, the contractor should document its findings, including the results of the survey and the description of any material removed from the site. The report should include the survey contractor's recommendations regarding additional DOE or government involvement at the site. The survey contractor will similarly document the results of the surveys for the sites that contain

radioactive residues that exceed appropriate guidelines or standards. In addition to documenting the sites radiological condition and remedial action recommendations, these reports should briefly assess the potential for human exposure and associated health effects or risks.

Step 2 - Decision Point: DOE-Division of Facility and Site Decommissioning (DFSD) Projects Determines if Site Should Be Designated for Remedial Action

During this step, DOE-DFSD staff will review all the data collected on each site and determine whether the site should be included or eliminated from the remedial action program.

If DOE-DFSD determines that radiation levels at the site exceed applicable guidelines or standards, the site will be designated for remedial action by notification from the Director of the Office of Remedial Action and Waste Technology to the Manager of Oak Ridge Operations Office. This designation provides the FUSRAP office in Oak Ridge (OR-TSD) the authority to proceed with the remedial action process. Remedial measures to be considered for a designated site will include restricted use and stabilization on site as well as decontamination of the site. As part of the designation provided to OR-TSD, DOE-DFSD will assign a remedial action priority to the site.* Other guidance will be provided by DOE-DFSD to OR-TSD with the site

*Headquarters will assign each designated site a high, medium, or low priority for remedial action. (see Appendix C) These priorities are assigned considering the potential for public exposure to radiation (dose), the potential for migration of the contaminants, and property use. The final remedial action scheduling priorities determined by OR-TSD with approval from DOE-DFSD take into account the designation priorities as well as other factors including but not limited to: Congressional mandates, availability of a disposal site, coincidence (proximity of projects), available funding and so forth.

designation as may be appropriate; e.g., criteria for remedial action, remedial action options to be considered, and cost/benefit considerations. Simultaneous with designation of the site, DOE-DFSD will notify the owner of the site and appropriate state, local, and Federal agencies and authorities of the findings and plans. In all cases the Department will notify the Environmental Protection Agency of designation actions.

If DOE-DFSD determines from review of the survey data that the site meets the applicable guidelines the findings will be documented and archived according to this protocol. If the site does not meet the DOE criteria but for one of the reasons stated above cannot be included in FUSRAP, the appropriate Federal or state agency will be notified to insure that proper consideration will be given to the site under other assessment efforts.

ENGINEERING AND REMEDIAL ACTION PHASE

The Engineering and Remedial Action Phase of this protocol encompasses conceptual and preliminary engineering activities as well as other activities necessary for the completion of the remedial action and establishment of the disposal site. The activities are to:

- Define and evaluate options for remedial action;
- Obtain required site-specific environmental and radiological characterization data;
- Select the preferred and alternative remedial actions to be assessed during the National Environmental Policy Act (NEPA) analysis;
- Identify environmental impacts and mitigating measures to be assessed during the NEPA analysis;
- Select the preferred remedial action option;
- Prepare the final engineering design (Title II) of the options;

- Implement the selected remedial action and waste disposal action; and

- Prepare the final report and assemble material for the certification docket (see Appendix D).

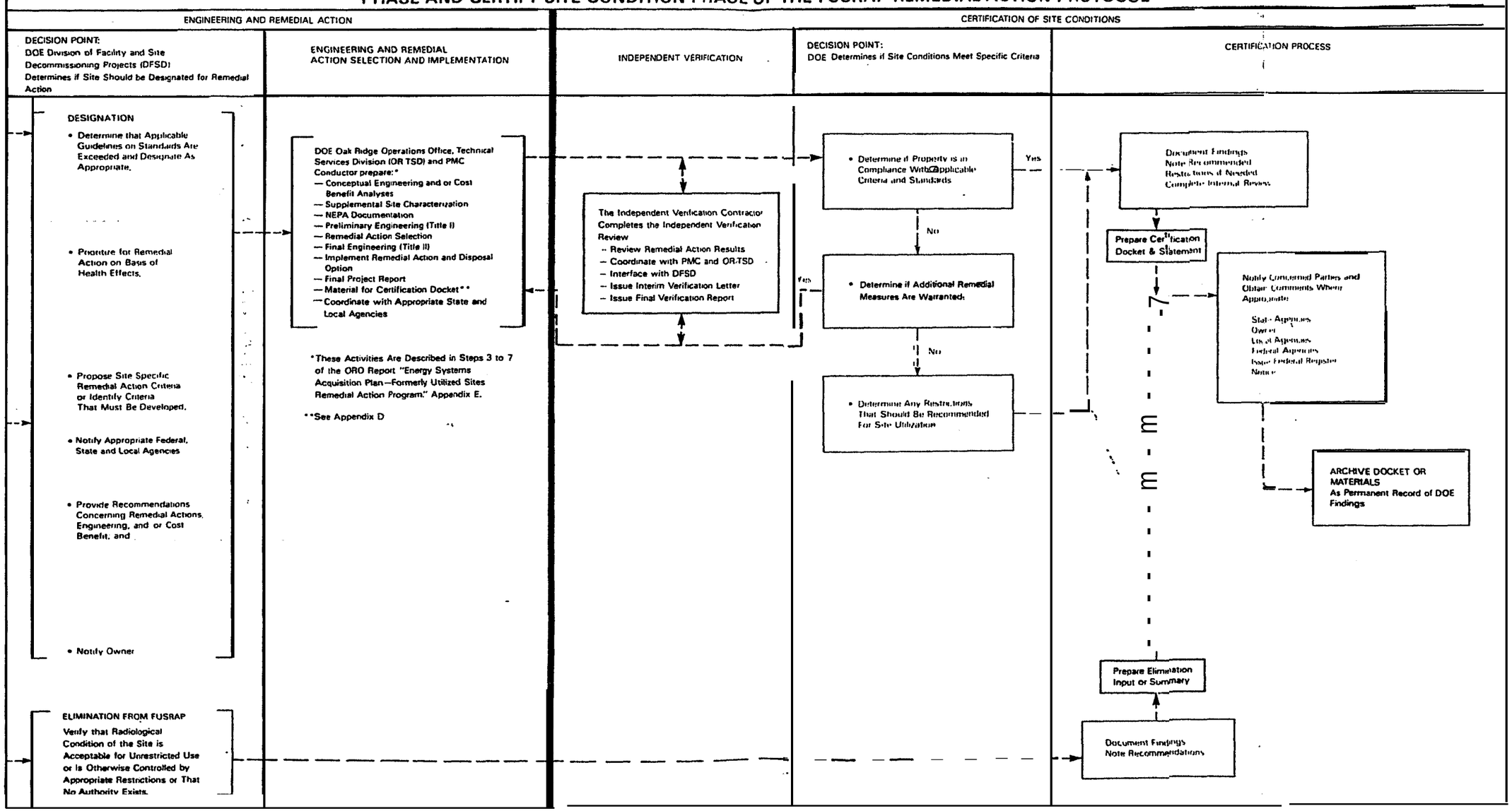
Implementation of this phase (Figure III) is the responsibility of the OR-TSD, the FUSRAP Project Management Contractor (PMC), and the FUSRAP NEPA Process Contractor. More detail is presented in the OR report, "Energy Acquisition Project Plan - Formerly Utilized Sites Remedial Action Program." The general flow chart of activities associated with this phase are shown in Appendix E (steps 3 through 7). The need for and level of preremedial action analyses and preliminary engineering is dependent on many factors including institutional and other nontechnical factors that may dictate the final selection of remedial action options. In such cases, the preparation of certain documents and/or such things as geological investigations may not be required. Decisions regarding the level and need for site-specific studies will be made by OR-TSD with input as needed from DFSD. OR-TSD will provide DOE-DFSD a site-specific project completion report for each remedial action project and prepare a certification docket* for the site.

OR-TSD will interface with DOE-DFSD on all key decisions such as remedial action selection and will supply periodic program status reports. Accomplishment of site decontamination to meet unrestricted use criteria or the achievement of site restrictions and adequate institutional control of residual contamination is the responsibility of OR-TSD.

*The contents of the certification docket are discussed in Appendix D and in the FUSRAP Certification/Verification Supplemented Protocol.

FIGURE III

SUMMARY FLOW SHEET FOR THE ENGINEERING AND REMEDIAL ACTION PHASE AND CERTIFY SITE CONDITION PHASE OF THE FUSRAP REMEDIAL ACTION PROTOCOL



CERTIFICATION OF SITE CONDITION PHASE

The Certification Phase is the responsibility of DOE-DFSD and OR-TSD. It utilizes data from the Remedial Action Phase as well as the other phases of the protocol especially the post-remedial action report or project completion report and involves three interrelated steps:

- o Independent verification of the remedial action
- o Decision on the adequacy of the remedial action
- o Certification process
  - Notification of concerned parties and the issuing of a Federal Register Notice and
  - Completion of the Certification Docket and archiving of the docket

These activities are described in detail in the Verification and Certification Protocol (Supplement 2 to this Protocol).

Step 1 - Independent Verification

An Independent Verification Contractor (IVC) contracted by DFSD, reviews the remedial action activities and conducts verification surveys as necessary to confirm the adequacy of the remedial action and/or the procedures used by the PMC to certify the site's condition. The IVC coordinates with the PMC and OR-TSD during the verification activity, but, is managed and contracted by DFSD to maintain independence and insure no conflict of interest. An interim verification letter is provided by the contractor to OR-TSD and DFSD upon completion of the initial analysis of the remedial action at a specific site within four weeks after completion of the remedial action. The final verification report is submitted sometime thereafter.

Step 2 - Decision Point: DOE Determines If Site Conditions Meet Specific Criteria for the Remedial Action

On the basis of the data provided during and after the remedial action by the PMC including the Post-Remedial Action Report and the information provided by the IVC, OR-TSD, with approval from DFSD, determines if the site was adequately decontaminated and meets DOE guidelines. This decision point is actually a continuous process that is conducted in conjunction with the verification activity and the certification process steps. DOE interacts regularly with the PMC and the IVC during the conduct of the remedial action and the post-remedial action and verification reviews and surveys. This interaction is necessary to insure that any conflicts or discrepencies that are identified are expeditiously resolved. The preparation of the certification docket, certification statement and associated draft Federal Register notice is conducted during the decision process. Any changes required in these documents as a result of the decision are implemented as part of the certification process step.

If the remedial action was accomplished adequately, the site certification process is completed. If the remedial action did not bring the site in compliance with criteria, DOE will determine whether further remedial action is needed or warranted and will provide appropriate direction to the PMC.

Step 3 - Certification Process

As soon as possible after the determination is made that the site will be certified (the remedial action is complete), OR-TSD provides the owner of the site with interim notification that the remedial action is complete and that a certification package is being prepared. In general, the notification of the concerned parties is the responsibility of OR-TSD as is the preparation of the certification statement (required to officially approve the remedial

action) and the draft Federal Register notice. Once approved by the DOE Oak Ridge Chief Counsel's Office and DOE Headquarters (the Office of Management and Administration (MA) and DFSD) the Federal Register notice is issued through DFSD in Washington.

The Certification Docket (Appendix D) is prepared by OR-TSD and the certification statement is signed at the Oak Ridge Field Office. Final approval is required through DFSD. DFSD will arrange to archive the Certification Docket and supporting data as a permanent record of the DOE findings and radiological condition of the site. DFSD will also have the information placed in the DOE Public Reading Room in Washington, D.C., for general availability to the public. Distribution of the dockets to other agencies (Federal, state, or local) as necessary, is made by OR-TSD. The Verification and Certification Protocol (Supplement No. 2 to this protocol) and Appendix F (Public Availability and Archiving of FUSRAP Records) provide additional information.

BIBLIOGRAPHY OF DOE SUPPORTING GUIDANCE DOCUMENTS

Formerly Utilized Sites Remedial Action Program, Designation/Elimination Protocol--Supplement No. 1 to the FUSRAP Summary Protocol, November 1985.

Formerly Utilized Sites Remedial Action Program, Verification and Certification Protocol--Supplement No. 2 to the FUSRAP Summary Protocol, January 1986.

U.S. DOE Energy Acquisition Project Plan (ESAPP), Formerly Utilize MED/AEC Sites Remedial Action Program (FUSRAP) (Revision 1), April 1985, and subsequent revisions.

NOTE

When this Summary Protocol was issued, Appendix B was Revision 1, dated July 1985 of the DOE Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites. Revision 2 (March 1987) of these guidelines have been issued subsequently, and it may be found as Communication Control Number (CCN) 046176.

J. C. Kaffer 3/29/89

J. C. KAFFER
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Oak Ridge, TN 37831

046176

Department of Energy
Washington, DC 20545

JUN 26 1987

Dr. Stanley Lichtman
Office of Radiation Programs (ANR-460)
U.S. Environmental Protection Agency
401 M Street, S.W.
Washington, D.C. 20460

Dear Dr. Lichtman:

Enclosed for your information is a copy of the revised "U.S. Department of Energy (DOE) Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program (FUSRAP) and Surplus Facilities Management Program (SFMP) Sites." These guidelines are used for DOE remedial actions and decommissioning projects conducted under FUSRAP and SFMP. Also enclosed is a summary of the changes between these guidelines dated March 1987 and the previous version dated July 1985. The changes are primarily procedural in nature and were made to simplify field implementation of the guidelines. The guidelines still encompass an annual dose of 100 mrem as the primary dose limit and incorporate the Department's as low as reasonably achievable (ALARA) philosophy through all aspects of the program.

We are presently updating the manual for implementing the residual radioactivity guidelines and hope to have it completed later this year. In the interim, the Department will continue to use the September 1985 version of the manual along with interim guidance for implementing the updated hot spot criteria. If you have any questions regarding these guidelines and their implementation, call Andrew Wallo of my staff at 301-353-5439.

Sincerely,

J. Fiore

James J. Fiore, Director
Division of Facility and Site
Decommissioning Projects
Office of Nuclear Energy

2 Enclosures

Summary of Changes to:
U.S. Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and Remote SFMP Sites.

INTRODUCTION:

The revised U.S. DOE Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program (FUSRAP) and Remote Surplus Facilities Management Program (SFMP) Sites dated March 1987, comply with DOE's standards for protecting the public and are effectively the same as the previous version of the Guidelines. They are consistent with recommendations of the International Commission on Radiological Protection (ICRP) and relevant and applicable Federal/state rules or regulations. The Guidelines encompass:

- Basic Dose Limits to be applied to remedial action projects.
- Residual Radioactive Material Guidelines for Remedial Actions.
  - --Generic and derived guidelines for concentrations of radionuclides in soil.
  - --Hot Spot Limits for radionuclides in soil.
  - --Radon in air guidelines.
  - --External Gamma radiation guidelines.
  - --Surface contamination guidelines for buildings and equipment.
  - --Air and water guidance.
- Authorized Limits for remedial actions (Site Specific Cleanup Limits).
- Guidance for Control of Residual Radioactive Material.
  - --Interim Storage.
  - --Long-term Management.
- Provisions for Supplemental Limits and Exceptions.
- Sources and references for guidelines.

Differences between the March 1987, version and the previous version of the guidelines are generally procedural in nature or are editorial changes made to clarify certain aspects of the guidelines that have been misinterpreted in the past. The procedural changes relate to the implementation of guidance for limiting hot spots and the exceptions or supplemental limit provisions of the document. The guidance for hot spots were modified to include an approach that is more conservative but can be more easily implemented in the field. The exceptions section of the previous version was changed to differentiate exceptions (deviations from the generic or derived guidelines that require restrictions on site use to ensure compliance with DOE standards) from supplemental limits (deviations from site specific generic or derived guidelines that occur due to specific considerations and do not require restrictions on site use to ensure compliance with DOE standards).

General discussion and considerations:

The guidelines for radionuclides in soil are generic or derived depending on the radionuclides of concern. For radium and thorium, generic guidelines have been adopted from the EPA Standards (40 CFR 192) for uranium mill tailings remedial actions. These EPA standards were determined to be appropriate for many FUSRAP sites and are generally applied at sites contaminated with uranium and thorium ores or tailings. Conditions at the FUSRAP site are sufficiently similar to uranium mill tailings sites and associated vicinity properties being remediated under the Uranium Mill Tailings Remedial Action Program (UMTRAP) to ensure equivalent levels of protection. However, in the Residual Radioactive Material Guidelines, the Department has recognized the limitations of the EPA standards for radium and thorium in soil in certain contamination situations. These DOE Guidelines provide specific guidance for the treatment of hot spots and take into account multiples or mixtures of radionuclides for situations when radionuclides other than or in addition to radium or thorium and their associated decay products are present. These additions effectively result in a more conservative guideline. In addition, the Department's guidance requires application of the As Low As Reasonably Achievable (ALARA) philosophy in the field which may be applied but is not required under the UMTRA standards.

In other areas as with soil contamination, the guidelines are consistent with UMTRAP and other EPA standards; however, the use of UMTRAP standards is not defacto. Care is taken to ensure appropriate application of standards or limits. Dose evaluations and engineering alternative analyses are applied to storage and disposal sites. In general, materials of higher concentrations are provided greater isolation as required to ensure the dose standards and other appropriate limits are consistently achieved. Similarly, application of the exceptions provisions of the DOE Guidelines require equal and in some cases more detailed analysis and evaluation than is required for UMTRAP Sites under EPA Standards. In order to obtain approval for a specific exception or supplemental limit, dose evaluations, assessments of potential impacts, and/or cost-benefit analyses are required. These analyses must go through a formal approval procedure which include multiple reviews to by DOE field offices and headquarters ensure that the public and environment are adequately protected.

The derived limits established via procedures described in the Guidelines and supporting material comply with DOE Standards for protecting the public and are consistent with ICRP guidance which is generally being applied to most areas of radiation protection. ICRP states that the lifetime dose to a member of the general public should correspond to a maximum 100 mrem/year limit of life-long whole body exposure from all sources, excluding background and medical related doses. ICRP-26 indicates that in most cases that limit can be achieved by application of the 500 mrem/year dose limit. The report suggests that the application of the 500 mrem/year limit is likely to produce dose equivalents of less than 50 mrem/year. The report also indicates that maximizing assumptions usually made in selecting the critical groups result in actual doses to the most highly exposed individuals of values less than that postulated.

The Department's Guidelines use the 100 mrem/year value as a limit for life time exposure (instead of the 500 mrem/year limit) and derive allowable concentrations postulating worse case plausible use scenarios. In general, while plausible, these scenarios are highly unlikely and actual or likely use scenarios would produce potential doses much less than 100 mrem/year, probably on the order of a few mrem/year. As the intent of the ICRP guidance is to ensure that actual doses to the general public do not exceed 100 mrem/year over a lifetime and not to limit potential worst case doses, the Department's Guidelines clearly achieve this goal. The application of ALARA to the Guidelines, reduce exposures to levels that are still further below the postulated limit.

In summary, the DOE Guidelines represent implementable limits for residual radioactivity in the environment that comply with DOE Standards for protecting the public, meet the intent of ICRP guidance, and are consistent with existing guidelines and standards. The DOE Guidelines for residual radioactive material also conform to the DOE policy for implementing radiation protection activities in a manner that is as low as reasonably achievable.

Major Changes:

As indicated previously the primary changes to the Guidelines occurred in two areas. These changes are outlined below.

Hot Spot Criteria:

The July 1985 version of the Departments residual radioactive material guidelines required that "guidelines for local concentrations" be applied to all areas less than 30 square meters found to exceed an authorized limit or guideline for a 100 square meter area by a factor of 3 or greater. The factor of 3 requirement was not a clean-up limit, but rather a screening or action level. Basically, specific dose calculations were required for each area less than 30 square meters if the area exceeded 3 times the 100 square meter soil concentration guideline or limit. These calculations were found to be impractical for field application due to the requirement to perform dose calculations in the field.

In addition, analysis of the screening value (factor of 3) indicated there was a remote possibility that that value would not provide adequate protection of the general public for certain radionuclides for areas of contamination exceeding about 15 square meters. As a result the Department established a working group whose purpose was to develop hot spot criteria that would:

- Protect the public and environment.
- Be consistent with the guidelines.
- Facilitate field implementation.

The working group recommended that the hot spot criteria be defined as a multiple of the soil guidelines or authorized limits which represent the general soil concentration for radionuclides permitted over a 100 square

meter area, and that the factor be the squareroot of (100/A) where A is the area of the hot spot. This factor was applied to several typical radionuclides found at DOE remedial action sites including selected alpha, beta, and gamma emmitters using the dose estimating methodology provided in the procedures manual, the "supplement" to the Guidelines. The assessment indicated that the $(100/A)^{1/2}$ factor is sufficiently conservative to ensure the basic 100 mrem/year dose limit is maintained. For field application, the working group recommended that the hot spot limit be implemented in four discrete steps as shown in the attached table. The use of discrete steps adds further conservatism to the approach as well as simplifying field application. The attached table provides a summary of the working groups recommendations for the criteria and its implementation. Theoretically, the $(100/A)^{1/2}$ limit could allow very small areas of contamination to have high concentrations of radionuclides. While this does not, in general, pose a problem to most DOE remedial action sites due to the nature of the residues and DOE's ALARA philosophy, a maximum factor of 30 times the authorized limit was established as limit for small areas of contamination.

On December 2, 1986, the Department approved the revised hot spot criteria and directed that it be incorporated in the FUSRAP and SFMP Guidelines for residual radioactivity. The revised guidelines adopted this procedure, but also allow for the use of the specific dose calculations where it is appropriate.

Supplemental Limits and Exceptions:

The intent and mechanism for applying the exceptions provisions of the previous guidelines were in some cases confusing. The revision of this section and the separation of exception provisions into two categories exceptions (areas requiring controlled or restricted use to comply with the 100 mrem/year dose limit) and supplemental limits (for areas requiring no restrictions to comply with the dose limit) were done in an attempt to clarify the process. The revision stresses the following points:

-Exceptions or supplemental limits are generally for use at a portion of a site or vicinity property where specific circumstances dictate the guidelines or authorized limit established for the entire site and vicinity properties are not appropriate for the specific area of concern.

-Every effort should be made to minimize the use of exceptions or supplemental limits.

-Supplemental limits must be justified on a case-by-case basis using site specific data and must consider ALARA policy.

APPLICATION OF HOT SPOT GUIDELINE

1. The method for determining Hot Spot Limits, which is based on the 100 mrem/year Dose Limit, as described in the FUSRAP procedures manual, shall still be applicable for determining allowable concentrations of radionuclides under inhomogeneous soil contamination conditions. However, the following approach, more appropriate for field applications, may be used in place of the Dose Limit method and is recommended for general applications.

2. For the alternative approach, the basic Hot Spot Limits will be calculated for each specific site by (see attached figure):

   $Shg = Sg * (100\ m^2/A)^{1/2}$

   where, Shg = the Hot Spot Limit (pCi/gram)
   Sg = the Authorized Limit for a specific site (pCi/gram)
   A = the area of the hot spot in square meters
   $(100/A)^{1/2}$ is the hot spot multiplication factor.

3. The limits shall be applied in the field over ranges of area with the factors being constant over a given area. The ranges and factors to be used are:

Range	Factor (Multiple of Authorized Limit)
<1 m^2	10*
1 - <3 m^2	6
3 - <10 m^2	3
10 - 25 m^2	2

   *Areas less than one square meter are to be averaged over the one square meter and that average shall not exceed ten times the Authorized Limit.

4. The average Authorized Limit is considered adequate to protect the public for areas larger than 25 square meters; hence, no special Hot Spot Limits are required for areas larger than 25 square meters.

5. Averaging of hot spots less than or equal to 25 square meters shall be done only over the local hot spot area.

6. Every reasonable effort shall be made to identify and remove any source which has a concentration of a radionuclide exceeding 30 times the Authorized Limit irrespective of area.

U.S. DEPARTMENT OF ENERGY GUIDELINES FOR RESIDUAL RADIOACTIVE MATERIAL AT FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM AND REMOTE SURPLUS FACILITIES MANAGEMENT PROGRAM SITES

(Revision 2, March 1987)

A. INTRODUCTION

This document presents U.S. Department of Energy (DOE) radiological protection guidelines for cleanup of residual radioactive materials and management of the resulting wastes and residues. It is applicable to sites identified by the Formerly Utilized Sites Remedial Action Program (FUSRAP) and remote sites identified by the Surplus Facilities Management Program (SFMP).* The topics covered are basic dose limits, guidelines and authorized limits for allowable levels of residual radioactive material, and requirements for control of the radioactive wastes and residues.

Protocols for identification, characterization, and designation of FUSRAP sites for remedial action; for implementation of the remedial action; and for certification of a FUSRAP site for release for unrestricted use are given in a separate document (U.S. Department of Energy 1986) and subsequent guidance. More detailed information on applications of the guidelines presented herein, including procedures

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* A remote SFMP site is one that is excess to DOE programmatic needs and is located outside a major operating DOE research and development or production area.

for deriving site-specific guidelines for allowable levels of residual radioactive material from basic dose limits, is contained in "A Manual for Implementing Residual Radioactive Material Guidelines" (U.S. Department of Energy 1987) referred to herein as the "supplement".

"Residual radioactive material" is used in these guidelines to describe radioactive materials derived from operations or sites over which the Department of Energy has authority. Guidelines or guidance to limit the levels of radioactive material to protect the public and environment are provided for: (1) residual concentrations of radionuclides in soil material, (2) concentrations of airborne radon decay products, (3) external gamma radiation level, (4) surface contamination levels, and (5) radionuclide concentrations in air or water resulting from or associated with any of the above.

A "basic dose limit" is a prescribed standard from which limits for quantities that can be monitored and controlled are derived; it is specified in terms of the effective dose equivalent as defined by the International Commission on Radiological Protection (ICRP 1977, 1978). The basic dose limits are used for deriving guidelines for residual concentrations of radionuclides in soil material. Guidelines for residual concentrations of thorium and radium in soil, concentrations of airborne radon decay products, allowable indoor external gamma radiation levels, and residual surface contamination concentrations are based on existing radiological protection standards or guidelines (U.S. Environmental Protection Agency 1983; U.S. Nuclear Regulatory Commission 1982; and Departmental Orders). Derived guidelines or limits based on the basic dose limits for those quantities are only used when the guidelines provided in the existing standards cited above are shown to be inappropriate.

A "guideline" for residual radioactive material is a level of radioactivity or of the radioactive material that is acceptable if the use of the site is to be unrestricted. Guidelines for residual radioactive material presented herein are of two kinds: (1) generic,

site-independent guidelines taken from existing radiation protection standards, and (2) site-specific guidelines derived from basic dose limits using site-specific models and data. Generic guideline values are presented in this document. Procedures and data for deriving site-specific guideline values are given in the supplement. The basis for the guidelines is generally a presumed worst case plausible scenario for a site.

An "Authorized Limit" is a level of residual radioactive material or radioactivity that must not be exceeded if the remedial action is to be considered completed and the site is to be released for unrestricted use. The Authorized Limit for a site will include limits for each radionuclide or group of radionuclides, as appropriate, associated with the residual radioactive material in the soil or in surface contamination of structures and equipment, and in the air or water, and, where appropriate, a limit on external gamma radiation resulting from the residual material. Under normal circumstances, expected to occur at most sites, Authorized Limits for residual radioactive material or radioactivity are set equal to guideline values. Exceptional conditions for which Authorized Limits might differ from guideline values are specified in Sections D and F. A site may be released for unrestricted use only if the conditions do not exceed the Authorized Limits or approved supplemental limits as defined in Section F.1 at the time remedial action is completed. Restrictions and controls on use of the site must be established and enforced if the site conditions exceed the approved limits, or if there is potential to exceed the dose limit if the site use was not restricted (Section F.2). The applicable controls and restrictions are specified in Section E.

DOE policy requires that all exposures to radiation be limited to levels that are as low as reasonably achievable (ALARA). For sites to be released for unrestricted use, the intent is to reduce residual radioactive material to levels that are as far below Authorized Limits as reasonable considering technical, economic, and social factors. At sites where the residual material is not reduced to levels that permit release for unrestricted use, ALARA policy is implemented by establishing controls to reduce exposure to levels that are as low as reasonably achievable. Procedures for implementing ALARA policy are discussed in the supplement. ALARA policies,

procedures, and actions shall be documented and filed as a permanent record upon completion of remedial action at a site.

B. BASIC DOSE LIMITS

The basic dose limit for the annual radiation dose received by an individual member of the general public is 100 mrem/year. The internal committed effective dose equivalent, as defined in ICRP Publication 26 (ICRP 1977) and calculated by dosimetry models described in ICRP Publication 30 (ICRP 1978), plus dose from penetrating radiation sources external to the body shall be used for determining the dose. This dose shall be described as the "Effective Dose Equivalent". Every effort shall be made to ensure that actual doses to the public are as far below the dose limit as is reasonably achievable.

Under unusual circumstances it will be permissible to allow potential doses to exceed 100 mrem/year where such exposures are based upon scenarios which do not persist for long periods and where the annual life time exposure to an individual from the subject residual radioactive material would be expected to be less than 100 mrem/year. Examples of such situations include conditions that might exist at a site scheduled for remediation in the near future or a possible, but improbable, one-time scenario that might occur following remedial action. These levels should represent doses that are as low as reasonably achievable for the site. Further, no annual exposure should exceed 500 mrem.

C. GUIDELINES FOR RESIDUAL RADIOACTIVE MATERIAL

C.1 Residual Radionuclides in Soil

Residual concentrations of radionuclides in soil shall be specified as above-background concentrations averaged over an area of 100 sq meters. Generic guidelines for thorium and radium are specified below. Guidelines for residual concentrations of other radionuclides shall be derived from the basic dose limits by means of an environmental pathway analysis using

site-specific data where available. Procedures for these derivations are given in the supplement.

If the average concentration in any surface or below surface area less than or equal to 25 sq meters exceeds the Authorized Limit or guideline by a factor of $(100/A)^{1/2}$, where A is the area of the elevated region in square meters, limits for "Hot Spots" shall also be applicable. These Hot Spot Limits depend on the extent of the elevated local concentrations and are given in the supplement. In addition, every reasonable effort shall be made to remove any source of radionuclide that exceeds 30 times the appropriate soil limit irrespective of the average concentration in the soil.

Two types of guidelines are provided, generic and derived. The generic guidelines for residual concentrations of the Ra-226, Ra-228, Th-230, and Th-232 are:

- 5 pCi/g, averaged over the first 15 cm of soil below the surface
- 15 pCi/g, averaged over 15-cm-thick layers of soil more than 15 cm below the surface

These guidelines take into account ingrowth of Ra-226 from Th-230 and of Ra-228 from Th-232, and assume secular equilibrium. If either Th-230 and Ra-226 or Th-232 and Ra-228 are both present, not in secular equilibrium, the appropriate guideline is applied as a limit to the radionuclide with the higher concentration. If other mixtures of radionuclides occur, the concentrations of individual radionuclides shall be reduced so that 1) the dose for the mixtures will not exceed the basic dose limit, or 2) the sum of the ratios of the soil concentration of each radionuclide to the allowable limit for that radionuclide will not exceed 1 ("unity"). Explicit formulas for calculating residual concentration guidelines for mixtures are given in the supplement.

C.2 Airborne Radon Decay Products

Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property

that are intended for unrestricted use; structures that will be demolished or buried are excluded. The applicable generic guideline (40 CFR 192) is: In any occupied or habitable building, the objective of remedial action shall be, and a reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL.* In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions by DOE are not required in order to comply with this guideline when there is reasonable assurance that residual radioactive materials are not the cause.

C.3 External Gamma Radiation

The average level of gamma radiation inside a building or habitable structure on a site to be released for unrestricted use shall not exceed the background level by more than 20 μR/h and shall comply with the basic dose limit when an appropriate use scenario is considered. This requirement shall not necessarily apply to structures scheduled for demolition or to buried foundations. External gamma radiation levels on open lands shall also comply with the basic dose limit considering an appropriate use scenario for the area.

C.4 Surface Contamination

The generic guidelines provided in the Table 1, Surface Contamination Guidelines are applicable to existing structures and equipment. These guidelines are adapted from standards of the U.S. Nuclear Regulatory

* A working level (WL) is any combination of short-lived radon decay products in one liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

TABLE 1 SURFACE CONTAMINATION GUIDELINES

Radionuclides [2]	Allowable Total Residual Surface Contamination (dpm/100 cm^2) [1]		
	Average [3, 4]	Maximum [4, 5]	Removable [4, 6]
Transuranics, Ra-225, Ra-228, Th-230 Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000 α	15,000 α	1,000 α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5,000 β-γ	15,000 β-γ	1,000 β-γ

1 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 measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

2 Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

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

4 The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

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

6 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 measuring the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. The numbers in this column are maximum amounts.

Commission (1982)* and will be applied in a manner that provides a level of protection consistent with the Commission's guidance. These limits apply to both interior and exterior surfaces. They are not directly intended for use on structures to be demolished or buried, but, should be applied to equipment or building components that are potentially salvageable or recoverable scrap. If a building is demolished, the guidelines in Section C.1 are applicable to the resulting contamination in the ground.

C.5 Residual Radionuclides in Air and Water

Residual concentrations of radionuclides in air and water shall be controlled to levels required by DOE Environmental Protection Guidance and Orders, specifically DOE Order 5480.1A and subsequent guidance. Other Federal and/or state standards shall apply when they are determined to be appropriate.

D. AUTHORIZED LIMITS FOR RESIDUAL RADIOACTIVE MATERIAL

The Authorized Limits shall be established to: 1) ensure that, as a minimum, the Dose Limits specified in Section B will not be exceeded under the worst case plausible use scenario consistent with the procedures and guidance provided, or 2) where applicable generic guidelines are provided, be consistent with such guidelines. The Authorized Limits for each site and vicinity properties shall be set equal to the generic or derived guidelines except where it can be clearly established on the basis of site specific data, including health, safety and socioeconomic considerations, that the guidelines are not appropriate for use at the specific site. Consideration

* These guidelines are functionally equivalent to Section 4 - Decontamination for Release for Unrestricted Use of NRC Regulatory Guide 1.86, but are applicable to Non-Reactor facilities.

should also be given to ensure that the limits comply with or provide an equivalent level of protection as other appropriate limits and guidelines (i.e., state, or other Federal). Documentation supporting such a decision should be similar to that required for supplemental limits and exceptions (Section F), but should be generally more detailed because it covers an entire site.

Remedial actions shall not be considered complete unless the residual radioactive material levels comply with the Authorized Limits. The only exception to this requirement will be for those special situations where the supplemental limits or exceptions are applicable and approved as specified in Section F. However, the use of supplemental limits and exceptions should only be considered if it is clearly demonstrated that it is not reasonable to decontaminate the area to the Authorized Limit or guideline value. The Authorized Limits are developed through the project offices in the field (Oak Ridge Technical Services Division for FUSRAP) and approved by the headquarters program office (the Division of Facility and Site Decommissioning Projects).

E. CONTROL OF RESIDUAL RADIOACTIVE MATERIAL AT FUSRAP AND REMOTE SFMP SITES

Residual radioactive material above the guidelines at FUSRAP and remote SFMP sites must be managed in accordance with applicable DOE Orders. The DOE Order 5480.1A and subsequent guidance or superceding orders require compliance with applicable Federal, and state environmental protection standards.

The operational and control requirements specified in the following DOE Orders shall apply to interim storage, interim management, and long-term management.

a. 5440.1C, Implementation of the National Environmental Policy Act

b. 5480.1A, Environmental Protection, Safety, and Health Protection Program for DOE Operations as revised by DOE 5480.1 change orders and the 5 August 1985 memorandum from Vaughan to Distribution

c. 5480.2, Hazardous and Radioactive Mixed Waste Management

d. 5480.4, Environmental Protection, Safety, and Health Protection Standards

e. 5482.1A, Environmental Safety, and Health Appraisal Program

f. 5483.1A, Occupational Safety and Health Program for Government-Owned Contractor-Operated Facilities

g. 5484.1, Environmental Protection, Safety, and Health Protection Information Reporting Requirements

h. 5000.3, Unusual Occurrence Reporting System

i. 5820.2, Radioactive Waste Management

E.1 Interim Storage

a. Control and stabilization features shall be designed to ensure, to the extent reasonably achievable, an effective life of 50 years and, in any case, at least 25 years.

b. Above-background Rn-222 concentrations in the atmosphere above facility surfaces or openings shall not exceed: (1) 100 pCi/L at any given point, (2) an annual average concentration of 30 pCi/L over the facility site, and (3) an annual average concentration of 3 pCi/L at or above any location outside the facility site (DOE Order 5480.1A, Attachment XI-1).

c. Concentrations of radionuclides in the groundwater or quantities of residual radioactive materials shall not exceed existing Federal, or state standards.

d. Access to a site shall be controlled and misuse of onsite material contaminated by residual radioactive material shall be prevented through appropriate administrative controls and physical barriers--active and passive controls as described by the U.S. Environmental Protection Agency (1983--p. 595). These control features should be designed to ensure, to the extent reasonable, an effective life of at least 25 years. The Federal government shall have title to the property or shall have a long-term lease for exclusive use.

E.2 Interim Management

a. A site may be released under interim management when the residual radioactive material exceeds guideline values if the residual radioactive material is in inaccessible locations and would be unreasonably costly to remove, provided that administrative controls are established to ensure that no member of the public shall receive a radiation dose exceeding the basic dose limit.

b. The administrative controls, as approved by DOE, shall include but not be limited to periodic monitoring as appropriate, appropriate shielding, physical barriers to prevent access, and appropriate radiological safety measures during maintenance, renovation, demolition, or other activities that might disturb the residual radioactivity or cause it to migrate.

c. The owner of the site or appropriate Federal, state, or local authorities shall be responsible for enforcing the administrative controls.

E.3 Long-Term Management

Uranium, Thorium, and Their Decay Products

a. Control and stabilization features shall be designed to ensure, to the extent reasonably achievable, an effective life of 1,000 years and, in any case, at least 200 years.

b. Control and stabilization features shall be designed to ensure that Rn-222 emanation to the atmosphere from the waste shall not: (1) exceed an annual average release rate of 20 $pCi/m^2/s$, and (2) increase the annual average Rn-222 concentration at or above any location outside the boundary of the contaminated area by more than 0.5 pCi/L. Field verification of emanation rates is not required.

c. Prior to placement of any potentially biodegradable contaminated wastes in a long-term management facility, such wastes shall be properly conditioned to ensure that (1) the generation and escape of biogenic gases will not cause the requirement in paragraph b. of this section (E.3) to be exceeded, and (2) biodegradation within the facility will not result in premature structural failure in violation of the requirements in paragraph a. of this section (E.3).

d. Groundwater shall be protected in accordance with Appropriate Departmental orders and Federal and state standards, as applicable to FUSRAP and remote SFMP sites.

e. Access to a site should be controlled and misuse of onsite material contaminated by residual radioactive material should be prevented through appropriate administrative controls and physical barriers--active and passive controls as described by the U S. Environmental Protection Agency (1983--p. 595). These controls should be designed to be effective to the extent reasonable for at least 200 years. The Federal government shall have title to the property.

Other Radionuclides

f. Long-term management of other radionuclides shall be in accordance with Chapters 2, 3, and 5 of DOE Order 5820.2, as applicable.

F. SUPPLEMENTAL LIMITS AND EXCEPTIONS

If special site specific circumstances indicate that the guidelines or Authorized Limits established for a given site are not appropriate for a portion of that site or a vicinity property, then the field office may request that supplemental limits or an exception be applied. In either case, the field must justify that the subject guidelines or Authorized Limits are not appropriate and that the alternative action will provide adequate protection giving due consideration to health and safety,

environment and costs. The field office shall obtain approval for specific supplemental limits or exceptions from headquarters as specified in Section D of these guidelines and shall provide to headquarters those materials required for the justification as specified in this section and in the FUSRAP and SFMP protocols and subsequent guidance documents. The field office shall also be responsible for coordination with the state or local government of the limits or exceptions and associated restrictions as appropriate. In the case of exceptions, the field office shall also work with the state and/or local governments to insure that restrictions or conditions of release are adequate and mechanisms are in place for their enforcement.

F1. Supplemental Limits

The supplemental limits must achieve the basic dose limits set forth in this guideline document for both current and potential unrestricted uses of the site and/or vicinity property. Supplemental limits may be applied to a property or portion of a property or site if, on the basis of a site specific analysis, it is determined that certain aspects of the property or portion of the site were not considered in the development of the established Authorized Limits and associated guidelines for the site, and as a result of these unique characteristics, the established limits or guidelines either do not provide adequate protection or are unnecessarily restrictive and costly.

F2. Exceptions

Exceptions to the Authorized Limits defined for unrestricted use of the site may be applied to a portion of a site or a vicinity property when it is established that the Authorized Limits cannot be achieved and restrictions on use of the site or vicinity property are necessary to provide adequate protection of the public and environment. The field office must clearly demonstrate that the exception is necessary, and the restrictions will provide the necessary degree of protection and that they comply with the requirements for control of residual radioactive material as set forth in Part E of these guidelines.

F3. Justification for Supplemental Limits and Exceptions

Supplemental limits and exceptions must be justified by the field office on a case by case basis using site specific data. Every effort should be made to minimize the use of the supplemental limits and exceptions. Examples of specific situations that warrant the use of supplemental standards and exceptions are:

a. Where remedial actions would pose a clear and present risk of injury to workers or members of the general public, notwithstanding reasonable measures to avoid or reduce risk.

b. Where remedial actions--even after all reasonable mitigative measures have been taken--would produce environmental harm that is clearly excessive compared to the health benefits to persons living on or near affected sites, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that can reasonably be anticipated.

c. Where it is clear that the scenarios or assumptions used to establish the Authorized Limits do not under plausible current or future conditions, apply to the property or portion of the site identified and where more appropriate scenarios or assumptions indicate that other limits are applicable or necessary for protection of the public and the environment.

d. Where the cost of remedial actions for contaminated soil is unreasonably high relative to long-term benefits and where the residual radioactive materials do not pose a clear present or future risk after taking necessary control measures. The likelihood that buildings will be erected or that people will spend long periods of time at such a site should be considered in evaluating this risk. Remedial actions will generally not be necessary where only minc. quantities of residual radioactive

materials are involved or where residual radioactive materials occur in an inaccessible location at which site-specific factors limit their hazard and from which they are costly or difficult to remove. Examples are residual radioactive materials under hard-surface public roads and sidewalks, around public sewer lines, or in fence-post foundations. A site-specific analysis must be provided to establish that it would not cause an individual to receive a radiation dose in excess of the basic dose limits stated in Section B, and a statement specifying the residual radioactive material must be included in the appropriate state and local records.

e. Where there is no feasible remedial action.

G. SOURCES

Limit or Guideline	Source
Basic Dose Limits	
Dosimetry Model and Dose Limits	International Commission on Radiological Protection (1977, 1978)
Generic Guidelines for Residual Radioactivity	
Residual Concentrations of Radium and Thorium in Soil Material	40 CFR 192
Airborne Radon Decay Products	40 CFR 192
External Gamma Radiation	40 CFR 192
Surface Contamination	Adapted from U.S. Nuclear Regulatory Commission (1982)
Control of Radioactive Wastes and Residues	
Interim Storage	DOE Order 5480.1A and subsequent guidance
Long-Term Management	DOE Order 5480.1A and subsequent guidance; 40 CFR 192; DOE order 5820.2

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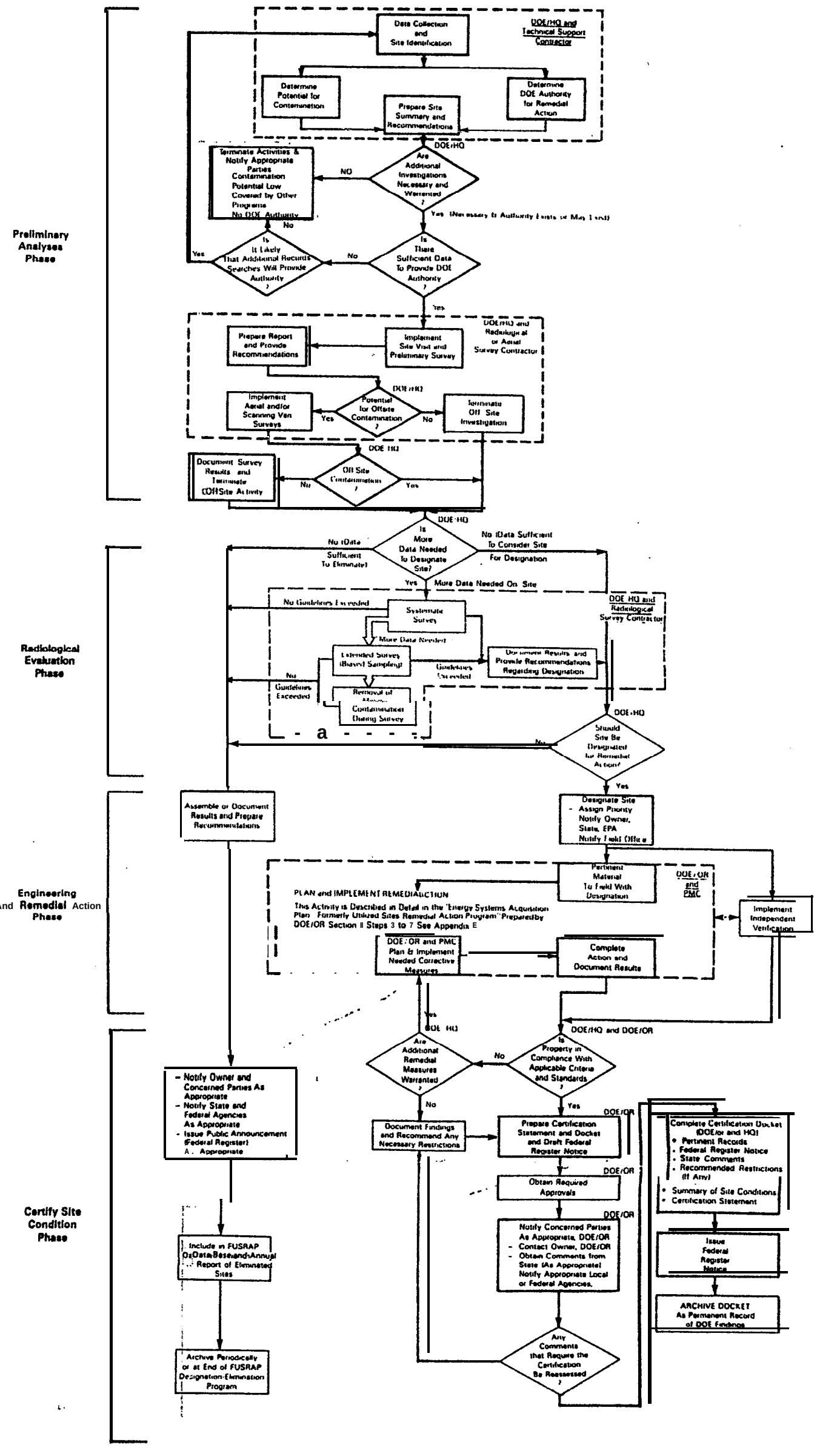
APPENDIX A
FLOWCHART FOR FUSRAP PROTOCOL



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APPENDIX C. DOE FUSRAP PROCEDURE FOR ASSIGNING SITE PRIORITIES

The assessment of potential health effects and the ranking of contaminated sites are complex and must take into account many influencing factors. The major hazard due to radiological contaminants is their potential to increase either the long or short term risk of cancer. The nature of these contaminants must be clearly defined. Furthermore, the risk from all pathways to an exposed individual or population group, as well as such exposure parameters as occupancy factors associated with the contaminated living or working areas and the population density around a contaminated site must be evaluated. Potential for migration of contaminants to the surrounding environs either through the air, water, soil, and the ecosystem and ultimately to man is of major importance.

Analyses to date have identified no site under current use conditions where there is an immediate health hazard; however, over the long term, the potential for accumulated exposure and unacceptable increases in risk do exist.[a] It should be noted, however, that dose and risk estimates completed as part of the assigning of priorities procedure are not absolute estimates. These estimates are

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(a) An unacceptable increase has been tentatively defined as an annual increased risk of getting a fatal cancer in excess of 5 chances in 100,000 per year of exposure. The values represent the approximate increase in risk of contracting a fatal cancer as a result of continuous exposure to the recommended guidelines (500 mrem/y) value for short term exposure (DOE-85) using a dose risk conversion factor of 10^{-7} effects/mrem of dose (ICRP-26). Because this procedure assumes risk to be proportional to dose, the equivalent whole body dose calculated as the sum of weighted internal and external doses (recommendation ICRP-26) can be directly compared to the 500 mrem limit to determine a priority. The short term guideline is appropriate rather than the long term guideline of 100 mrem/year because the implementation of remedial actions to remove material causing the potential exposures are expected to begin in a short period (about 5 years or less following designation).

relative comparisons of the potential for exposure at the specific sites and are intended to be compared to estimates at other designated sites for the purpose of assigning a remedial action priority. The health effects or dose estimates are not intended or necessarily applicable for other uses.

The Department is using a three-category system for ranking contaminated sites based on health effects (see Figure C-1). The categories are:

High

- o Ranking a site as a high priority indicates that the site is contaminated above guidelines, and
  - there is potential for individuals at a site under present use conditions to receive an unacceptable increase in cancer risk,[a] or
  - there is significant potential for a larger group of individuals not directly associated with a site to be exposed to levels of radiation that could increase the number of expected cancers to an unacceptable level,[b] or

(a) See Note (a) on previous page

(b) An unacceptable increase to a group of individuals has been tentatively defined as an annual increased risk of getting a fatal cancer in excess of 1 in 100,000. This value, as the similar one defined for individual risk, is preliminary; it is based on the increased risk that would occur if a group of persons were exposed to the standard for large groups (100 mrem/y, FRC* 1960) over their entire lives. This is the approximate annual risk estimated using the 100 mrem/y standard and a dose risk conversion factor of 10^{-7} effects/mrem of dose from ICRP-26. Because the procedure assumes risk to be proportional to dose, the equivalent whole body dose calculated as recommended in ICRP-26 (the sum of weight internal and external doses) can be directly compared to the 170 mrem dose limit to determine priorities.

*Recommendations of the Federal Radiation Counsel.

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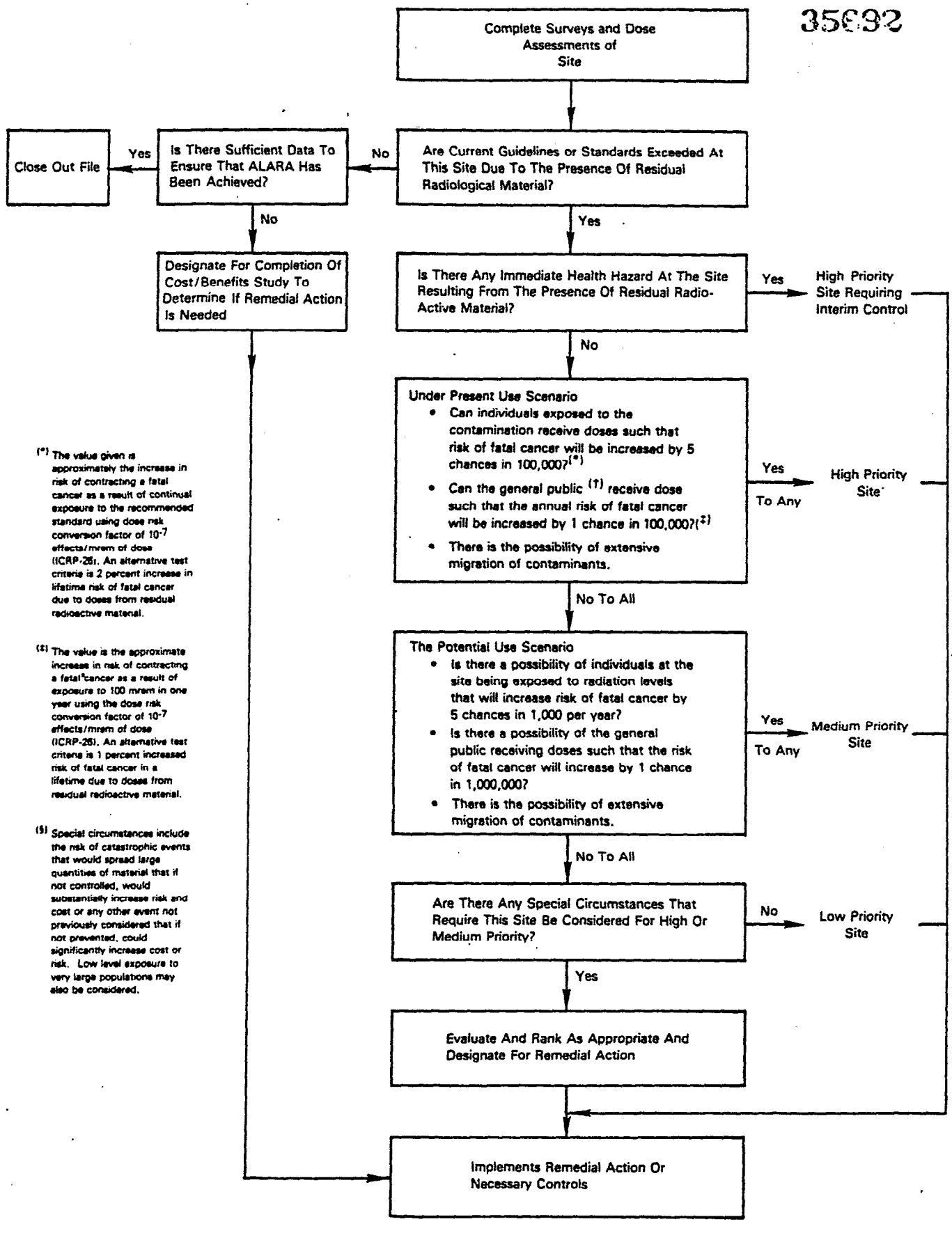


(*) The value given is approximately the increase in risk of contracting a fatal cancer as a result of continual exposure to the recommended standard using dose risk conversion factor of 10^{-7} effects/mrem of dose (ICRP-26). An alternative test criteria is 2 percent increase in lifetime risk of fatal cancer due to doses from residual radioactive material.

(‡) The value is the approximate increase in risk of contracting a fatal cancer as a result of exposure to 100 mrem in one year using the dose risk conversion factor of 10^{-7} effects/mrem of dose (ICRP-26). An alternative test criteria is 1 percent increased risk of fatal cancer in a lifetime due to doses from residual radioactive material.

(§) Special circumstances include the risk of catastrophic events that would spread large quantities of material that if not controlled, would substantially increase risk and cost or any other event not previously considered that if not prevented, could significantly increase cost or risk. Low level exposure to very large populations may also be considered.

Figure C-1. DOE Prioritization Procedure

- there is extensive migration or there is significant potential for extensive migration of the contamination into the surrounding environs.

Medium o Ranking a site as medium priority indicates the site is contaminated above guidelines, and

- there is no immediate hazard to individuals at a site under current use conditions, but there is potential (due to possible change in use or occupancy) for individuals to be exposed to levels of radiation that may increase the risk of cancer above an acceptable level,[a] or

- there is potential for a site to be exposed to levels of radiation that could increase the number of cancers to an unacceptable level[b] if the present use conditions of the site were to change, or

- there is a moderate possibility that contamination may migrate offsite and result in exposure to individuals around the site.

Low o Ranking a site as low priority indicates that the site is contaminated above guidelines; however,

- the exposure level is very close to the level where no discernible increase in cancer risk to individuals under current or near term (10 year period) future use of the site is expected, or

- there is no foreseeable chance of the surrounding population being exposed to levels of radiation that would increase their risk of cancer, or

- there is little or no chance of, or little significance in, migration of contamination from the site.

Dose/Health effects based priorities are only one factor in determining a sites remedial action priority. Other factors (discussed in the text of the protocol) will be assessed by the OR/TSD and DFSD after designation and are used along with health effects priorities to provide the overall remedial action priorities. It is also important to note that the dose/health effects calculations are used in determining priorities but designations are base on comparison of the site to DOE guidelines.

REFERENCES

DOE-85, U.S. Department of Energy Guidelines for Residual Radioactivity at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites, Rev. 1, July 1985.

ICRP-26, Annuals of the ICRP Report, November 26, January 7, 1977.

APPENDIX D. CERTIFICATION DOCKET

The purpose of the Certification Docket is to provide a consolidated and permanent record of DOE activities at the specific site and of this site's radiological condition at the time of certification. This record will be placed in the DOE Public Reading Room in Washington, D.C., and subsequently will be microfilmed for Federal Archives. The certification package will contain a summary of DOE (and predecessor agencies) activities at the site, the supporting documentation, and a bibliography of relevant documents that are not included in the docket. The outline for the final docket is:

(A) Introduction to the Docket

(1) Purpose and Contents of the Docket

(2) Property Identification (general description and drawings of property being certified)

(B) Exhibit I - Summary of Activities at the Specific Site

(1) Site History (MED/AEC use; ownership history and use; and FUSRAP activities at site)

(2) Site Description (past and current)

(3) Radiological History and Status (survey and monitoring information, and criteria for determining need for remedial action)

(4) Selection of Remedial Action (option selected; criteria for the remedial action; cost-benefit analysis; and health effects evaluation)

(5) Summary of Remedial Action (what was done; waste volume and waste types; costs; and occupational and public exposures)

(C) Exhibit II - Documents Supporting the Certification of the Site

These include but are not limited to:

(1) Decontamination or Stabilization Criteria

(2) NEPA Documents

(3) Agreements (with owner, state, and so forth)

(4) Post Remedial Action Survey and Monitoring Data

(5) State, County, and Local Comments On Adequacy of Remedial Action (and others as appropriate)

(6) Recommended Restrictions and Actions Taken to Implement

(7) Federal Register Notice

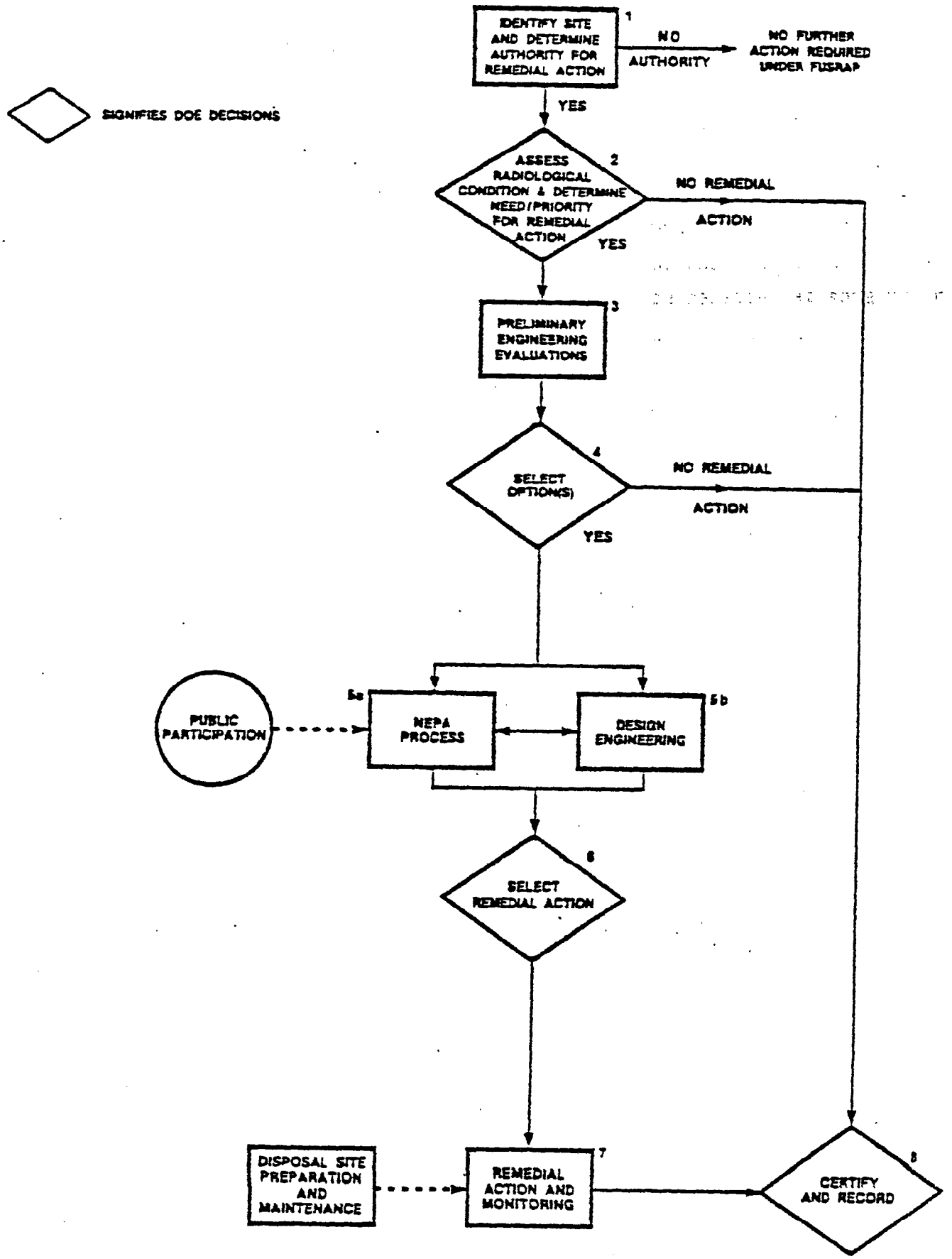
(8) Approved Certification Statement

(D) Exhibit III - Diagrams and/or Figures or Tables Supporting the Certification

(E) List of Relevant Documents

The Certification Docket shall be prepared by OR-TSD for each completed remedial action and will include state, county, and local comments (as appropriate), Federal Register notice, and Approved Certification Statement. The certification statement is signed at DOE Oak Ridge Operations and is approved at Headquarters. OR-TSD drafts and obtains the required concurrences for the Federal Register notice which is issued by Headquarters.

APPENDIX E. BASIC STEPS INVOLVED IN THE REMEDIAL ACTION PROGRAM (FUSRAP ESAPP, APRIL 1985)



APPENDIX F. PUBLIC AVAILABILITY AND ARCHIVING OF FUSRAP RECORDS

Introduction

Documentation on all FUSRAP site investigations and activities (for eliminated as well as certified sites) will be prepared and archived by the Department of Energy as permanent records of the program. This activity is required by this protocol for the purpose of ensuring that investigations completed under FUSRAP do not have to be repeated at some future date. It is DFSD's responsibility to ensure that actions are taken to permanently preserve these records.

Throughout the FUSRAP project DFSD, with its technical assistance contractors and the FUSRAP project office (OR-TSD), will maintain records that document program activities including site identification, characterization, designation or elimination, and site remedial action planning, implementation, and certification. DFSD and the Technical Assistance Contractor will maintain these records documenting site identification, characterization, and designation or elimination activities. DFSD and the FUSRAP Project Office (OR-TSD) will maintain those records documenting remedial action planning, implementation, and certification activities at each site. The certification dockets assembled by OR-TSD as described in Appendix D will be the primary record for those sites designated for remedial action. Elimination reports, including authority reviews and supporting documentation, assembled by the DFSD Technical Support Contractor will be the primary record for sites identified but not included in the remedial action program. In addition, the primary record file will include general information regarding program policy, decisions, and other pertinent information required to reflect as complete as possible history or chronology of activities associated with each FUSRAP site.

Temporary Public Access

The Certification Dockets, major FUSRAP announcements, press releases and, where appropriate, elimination reports will be made available at the Department of Energy Public Reading Room in Washington, D.C. Upon receipt of the primary records assembled by OR-TSD and/or the Technical Assistance Contractor, DFSD will transfer copies of the subject documents to the reading room through a memorandum to the Department's Public Information Office (MA-232.1). The official record copies will be maintained by DFSD or the program office until they are archived. The memorandum will request that MA-232.1 make the copies of the documents available to the public at the reading room for a period from 3 to 5 years, after which time they will be destroyed.

Permanent Archiving of FUSRAP Records

At the termination of FUSRAP, or at an appropriate interval to be determined, DFSD will assemble and prepare these records in accordance with pertinent records management procedures for transfer to the National Archives for permanent retention. The Office of Nuclear Energy Records Liaison Office (NE-73), at the request of DFSD, will coordinate with the Department Records Officer (MA-232.3) to have the records identified for permanent retention by the National Archives. The records will then be available to interested parties through the National Archives.

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FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM

DESIGNATION/ELIMINATION PROTOCOL--
SUPPLEMENT NO. 1 TO THE
FUSRAP SUMMARY PROTOCOL

January 1986

Division of Facility and Site Decontamination Projects
Office of Nuclear Energy
U.S. Department of Energy

35632

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FUSRAP DESIGNATION/ELIMINATION PROTOCOL
SUPPLEMENT TO THE FUSRAP SUMMARY PROTOCOL

INTRODUCTION

This supplement to the Formerly Utilized Sites Remedial Action Program (FUSRAP) Summary Protocol provides additional detail regarding the designation/elimination process, It is intended as an amplification of the information provided in the FUSRAP Summary Protocol and relates to those activities conducted prior to Step 2, Figure II, of that document (the final decision for designation into or elimination from FUSRAP). This supplement is to be used along with the guidance provided in the summary protocol and not in place of it.

The primary objective of the designation/elimination activity is to determine if specific sites are in need of and eligible for remedial action under FUSRAP. Basically, the investigations must provide evidence that a site is contaminated above the current FUSRAP guidelines with radioactive material that resulted from past DOE predecessor activities and that there is authority under the Atomic Energy Act of 1954 as amended (AEA) to conduct remedial action at the site. If these criteria are met, the site is included in FUSRAP. The activities involved in making this determination and the criteria used for the determination are explained in this protocol. A brief discussion of the data collection activities that precede the preparation of the designation or elimination report is also included. The initiation of the designation/elimination activity for a given site is totally dependent on the data collection process.

DESIGNATION/ELIMINATION PROTOCOL

Data Collection

Data to support the designation or elimination activities are derived from several sources. Historical information required to support findings related to the potential for contamination of the site (characterize the radiological condition of the site) and to establish if the Department has authority under the AEA to conduct any necessary remedial actions at a site, is primarily obtained through records searches and also through interviews with cognizant individuals (such as former facility or Atomic Energy Commission employees). In addition, as required and appropriate, new radiological data and/or site specific information are collected through site visits or surveys or contacts with owners.

Records Searches and Interviews. There are essentially two types of records searches that are employed to support the designation/

elimination activity. The first is the systematic review. The Department as part of its site identification and characterization effort has investigated the Manhattan Engineer District (MED) and Atomic Energy Commission (AEC) records stored at various records centers and records storage locations to identify records that are or may be pertinent to FUSRAP. The investigations involve several stages of screening to identify records that require detailed review. As part of the systematic reviews, the pertinent records are examined to determine their subject area, the sites they address, and to obtain copies of material that would support the designation/elimination reviews. The material is reviewed and copied as appropriate for all sites addressed. In addition, notes are taken on the particular records reviewed so that if materials that are not needed for designation/elimination actions are later necessary for other purposes (litigation or Freedom of Information Act responses) their location is easily determined and the required records can be easily retrieved. The systematic approach is the most efficient and cost effective because, the records need only be reviewed once. However, the method does not allow easy or accurate scheduling of results. Because the records are not well categorized and are not generally filed by site [records are in most cases stored by date (FY43 and so forth and by departmental division (Feed Materials Division and so forth)], there is no way of determining when or if enough information will be assembled on any one site until enough material has been collected or all the records have been reviewed.

The second type of search is the site specific review. Under this type of review all the records identified that may contain material on a selected site are screened to attempt to locate those records that probably contain information on that site. These high probability records are then scanned to identify site specific records and only the site specific records are reviewed for designation/elimination information. This search method produces relatively fast site specific results with reasonable probability that all the important facts pertaining to a specific site are identified. Searches completed in this manner can also be scheduled somewhat more precisely than can the results of systematic searches. However, the site specific reviews produce useful information for only one site at a time and result in a more costly and less effective review because the same records groups have to be visited and reviewed several times to extract all the useful data from them.

Though it has the scheduling drawbacks the systematic search is generally the favored approach for the site identification and characterization effort. The site specific searches are only conducted when there are priority requirements to complete investigations on a specific site.

Interviews are generally conducted toward the end of an investigation on a specific site or when it appears that the records will not

be sufficient on their own to support a designation or elimination. As a result, most interviews are site or subject specific; however, at the time of the interview the cognizant individuals are also interrogated for information on other sites or subject for future reference.

Site Visits and Preliminary Surveys. Visits or preliminary surveys are normally only conducted when there is significant probability of residual contamination being present at a site and if there is authority to conduct remedial action at the site if the radiological conditions are found to be unacceptable. The primary purpose of the visits or surveys is to obtain information needed for the site designation or elimination which can not be obtained through the records search activity.

Additional details regarding the implementation of the site visit and survey activities and the records search actions are provided in the Preliminary Analyses Phase section of the general FUSRAP protocol.

Designation/Elimination Analyses

The designation or elimination analyses are completed in two parallel analyses. The site data are reviewed (1) to determine if the sites are contaminated above DOE guidelines or if there is potential contamination on the site due to DOE predecessor operations and (2) to determine if the Department has authority to correct any unacceptable radiological conditions that might be identified at the site. The two analyses are different and require somewhat different supporting data; however, much of the analyses is interdependent and as a result, the reviews are implemented in a manner that requires significant interaction.

A positive determination must be made on both reviews for a site to be included or designated into FUSRAP; the site must be potentially contaminated above guidelines with residual material resulting from DOE predecessor operations and there must be authority for DOE to conduct any required remedial actions. If either of the reviews produce a negative finding (no authority or no potential for contamination) the site is eliminated from consideration for inclusion in FUSRAP. Figure 1 and Figure 2 outline the decision tree for the designation/elimination process. Figure 1 shows the paths and options in a case where the authority is determined first, while Figure 2 represents the case where the potential for contamination (or site characterization) is determined first.

The potential for contamination is determined through the review of the operating history of the site and considers such things as type of operation, length of time the facility operated under AEC contract, quantity of material processed, methods of disposal of wastes, radiological data and so forth. It has been found that sites at which

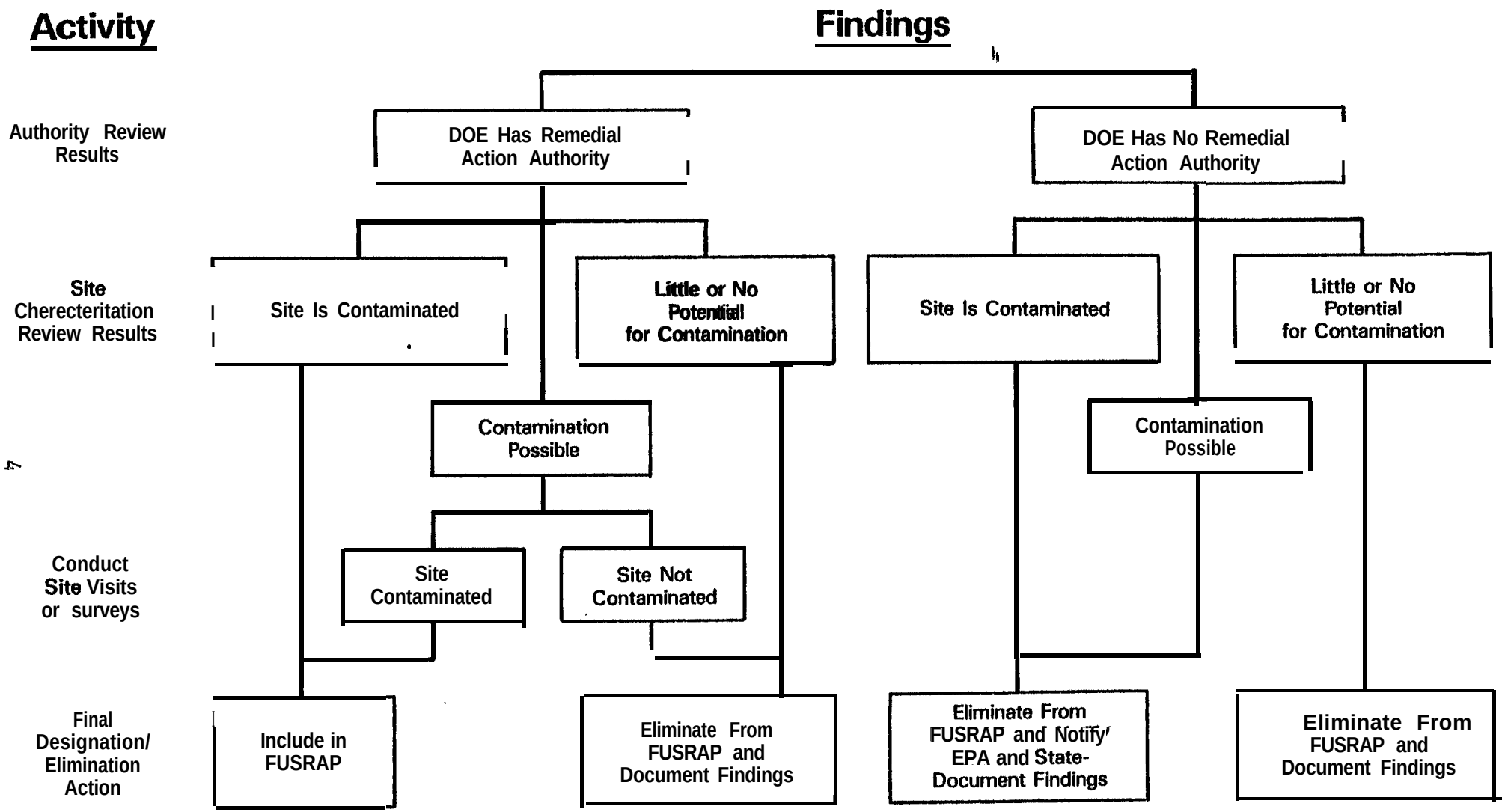

Figure 1. Decision Tree for the Designation/Elimination Process — Alternative 1 — Authority Review Completed First

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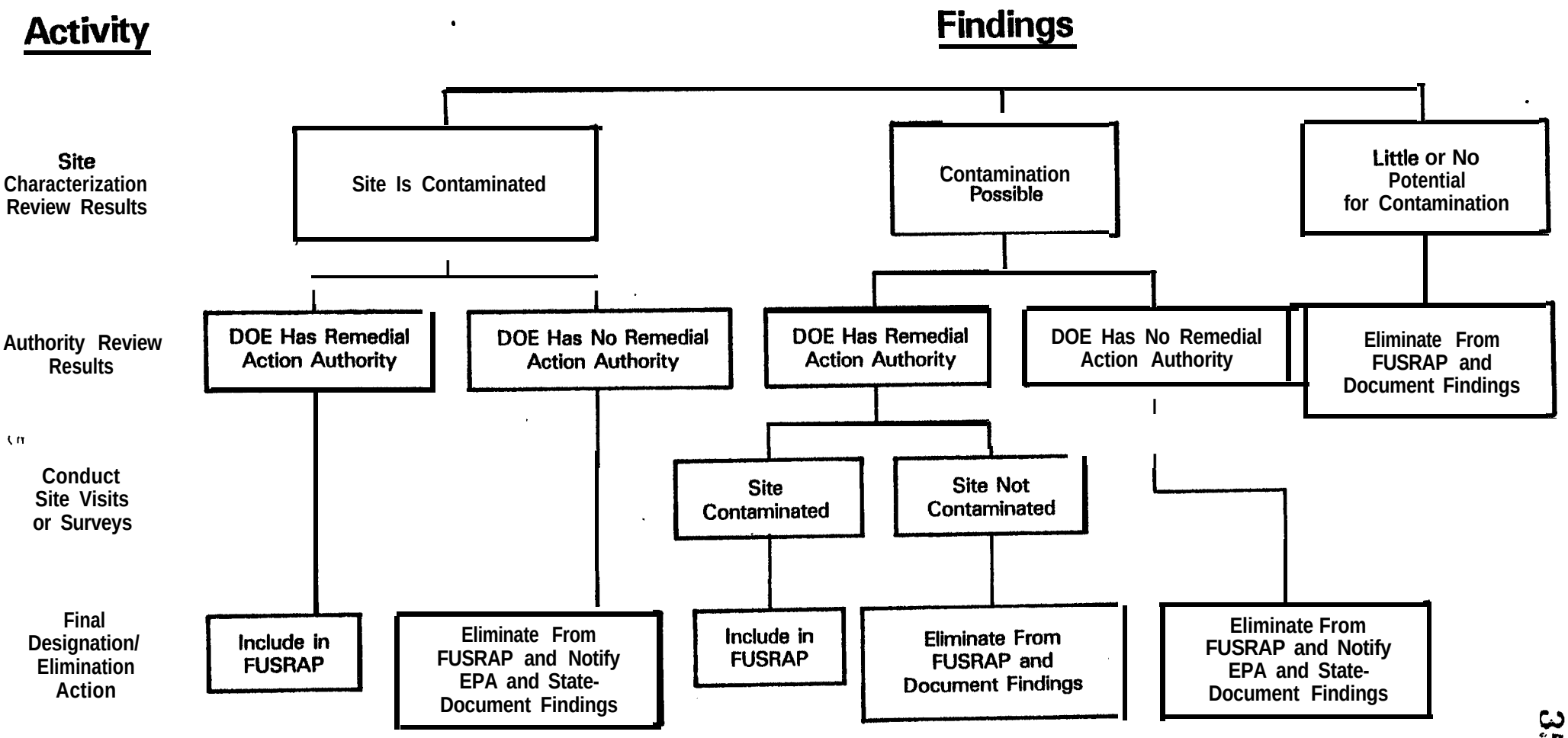

Figure 2. Decision Tree for the Designation/Elimination Process — Alternative 2 — Site Characterization Review Completed First

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little work or only small quantities of material were handled, in general, have fewer records in the files and the larger facilities handling significant amounts of radioactive materials are referenced frequently in the records. Therefore, the frequency of reference in the old records is also used as an indicator of potential for contamination.

· The authority review considers the contractual agreements and final close-out information, the DOE predecessors involvement in the facility and its operation, and health and safety responsibilities. Other important factors considered, include the license status of the site, types and amounts of commercial or other governmental work conducted at the site and current site activities. The types of records or information used in each of the authority and site characterization analyses are outlined in Figure 3 along with some of. the references normally sought during the records searches.

The criteria for determining if DOE will have authority to conduct remedial action at a given site are a series of questions derived by Division of Facility and Site Decommissioning Projects (DFSD) and the Office of General Counsel. The site specific answers to these five generic questions and the supporting reference material are used as the basis to determine if there is DDE authority for remedial action and if the site needs to be considered for FUSRAP. The five questions are listed in Figure 4. The first two questions are generally answered solely on the basis of historical data. The last three questions, however, assume that there is contamination on the site. Therefore, the review of radiological conditions must be completed before the final responses to the authority questions can be developed and the final designation decision made. Initially, if the review or evaluation of radiological condition is not complete, the last three questions are answered tentatively, assuming the site was contaminated with materials associated with past AEC/MED operations. Then a preliminary authority determination is made with the condition that it would have to be shown that the site was contaminated with residues from DOE predecessor operations before a final decision supporting authority can be made. A negative authority finding at the initial stage (prior to a final determination regarding site contamination) will generally result in the site being eliminated from the program. However, if on the basis of this draft authority review the answers to the questions indicate that DOE might have authority for remedial action at the site, additional investigations which may include site visits and/or surveys and contacts with the owner, are implemented as required to provide additional material to support the review. The final authority determination is then made on the basis of the final answers developed using the additional information.

The authority review is an iterative process. Ideally, the authority determination is done with the minimal amount of records review as is possible and practical. As soon as there appears to be

- Site Description
  - Location (address and maps)
  - Facility size
    - Entire site
    - MED/AEC portion
    - Area around the site (population and environs)
- Contractual information (MED/AEC)
  - Size of contract
  - Length of contract
  - Type of contract
  - Products
  - Areas utilized for contractual activities
  - Health and safety provisions
  - Closeout provisions
  - Special provisions
  - Contracting Division or organization
- Contractual information (non-DOE predecessors)
  - Same as above including estimates of fraction of facility and work that was not MED/AEC related
- License information
  - Type of license
  - Length of license
  - Areas and work covered under license
  - Violations
  - Current status
- History of MED/AEC operations
  - Type of operation (materials processed, quantities, waste disposal practices and so forth)
  - DDE predecessor control and involvement at the site
    - Ownership of lands, buildings, or equipment
    - Personnel stationed at the site
    - Frequency of visits to monitor or manage operations
    - Health and safety inspections and so forth
  - Periods of operations and stand-by status
  - Size of staff (production, research, engineering, health and safety and so forth) and portion of time spent on non-MED/AEC operations
  - Final closeout
    - Surveys
    - Property Transfer
    - Status and final releases
- Current status of site
  - Radiological status
  - Current and planned or future uses
  - Proximity of active areas and summary of operations
- Typical References
  - Contracts
  - Processing records
  - Surveys and health and safety reports
  - Correspondence with MED/AEC managers on pertinent issues
  - Closeout records
  - Licenses and inspections
  - interviews

Figure 3. Information Collected and Utilized in the Designation/Elimination Process

Five Questions Used to Evaluate Authority for Remedial Action

1. Was the site/operation owned by a DOE predecessor or did a DOE predecessor have significant control over the operations or site?

2. Was a DOE predecessor agency responsible for maintaining or ensuring the health, safety, and environment of the site (i.e., were they responsible for cleanup)?

3. Is *the* waste, residual, or radioactive material on the site the result of DOE predecessor related operations?

4. Is the site in need of further cleanup and was the site left in unacceptable condition as a result of DOE predecessor related activities?

5. Did the present owner accept responsibility for the site with knowledge of its contaminated condition and that additional remedial measures are necessary before the site is acceptable for unrestricted use by the general public?

Figure 4. Factors Considered in Authority Reviews

sufficient data to answer the five questions (at least tentatively) and to make a determination, a draft authority *review* package is prepared and submitted to the Office of General Counsel (GC). The authority review package contains:

1. A summary of the site's operation,

2. Available information on the current condition of the site,

3. Specific answers to the questions in Figure 4; and

4. Copies of pertinent documents supporting the answers.

If GC recommends that there is insufficient data to 'make a' determination, efforts are made to identify and collect the required materials. However, if the searches prove unsuccessful and it is unlikely that any additional useful information will be derived from future records searches the authority review and determination are completed on the basis of the available information. In general, insufficient data will result in a no authority determination.

If GC recommends that the data provided is sufficient to make an authority determination, then the authority finding is made, the authority review is finalized and the next step in the process is implemented. The next step depends on the status of the site radiological evaluation effort. If the potential for contamination has been established through historical data or survey data then the elimination or designation package is prepared. If it has not, then additional investigations are conducted.

If the finding is for no authority and there is, or is potential for, contamination at the site, an elimination report is issued. The site owner, appropriate state agencies, EPA, and other appropriate Federal agencies are notified that there is (or is potential for) contamination at the site and that DOE has no authority under the AEA to conduct any remedial actions at the particular site if they are found necessary. The elimination report is made available to the owner, state agencies, EPA, and the other appropriate Federal agencies. The report is placed in the DOE Public Reading Room for at least a Z-year period and is permanently archived by DOE in accordance with procedures described in Appendix F of the FUSRAP Summay Protocol.

If the finding is for authority, the radiological and operating data are summarized to determine if additional radiological characterizations are needed to determine if the site should be considered for remedial action. If additional data are needed the site survey is planned and implemented and a designation package *(or* elimination package as appropriate) is prepared after the survey is completed. If adequate information is already available, then the designation or elimination package is prepared. The owner and the

appropriate state agencies are notified of the designation of the site for remedial action.

In those situations where the potential for contamination is low or non-existent, the sites are eliminated from the program irrespective of the DOE authority. If the authority issue has not been resolved at the time that the determination of no potential for remedial action is made, then the authority review is terminated.

Designation/Elimination Reports. Designation/elimination reports are prepared to document the analysis and to summarize the data available on a specific site. The draft designation report and supporting material is used as the basis for the designation determination. In order for a site to be included in FUSRAP the report must indicate that:

- The site is potentially contaminated (above FUSRAP criteria) with radioactive residues that resulted from DOE predecessor operations, and
- DOE has authority to conduct remedial action at the site.

The site will not be included in FUSRAP if it is already included under some other remedial action program or is under NRC or state license.

The contents of the designation reports vary slightly from site to site and may include the following types of materials:

1. A summary which discusses the past operations at the site, the current status of the site, disposal practices, radiological history and so forth.
2. A description of the current status-of the site and its location and size.
3. A summary of the authority review completed on the site.
4. An analysis of potential doses that might be received by members of the general public as a result of exposure to contamination on the site (using available radiological data).
5. A comparison of the levels of residual radioactive material on the site and potential doses to guidelines and standards.
6. A preliminary ranking of the site on the basis of potential health effects using the DOE/FUSRAP prioritization procedure (only for those sites that are designated), and
7. References and supporting data.

Elimination reports may also contain similar information, however, depending on circumstances will generally be much briefer. The elimination may be based on a finding from historical records of little potential for contamination or that the site is covered under another remedial action program and so forth. In cases where the authority review is completed first and the finding is that DOE has no authority, the authority review may be used in place of the elimination report.

Activities Following Designation/Elimination

Designated Sites. Once a determination is made that a site qualifies for designation under FUSRAP, the DOE Oak Ridge Operations Office Manager and the Technical Services Division (OR-TSD) Director are notified by the Director of the Office of Remedial Action and Waste Technology (the superior office for DFSD) that remedial action is authorized under FUSRAP. OR-TSD (the FUSRAP project office) is then responsible for taking appropriate steps to complete any necessary characterization of the site and remedial actions determined to be required. The remedial action process is outlined in more detail in the FUSRAP Summary Protocol. Following completion of the remedial action the site is certified in accordance with procedures also outlined in the FUSRAP Summary Protocol and Supplement No. 2 to the FUSRAP Summary Protocol (verification/certification) November 1985.

Eliminated Sites. Sites eliminated from consideration for FUSRAP are in two general categories:

1. Sites that have little or no potential for being contaminated with radioactive residues for which DOE either does or does not have authority for remedial action.

2. Sites for which DOE has no authority for remedial action that are or are potentially contaminated with radioactive residues or material.

For a site in the first category, the elimination report is issued and filed and the information on the site is updated in the FUSRAP sites data base. At the end of each'year a summary report documenting the status of all the sites reviewed during the past year is prepared. This report along with the supporting elimination information are eventually archived to ensure that a record of the investigations will be permanently available.

Similar reports are prepared for the sites in the second category, and the information is documented in a similar manner. However, in order to ensure the attention of appropriate government agencies to conditions that may impact negatively on the general public or the environment, DOE notifies EPA and other appropriate Federal and/or state agencies of the findings and potential hazards associated with

the site. DOE is available to assist these agencies in the state in interpreting results or in assessing data on the sites; however, unless DOE is provided authority for the site through another mechanism (such as a legislative mandate) all activities excepting assistance to other agencies are terminated.

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FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM

VERIFICATION AND CERTIFICATION PROTOCOL--
SUPPLEMENT NO. 2 TO THE
FUSRAP SUMMARY PROTOCOL

NOVEMBER 1985
REVISION 1

Division of Facility and
Site Decommissioning Projects

Office of Nuclear Energy

U.S. DEPARTMENT OF ENERGY

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INTRODUCTION

This supplement to the general Formerly Utilized Sites Remedial Action Program (FUSRAP) protocol outlines the procedures for the verification of remedial action and the ultimate certification of a FUSRAP site's radiological condition. This supplement is intended as an amplification of the description of the certification process presented in the "Certification of Site Conditions Phase" section of the FUSRAP Summary Protocol. The certification process includes the collection of data necessary to confirm the compliance of the remedial action with applicable radiological guidelines and the preparation of materials required to permanently document the radiological condition of the site following completion of remedial action activities.

The verification and certification activities involve several elements including (1) post-remedial action measurement, (2) independent verification (independent verification of results and/or procedures by the Department of Energy (DOE) and/or others as appropriate), (3) interaction with and/or notification of concerned parties, and (4) final project documentation.

The various activities and subelements of the certification process are managed and implemented by the FUSRAP project office at the DOE Oak Ridge Operations (Technical Services Division, OR-TSD) and their contractors. The discussion to follow outlines the activities within the certification process and discusses responsibilities.

CERTIFICATION

The discussion of the certification process is divided into three general types of activities in the discussions to follow:

- Remedial Action Measurements
- Independent Verification (by DOE and others)
- Certification Docket (Federal Register Notice and Owner Notification)

  Preparation,
  Review, and
  Distribution

The first activity is the final step in the remedial action phase of FUSRAP and is the major source of data supporting the certification effort. The other two elements makeup the portion of FUSRAP known as

the certification phase. Figure 1 is a conceptual diagram of the process and its relationship to the remedial action phase. Figure 2 is a conceptual time line showing the relative time relationships of these activities. Figure 3 is a conceptual flow chart of the certification process.

The process outlined in this supplement begins with activities conducted durin the remedial action Phase by the Project Management Contractor (PMC4. These activities involve excavation/decontamination control measurements, supportive sampling and analyses, and preparation of the post-remedial action report. They are implemented by the PMC and managed and overviewed by OR-TSD.

The independent verification activities, for the most part, run parallel with remedial action and post-remedial action activities, The reviews, surveys, measurements and documentation prepared during this element of the certification process are prepared by an independent DOE contractor not involved in the remedial action activity. Additional information may be received from state or other Federal agencies. As with the DOE independent verification activity, the State and other Federal agency activities may involve independent review of the remedial action contractors reports as well as independent measurements.

The draft certification docket is compiled by the OR/TSD (FUSRAP Project Office) and includes a summary of the action, documentation supporting the compliance with criteria, a copy of the interim letter to the property owner, the draft certification statement, and the draft Federal Register Notice. The complete draft docket is sent to Division of Facility and Site Decommissioning (DFSD) for review and comment. The final certification statement is approved by the field office and the final Federal Register notice is signed and issued by DOE Headquarters (Figure 4).

FUSRAP remedial actions involve activities to clean-up or stabilize radioactively contaminated land and structures. While the remedial actions are conducted.in a manner that would insure that no user of the site would receive doses in excess of those allowable (reference FUSRAP and Remote SFMP Criteria and Guidelines), the criteria for clean-up of structures differ from those used for the clean-up of land. Criteria used in the decontamination of structures are primarily surface contamination guidelines and external gamma limits.* Maximum permissible concentrations of radionuclides in the air and radon/radon daughter limits are also used.* For open areas or land, allowable soil concentration guidelines are used as remedial action criteria.* As a result of the differences in the types of

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*The U.S. Department of Energy Guidelines for residual radioactivity at Formerly Utilized Sites Remedial Action Program and remote Surplus Facilities Management Program Sites (Rev. 1, July 1985).

FIGURE 1. Conceptual Diagram of the **Certification** and Its **Relationship** to the Remedial **Action** Phase

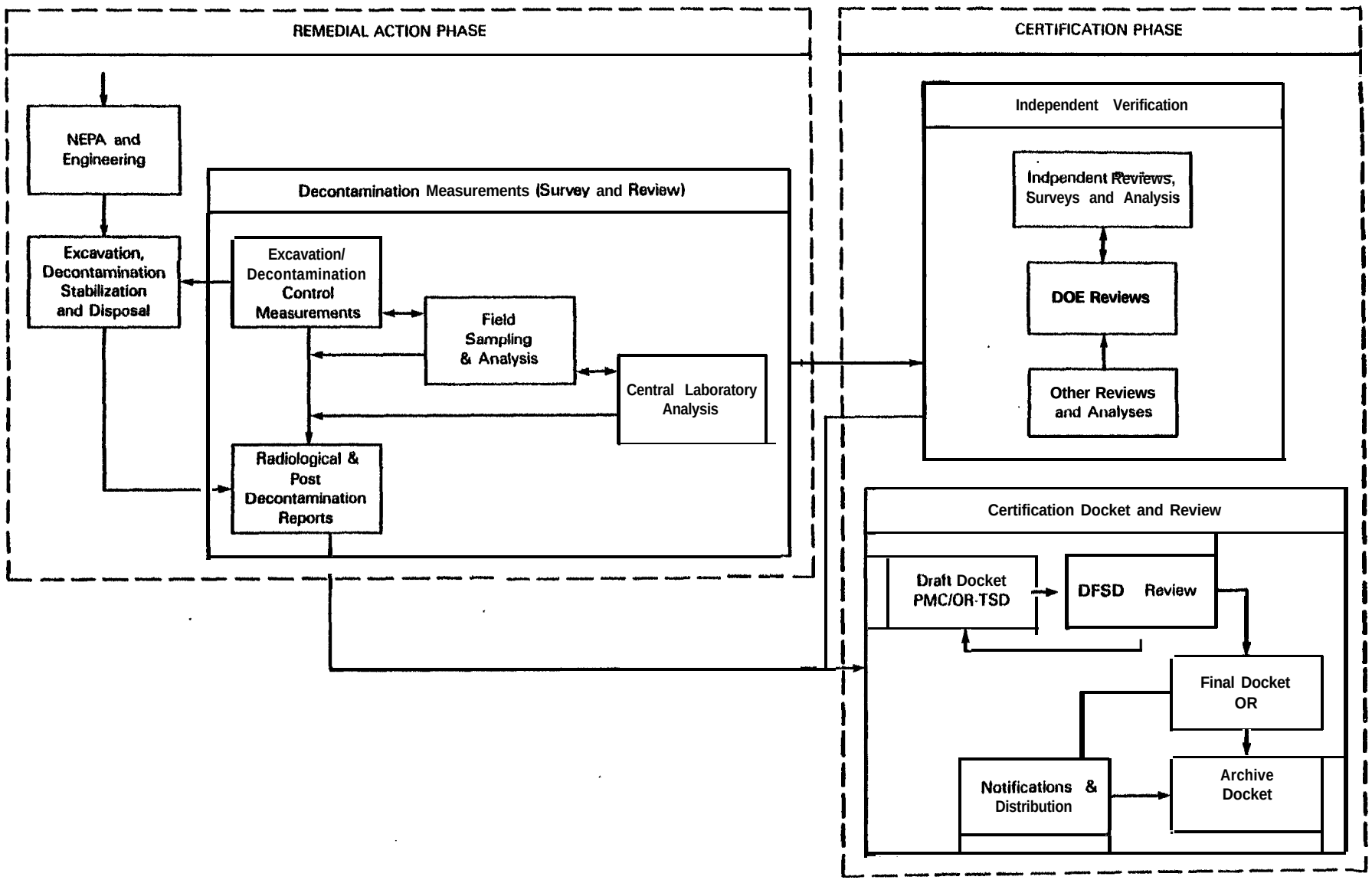
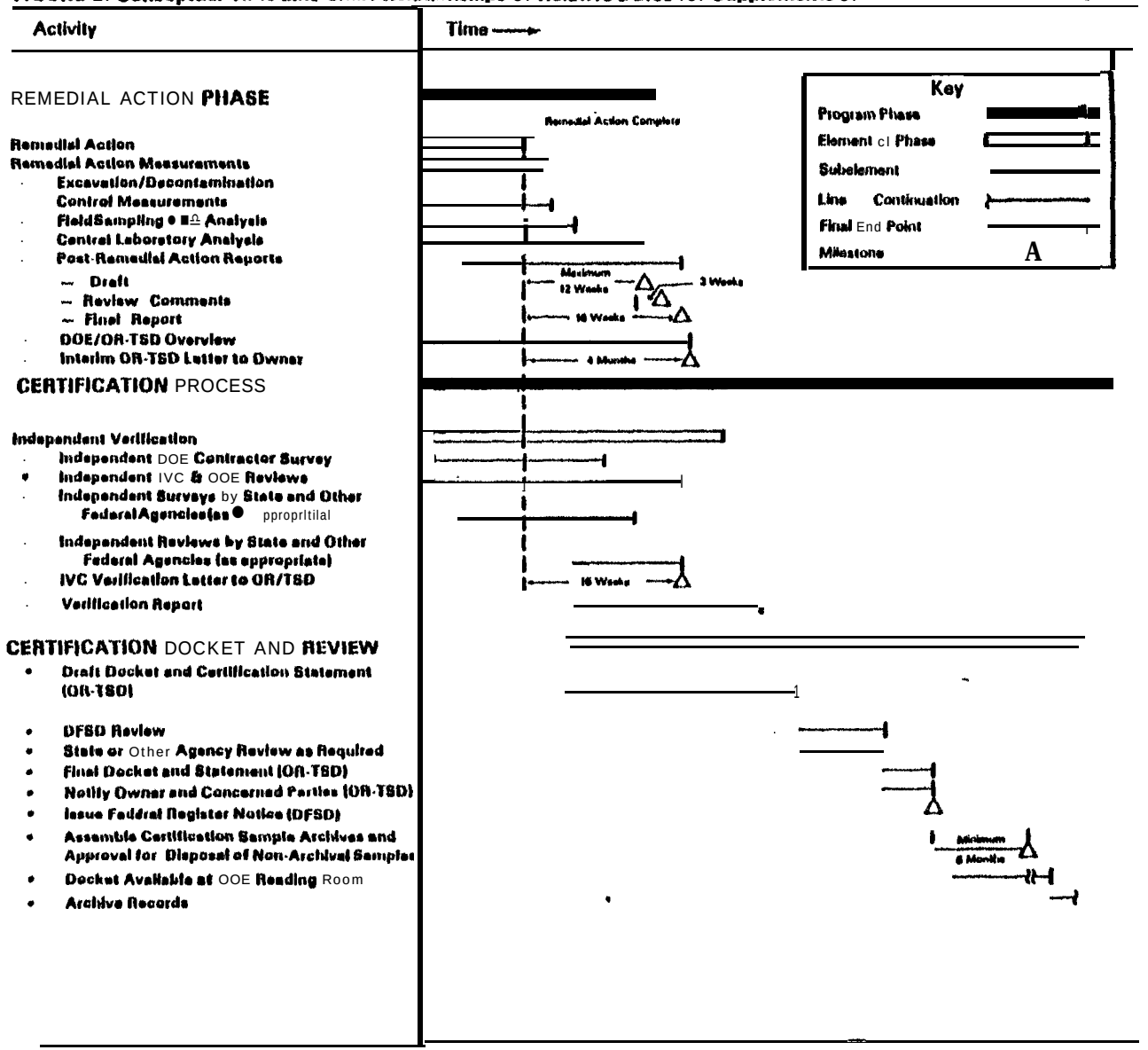


FIGURE 2. Conceptual Time Line Chart-Relationships of Relative Dates for Supplements of the Certification Process*



*Actual Time Units Are Remedial Action Specific This Chart Is Presented to Identify Relative Initiation and Completion Times of Various Subelements of the Certification Process.

05032

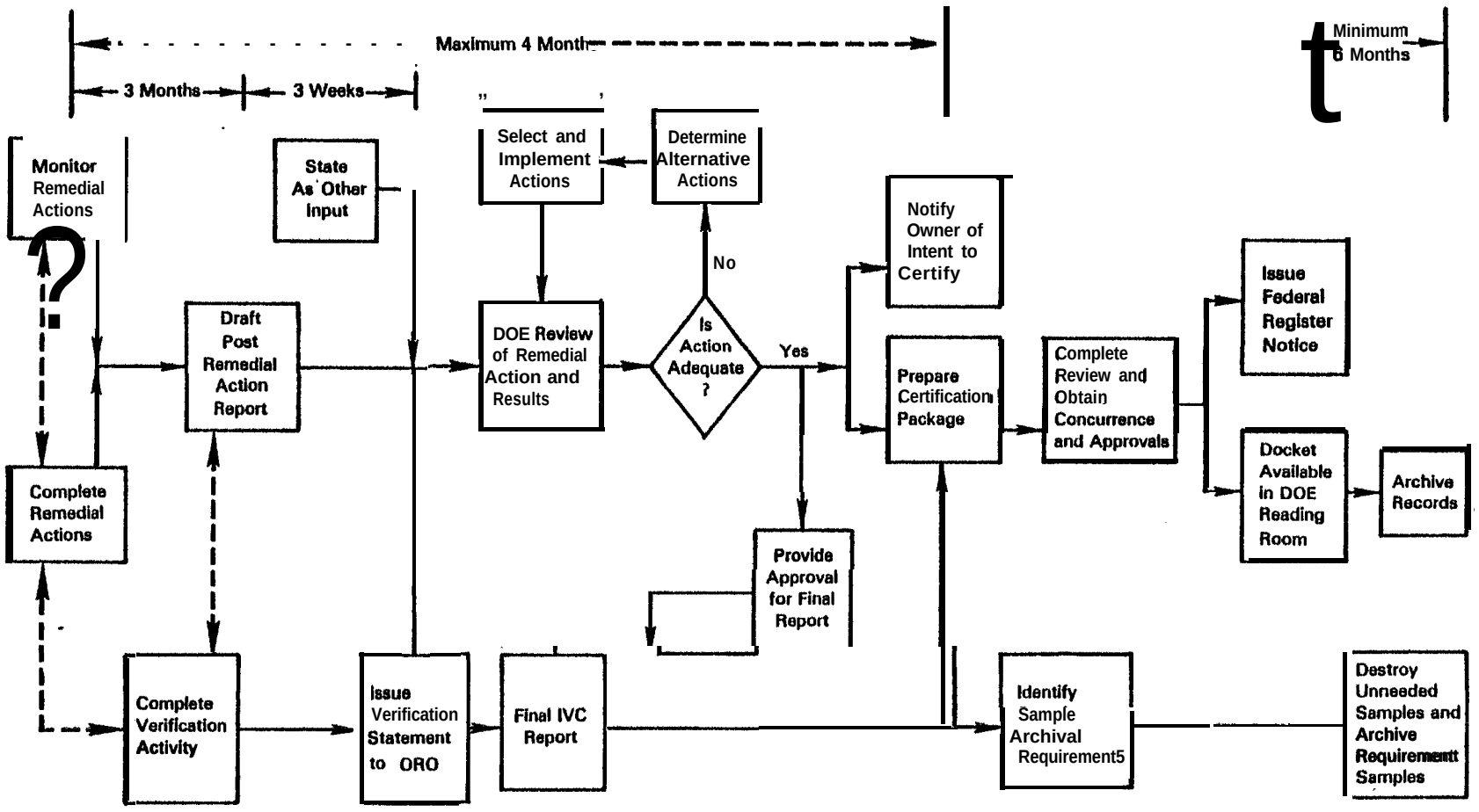


Figure 3. GENERAL FLOW CHART FOR CERTIFICATION PROCESS

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Sequence for Final Docket Sign-Off and Assembly

1. The Field obtains approval of the certification statement and the Federal Register Notice from the Field Office Chief Counsel. The Federal Register Notice submitted for approval should include a copy of the concurrence chain.

2. The Field obtains approval of the draft Federal Register Notice from DOE Headquarters MA-213.13.

3. The following is transmitted to NE-23 for final approval after Chief Counsel concurrence of the certification statement and MA concurrence of the Federal Register Notice:

   a. Memorandum for signature (to NE-20 from NE-23 recommending certification).

   b. Federal Register Notice for signature by NE-20.

   c. Bound certification docket.

   d. Published documents referenced in Exhibit 11 of the bound docket. (Items a, b, and c include DOE F 1325.10, Official File Copy, to indicate appropriate concurrence.)

   A copy of the memorandum (a) will be included in the final docket as is the signed certification statement and signed Federal Register Notice.

4. NE-20 signs the Federal Register Notice.

   a. Copies of the signed Federal Register Notice are transmitted to the field for inclusion in the final docket.

   b. The original plus two copies or two signed duplicate originals of the Federal Register Notice and DOE F 1325.10 (Official File Copy) are sent to MA-213.13 by DFSD for publication.

5. The Field inserts copies of the signed memorandum, the Certification Statement and the Federal Register Notice into the bound docket and makes distribution, as appropriate, to the local public document room, state, etc. {Five copies of the bound docket, along with the referenced published documents, are sent to DFSD for entry into DOE public document room at Washington, D.C., and headquarters distribution.)

6. The Field Office will be responsible for notifying the State and local governments, as necessary, and property owners of the certification action. The state will be requested to insert a notice in land record offices if appropriate.

Figure 4. The Certification Procedure/Chronological Outline

criteria and guidelines applied to building and land decontamination, the requirements for verification sampling and analyses vary somewhat for buildings and land. These differing requirements are discussed in this protocol.

Remedial Action Measurements. Excavation/decontamination control measurements (using portable gamma-, beta-gamma-, and, where necessary, alpha-measuring instruments) will be used by field personnel to guide the remedial action and to make the preliminary determination as to the extent of the excavation and/or decontamination required. For cases of soil contamination, upon. completion Of each planned segment of a remedial action (as determined by the excavation control measurements and prior to backfilling activities) the on-site radiological contractor will take representative soil samples and analyze them at the field laboratory. If these analyses confirm that the remedial action criteria have been achieved, the backfilling can proceed. If the samples indicate that additional material must be- removed, the remedial action contractor will be informed of the requirements and take appropriate action.

A representative number of the remedial action soil samples will also be sent to a central laboratory for final sample confirmation. The results of these analyses will be compared with the field data to ensure compliance with the remedial action criteria.

Compliance with criteria in structure or building decontaminations will be demonstrated by field measurements except in those cases where air sampling is required. Surface contamination and gamma measurements will be taken to ensure compliance with the FUSRAP criteria and guidelines or standards referenced in that criteria document. As appropriate, representative samples will be taken from the air, water, and residue samples that were' analyzed in the field and used to support the confirmation of the site's condition. Again, as appropriate, samples will be sent to a central laboratory for confirmatory analyses.

These activities will also include the review of radiological data after the completion of the remedial action by DOE/OR. The results of the radiological support contractors surveys and confirmatory analyses will be documented and included as part of the PMC's post-remedial action report. A draft of the report will be issued for DOE and Independent Verification Contractor (IVC) review within 3 months of the completion of the remedial action. The final report will be issued about 1 month later presuming DOE and IVC comments are provided within a 3-week period after receipt of the draft report.

Independent Verification by DOE. DFSD will provide an independent overview evaluation of the remedial action through review of the reports prepared by the PMC. Independent measurements, sampling and analyses and review of procedures and remedial action survey results will be completed by a DOE contractor (IVC) not directly associated

with the remedial action. The IVC will have primary responsibility for the scope of its field investigation. The IVC will prepare a generic plan outlining the procedures to be used during verification activities. The plan will be submitted to the Field Office and DFSD for approval. The IVC will then provide DSFD and the Field Office with only a brief outline of site specific plans for sites selected for verifications. The outline will reference the generic plan and note special concerns. The generic plan will describe the types of verification actions that may be taken and the reasons for applying certain procedures to specific types of sites. The IVC may conduct two types of verification reviews (types A and 3) at a site or group of properties. Type A verifications will include review of the remedial action and radiological contractors data and possibly the analyses of some split samples. Type B verifications will include an on-site visit and survey involving direct measurements and sampling and/or split sample analyses. The primary purpose of both of these actions is to confirm the adequacy of the procedures and methods used by the remedial action and radiological contractors. In the field, the IVC may increase or decrease the scope of the independent verification survey on the basis of field data. Appendix I outlines the procedures to be used by the IVC for independent verification of remedial action and procedures to correct for any discrepancies found during the verification process. The OR/TSD will be responsible, through their management function, to assure that the verification activities are consistent with this protocol.

Independent verification will be accomplished on all FUSRAP remedial action sites. The level of verification required will be decided by DOE with input from the IVC. Off-site or vicinity property remedial actions may be verified in groups where so recommended by the IVC and approved by the DOE. These independent evaluations will further verify that the remedial action was accomplished in accordance with standards and criteria appropriate for the project. Within 4 months after the completion of a remedial action, the verification contractor will issue a verification statement and provide copies to DFSD and OR/TSD. In the case were vicinity properties were grouped and verifications were only completed on selected properties, the verification statement shall be written to cover all the properties in the group on the basis of the results of the selected properties. Upon receipt of this verification statement, OR/TSD will send an interim letter (notification of intent to certify) to each of the site or property owners.

The results of the verification survey will be presented in a final report, and like the post-remedial action report, will be reviewed by DOE-Headquarters, OR/TSD, and, as appropriate, the state and other Federal agencies. In addition to the final reports, representative samples from the remedial action survey and the verification survey will be properly labeled, retained and archived for an appropriate period (see Appendix II). The samples shall not be discarded until such time as the final certification package for the specific site is

completed, undergoes review, and is archived following an appropriate period of availability at the DOE public document reading room (see Appendix II). Throughout the planning, implementation, reporting and archival activities associated with this process, the IVC and PMC will work closely to optimize overall performance. The IVC and PMC will make every effort to resolve scheduling conflicts and expedite information exchange and on-site activities. Procedures to handle minor discrepancies in the field shall be developed and agreed upon by the IVC, PMC, and remedial action contractor. DOE (DFSD and OR/TSD) should be notified of any problem that cannot be handled by the contractors as far in advance of the verification statement as possible, and will take expeditious actions to insure that the remedial action and verification are adequately implemented.

If it is determined by the IVC that the remedial action was not successfully completed or that the radiological data and supporting information or procedures are not adequate to allow certification of the site, such findings will be reported to DOE immediately. OR/TSD, with assistance from DFSD, will review the problems and take appropriate steps to have deficiencies corrected or resolve the IVC defined problems. The 4 month maximum time period from completion of remedial action to notification of the owner by OR/TSD will not be in effect in cases where adequacy of certification data is in question. The time limitation will again be in effect once the issue is resolved.

Independent Verification By Others. Upon request made to DOE in advance of the initiation of remedial actions, qualified Federal, state, and local agencies will be given the opportunity to perform independent measurements and analyses or to analyze split samples taken during DOE radiological surveys. Each agency will also be given the opportunity to review the radiological support contractors measurement, sample collection and preparation and analytical procedures and the resulting data. Local groups desiring to implement such actions will have to do so through their state or local governments.

Certification Docket Preparation and.Review. Following completion Of the post-remedial action report and the verification statement, DOE/OR-TSD will be responsible for (1) providing the owner, within 4 months after completion of a remedial action activity, an interim notification of DOE's intent to certify the remedial action; and (2) the draft certification docket for the specific site (outlined in the Certification of Site Conditions Phase section of the FUSRAP Summary Protocol). The final docket (see Figure 5) and certification statement will be issued after completion of the docket review cycle also discussed in the FUSRAP protocol. The draft certification docket shall be prepared by OR-TSD for each completed remedial action. The dockets may be prepared by phase (if the remedial action is conducted in phases) and may include groups of vicinity properties as appropriate. A docket is to include the items discussed above and

Certification Docket

(A) Introduction to the Docket

(1) Purpose and Contents of the Docket
(2) Property Identification (general description and drawings of property being certified)

(B) Exhibit I - Summary of Activities at the Specific Site

(1) Site History (MED/AEC use; ownership history and use; and FUSRAP activities at site)
(2) Site Description (past and current)
(3) Radiological History and Status (survey and monitoring information, and criteria for determining need for remedial action)
(4) Selection of Remedial Action (option selected; criteria for the remedial action; cost-benefit analysis; and health effects evaluation, where appropriate)
(5) Summary of Remedial Action (what was done; how it was done; waste volume and waste types; disposal location; cost breakdown; and occupational and public exposures)

(C) Exhibit II - Documents Supporting the Certification of the Site

These include but are not limited to:

(1) Decontamination or Stabilization Criteria
(2) Designation or Authorization Documentation
(3) Characterization Report
(4) NEPA Documents
(5) Agreements {with owner, state, and so forth)
(6) Post Remedial Action Survey and Monitoring Report
(7) Verification report and interim verification letter to the owner.
(8) State, County, and Local Comments On Remedial Action {and others as appropriate)
(9) Recommended Restrictions and Actions Taken to Implement Them
(10) Federal Register Notice
(11) Approved Certification Statement

(D) Exhibit III - Diagrams and/or Figures or Tables Supporting the Certification

(E) Relevant Documents

Figure 5. Certification Docket Contents and Outline

listed in Figure 5. The final Federal Register notice, and approved Certification Statement will be included in the docket and any required changes will be made to the text summaries at the time of DOE DFSD review of the draft docket.

The certification statement will be forwarded by the field office (OR/TSD) to the property owner and the state in which the property is located. A notice will also be published by DOE headquarters in the Federal Register. The state or local government as appropriate will be requested to have the land records annotated to indicate completion of the remedial action and to establish a public record of the certification that the remedial action criteria, guidelines or standards have been achieved or that restrictions are required for continued use of the site. Upon publication of the Federal Register notice, the certification docket containing a complete historical record of the remedial action, including the certification statement and the final project reports, will be placed in the DOE public document room at Washington., D.C., and the Field Office locations for a suitable period of time before it is permanently archived.

APPENDIX I

Procedure for Independent Verification of Remedial Action and Correction of Discrepancies at FUSRAP and Vicinity Properties

INTRODUCTION

Independent verifications will be carried out for FUSRAP sites and vicinity properties in order to provide additional assurance for certification that the authorized limits for the remedial action have been achieved. The FUSRAP remedial action activities are managed by the DOE Oak Ridge Operations Technical Services -Division (OR/TSD). Onsite verification surveys will be carried out for some vicinity properties and for all sites. Heavily contaminated vicinity properties, or properties where independent surveys are requested by the owner, local or State officials, will have onsite verification surveys. The procedure for conducting. and reporting the independent verification is described below.

INDEPENDENT VERIFICATION PROCEDURE

The Independent Verification Contractor (IVC) will perform all or some of the following verification activities: (1) review the site characterization survey, the remedial action plan, available progress reports, and data for the remedial action and restoration of each property or site; (2) schedule a visit to the selected property or site immediately following remedial action, without significantly delaying or interrupting the restoration efforts or some time after the restoration of the site; (3) perform gamma scans for selected locations where excavation has occurred; (4) perform discrete gamma measurements at specific grid locations for comparison with remedial action authorized limits; (5) perform beta-gamma and alpha measurements as required to verify decontamination of structures and/or equipment; (6) perform independent soil sampling and analysis of excavated areas for comparison with remedial action authorized . limits; (7) perform independent environmental sampling and analysis as required to confirm that radionuclides in air and water are within required limits for the specific remedial action; (8) perform independent analyses of soil samples selected from the contractor's archives for the vicinity properties at which independent gamma surveys or soil sampling were not performed; the selection of the archive samples for independent analysis will be based on statistical . guidelines as determined by the IVC; (9) prepare a verification letter.

The number of these activities'and the detail to which they are conducted will depend on the type of verification activity being implemented. Type A verifications in general will include the review of the radiological and remedial action contractor results and, in some cases, an analysis of split samples. Where necessary to confirm results after the restoration, a visit to the site may be warranted. Type B verification will be more thorough and may include all of the nine activities depending on the site conditions and magnitude of the action. The verification letter and report are prepared for both type of surveys.

Review of Remedial Action

All site designation and characterization reports, remedial action plans, progress reports, and survey data pertaining to the specific site of interest will be made available to the IV& for review. These reviews will be conducted as part of Type A verifications and to plan the Type B verification surveys and to determine whether the remedial action plans were changed during the course of remedial action in a manner which would affect the site conditions or the conduct of the verification survey. Post-remedial action data will also be provided to and reviewed by the IVC for both Type A and Type B surveys. The post-remedial action data will be provided to the IVC in a timely manner such that review of the information can be completed and the verification letter sent within 3 months of the completion of remedial action.

Site Visits

A visit will be scheduled to a selected vicinity property or site undergoing remedial action prior to restoration or immediately following the remedial action. Every effort will be made to establish an open communication by both the IVC and the remedial action contractor to avoid interruption or delay of the construction schedule. The IVC will notify OR and/or the PMC of those vicinity properties and site areas which will be sampled or surveyed for verification prior to closure. OR or PMC, as appropriate, will notify the IVC at least 72-hours prior to closure of these selected sites. The notice may be given on the basis of a group of properties, not necessarily for each vicinity property. The IVC is responsible to accomplish any verification survey and sampling without interference with the construction schedule providing at least a 72-hour advance notice is given.

Gamma Scanning and Discrete Measurements

A gamma scan and possibly a set of discrete measurements will be performed on either excavated vicinity properties or site areas. The survey will be performed to the site characterization and remedial action survey grids and will be performed in accordance with

ORNL/TM-8600*, its equivalent, or other guidance provided through the field office and approved by DFSD. The exposure rates will be recorded on a map of the property or site area for comparison with the data taken by the remedial action contractor. This map will be compared with the authorized limits.

Other Direct Measurements

Beta-gamma and alpha measurements performed, as required, in areas, structures, and/or equipment affected by the remedial action or decontamination, will be tied to previous remedial action related surveys. These measurements and scans will be performed in accordance with procedures in ORNL/TM-8600 or its equivalent. The results will be recorded on maps, drawings, or tables of the structures, equipment, or areas and compared to authorized limits.

Soil Samples

About five verification soil samples will be taken from a selected 'excavated vicinity property or site area on a systematic pattern. This number may change according to the size of the vicinity property or site area and the contamination pattern. The soil samples will be obtained from the surface (15 cm depth) or subsurface (15 cm depth) of the decontaminated area. These soil samples will be analyzed by the IVC for the radionuclides specified in the remedial action plan and will be compared with the authorized limits. If no soil sample is taken from a property or area by the IVC, an independent analysis will be performed by the IVC using selected soil samples taken from the remedial action contractor's archive. The samples will be selected and analyzed in accordance with the procedures in ORNL/TM-8600, its equivalent, or other guidance provided through the field office and approved by DFSD.

Air and Water Samples

Representative verifications samples of air or water will be collected and analyzed when determined necessary through reviews of the site data. Sufficient samples will be collected at discrete locations by the IVC to confirm the remedial action contractors results and verify compliance with the appropriate criteria. The samples will be collected and analyzed in accordance with procedures in ORNL/TM-8600, its equivalent, or other guidance provided by the field office and approved by DFSD.

*ORNL/TM-8600, "Procedure Manual for the ORNL Remedial Action Survey and Certification Activities (RASCA) Program"

Comparison of Results

Procedures for comparison of IVC results to those of the radiological contractor will be discussed in the IVC's generic plan. In general, comparison of split samples will be done on a sample to sample basis. The IVC and remedial action contractors results should agree within the expected statistical deviations of the analysis methods used. IVC survey results (direct measurements, sampling and analysis) should be compared to the remedial action contractor results on the basis of the criteria, taking into account averaging requirements as well as sampling and analysis considerations.

Corrective Action for Discrepancies

If the IVC verification survey or sample analyses show that any result is above authorized limits for the remedial action (a discrepancy), a corrective action to resolve this discrepancy must be taken by OR. The IVC will notify DFSD (NE-23) and OR of the discrepancy as soon as possible. OR will determine and instruct the remedial action contractor whether additional cleanup action will be taken or an exception will be requested as specified in the FUSRAP/Remote SFMP Guidelines. The IVC will re-verify the property or site area after corrective action. The corrective action and any exception will be recorded in a corrective action section of the final report or closeout report prepared by the remedial action contractor.

Verification for Post-Remedial Action Report

After the completion of the post-remedial action, verification survey or review, radiological survey and laboratory analyses of soil samples, a verification letter and report will be prepared by the IVC for each vicinity property or site. The authorized limits and the background levels of radiation will be compared to the verification results. The verification letter will address the comparative results of the verification activities and include a statement of verification. The verification report will include the field and laboratory analyses results and any anomalies that were noted during independent verification survey and any reverification survey. Appropriate tables and a listing of results will be included as well as illustrations of the areas surveyed; i.e., soil sample locations and identifications, gamma levels, etc. In the case of the Type A verifications the report will summarize the basis for the IVC's finding of the adequacy of the action (or discrepancy) and reference supporting data or reports. The conclusion of the verification report, whether Type A or B verification, will be a finding of whether the authorized limits for the remedial action were met and a statement of any exceptions.

Where data are available, the post-remedial action report will include (summarize) the findings of the verification report or, as appropriate, reference the verification report and/or letter.

APPENDIX II

Certification and Verification Sample Maintenance and Archiving Process

All samples collected by the remedial action contractor and the DOE Independent Verification Contractor for the purposes of certifying a specific site or property will be logged and maintained by them until the certification process is complete.

Six months following the issuance of the Federal Register notice of certification and the availability of the docket in the public document room, the certification/verification sample archival process will be initiated. At that time or thereafter, the IVC will assemble, log, and archive a representative number (as defined below) of certification *or* verification samples (at least 500g/sample, if possible) to be maintained over a 5-year period. These samples will be held as evidence of the adequacy of the remedial action and to backup the certification docket. All other samples may be disposed of (in an appropriate manner) by the contractors following the establishment of the sample archives for the particular site and/or vicinity properties.

The majority of the archival samples are expected to be derived from the IVC collection of samples; however, the IVC will review his samples and those of the remedial action radiological contractor to determine if any of these samples should be consolidated into the archives.

The IVC will provide the remedial action contractor with guidelines and specific directions regarding samples required for the archive from his inventory. The remedial action contractor will be 'responsible for the correct labeling, packaging, and transmittal of these samples to the IVC and for providing information accurately identifying the locations where the samples were derived. Guidance with regard to sample collection, handling, labeling, and storage is available in documents prepared or referenced in the generic verification plan by the IVC.

The IVC will take similar actions with their samples and will consolidate the two sets of samples into one group with common keys and legends identifying the sampling locations. These samples will then be archived by the IVC. The IVC may then take steps to appropriately dispose of any excess samples and will notify the remedial action contractor that they are free to do the same. The archived samples will be held for a minimum period of five years and the IVC will notify DOE and obtain approval prior to disposal of the archived samples.

SAMPLE SELECTION

The selection of samples for the archives will be done in a systematic manner. Approximately 10 percent, but not less than five samples, of all certification or verification samples taken for each site, vicinity property, or each group of properties will be archived. Proper care shall be taken to ensure that adequate samples are taken for each site. Grouping of vicinity properties for the purpose of sample archiving is permissible in cases where many small vicinity properties are located near one another, contamination removed from the area were of a similar nature, or the remedial actions were completed during the same construction period or season without any significant interruptions. Samples from a site and vicinity properties which are contiguous with the site and were decontaminated during the same period may also be included in the same sample selection process and archived together.

In general, samples will be selected out of the total sample population with the only restriction being that the samples should provide a representative areal cross section of the site or properties being certified.

For cases where some special circumstances exist, a greater number of samples may be selected to better represent the post-remedial action conditions at the location of interest. Examples of such locations include:

Areas that had exceptionally high concentrations of radionuclides prior to remedial action.

Areas that were the subject of some conflict, question, or discrepancy between DOE and other groups, including owners, states, other Federal agencies, or local groups.

Areas at which the IVC and the radiological contractors data initially disagreed or areas where the independent verification survey identified discrepancies that had to be resolved.

Areas for which exceptions to the designated site criteria were requested.

The number of samples archived will be proportional to the area of the site. If the area of concern covered a large area (several hundred square meters) and was very non-uniform in nature (varied isolated depths, varied concentrations and nuclide make-up) extra samples should be preserved.

DOE F 1325.8
(8-89)
EFG (07-90)

United States Government

098808

Department of Energy

memorandum

DATE: DEC 2 1992

REPLY TO ATTN OF: EM-421 (W. A. Williams, 903-8149)

SUBJECT: Authority Determination--Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts

TO:

The File

The attached review documents the basis for determining whether DOE may have authority for taking remedial action at the site owned and operated by the former Chapman Valve Manufacturing Company in Indian Orchard, Massachusetts under the Formerly Utilized Sites Remedial Action Program (FUSRAP). The Chapman Valve facility was used by the AEC for uranium metal machining during the late 1940s. The following factors are significant in reaching this determination:

- Available records indicate that Chapman Valve was directly supervised by the AEC, which also approved the use of the facility;
- As a part of the operations at the site, there were requirements concerning security, materials accountability, health, and safety. These were controlled by the AEC directly;
- The uranium metal machined at the site was owned by the government. Other government owned property was furnished to support the production activities;
- Final cleanup of the site was supervised by the AEC and other AEC contractors; and
- Radiological survey activities conducted in September 1991 indicate the presence of residual uranium contamination at the site in excess of DOE's guidelines.

A draft copy of this authority review was furnished to the Office of General Counsel for review. The draft authority was modified to incorporate the changes recommended by that office.

After review of the available historical records and the authority review, I have determined that the Department of Energy has authority to conduct remedial action at the former Chapman Valve Manufacturing Company in Indian Orchard, Massachusetts.

W. Alexander Williams, PhD
Designation and Certification Manager
Division of Off-Site Programs
Office of Eastern Area Programs
Office of Environmental Restoration

Attachment

cc:
S. Miller, GC-11

098808

bcc:
Weston
Booz, Allen and Hamilton, Inc.

Distribution:
EM-40 (2)
EM-42 (3)
Williams Reader

EM-421:wagoner:djn:903-8145:12/2/92:Chapman.aut

Williams
EM-421
12/3/92

Wagoner
EM-421
12/2/92

Authority Review for the Chapman Valve Manufacturing Company Site in Indian Orchard, Massachusetts

1. INTRODUCTION

As part of the Formerly Utilized Sites Remedial Action Program (FUSRAP), the U.S. Department of Energy (DOE) has reviewed available information on the Chapman Valve Manufacturing Company site in Indian Orchard, Massachusetts. This site is being investigated as a candidate for inclusion in the FUSRAP, which includes certain sites that were previously involved with activities of the Manhattan Engineering District (MED) or U.S. Atomic Energy Commission (AEC), both DOE predecessors. Such sites may require remedial action, if they have residual contamination from those previous activities. This review is conducted to determine whether DOE would have the authority to conduct remedial action at the Chapman Valve site.

The site is located in Indian Orchard, Massachusetts, at 203 Hampshire Street and is bounded by Moxon and Goodwin Streets. Indian Orchard is a suburb of Springfield. The site apparently had a shipping address of Oak Street in the 1940s.

Although principally a supplier of regular and special valves, Chapman Valve machined extruded uranium rods for the Brookhaven Laboratory in 1948 and may also have conducted rolling operations on uranium metal in 1949. A radiological survey conducted at the direction of DOE in 1991 indicated that residual uranium contamination was present inside the facility.

The remainder of this review consists of the following sections:

2. Operational History
3. Current Conditions
4. Authority Analysis
5. Discussion and Conclusions
6. Copies of References

The information presented in these sections is in summary form. Pertinent references are identified in the text and provided in Section 6 for further use.

2. OPERATIONAL HISTORY

The Chapman Valve Manufacturing Company was a principal supplier of regular and special valves and manifolds for the MED and the AEC Y-12 Plant. Contracts identified to date include W-7412-eng-1, W-7401-eng-136, and W-7401-eng-137.

Records also indicate that Chapman Valve, under contract with the Brookhaven Laboratory, machined uranium metal during the period January through November 1948 (Wolf 1947). By letter (Shugg and Sturges) dated January 9, 1948, shipment of approximately 26 tons of Brookhaven extruded rods was directed to Chapman Valve's Oak Street facility in Indian Orchard,

Massachusetts. However, machining operations did not start until May 1948 (Wolf 1948a through 1948e). Chapman Valve's Monthly Accountability Report for Source and Fissionable (SF) Material, January through July 1948, indicates that 141,223 pounds of SF Material were received and 42,571 pounds were shipped during the period (Morgan 1948a). No reference has been found that would indicate AEC licensing of the Chapman Valve facility during this period.

During the course of operations, uranium rods were brought to the facility by railroad car on a track that ran immediately adjacent to the building (Ungerland 1987), according to E. Dvorchak, a former Chapman Valve employee, in an affidavit. The rods were cut by a mechanical saw and then machined to the shape requested by the customer. Termination of these operations was indicated in a Chapman Valve letter dated November 8, 1948 (Fox), that requested termination of AEC film badge services. Other records addressing the operation and the disposition of materials include Morgan 1948b and 1948c, Karl 1948, Sargent 1948, Hunter 1949, Morgan 1949, Fox 1949, Wolf 1949, Williams 1949, and Burman 1949.

During the work in 1948, Chapman set aside approximately one-third of Department 40 area (200 feet in length, 60 feet in width) at the Chapman Plant Site, 203 Hampshire Street to engage in this program for Brookhaven Labs. Segregation of the facility was achieved by installing a floor to ceiling wooden partition. The ceiling was more than 50 feet high. Special services provided for the facility included building shields, quenching tanks, suction systems and duct work. In 1948, a fire involving uranium metal occurred.

When the program was completed at the end of 1948, all the machinery and equipment used in the program and the wood block floors were removed. Any uranium cuttings that were still on site were set aside and also removed. After the equipment and residual cuttings were removed, a crew entered the premises and washed down the walls with detergent.

The preceding discussion of operational history was based primarily upon information from historical records assembled through 1987 and the August 1991 radiological survey (Foley and Uziel 1992). However, a September 1991 search for historical documents revealed that Chapman Valve Manufacturing Company probably performed rolling operations on uranium in 1949 (Williams 1949) and was at least considered for additional machining in 1951 (Fry 1951).

The only other indication found of the use of radioactive material at the Indian Orchard site is in the Nuclear Regulatory Commission License No. 20-00518-02 which was issued to the Crane Company for possession of a sealed source or sources used in industrial radiography. The license was terminated on February 20, 1987, by the Nuclear Materials Safety and Safeguards Branch, Region I, King of Prussia, Pennsylvania.

3. CURRENT CONDITIONS

In 1991, a DOE-directed radiological survey was performed by the Oak Ridge National Laboratories (Foley and Uziel 1992).

The building was still standing and in reasonably good condition. However, the survey team judged the roof unsafe to conduct a survey. The walls had been painted since 1948. The wooden partition had been removed. The space had been vacant since 1987 when Crane discontinued all manufacturing at Indian Orchard. The survey indicated the presence of residual uranium contamination, in excess of DOE guidelines (U.S. Department of Energy Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites. Revision 2, March 1987). The contamination was found inside on floors, walls, and overhead beams. The uranium in one dirt sample was slightly enriched in the U-235 isotope; the presence of enriched uranium is indicative of AEC activities at the site.

4.0 AUTHORITY ANALYSIS

The authority determination is made according to FUSRAP protocol by considering the answers to five questions. The answers to these questions are based upon information from all sources, including the results of the radiation surveys are provided below.

4.1 Was the site/operation owned by a DOE predecessor or did a DOE predecessor have significant control over the operations or site?

A DOE predecessor never owned the site; however, Chapman Valve performed the work under AEC direction and was subject to accountability controls for SF material. The work area used for the uranium work was separate and restricted. Over 27,000 pounds of metal scrap, oxides, sweepings, etc., were identified for removal several months after the contract was complete.

4.2 Was a DOE predecessor agency responsible for maintaining or ensuring the environmental integrity of the site (i.e., was it responsible for clean up)?

Yes. AEC established a health and safety program at the facility, including the use of film badges. Following the uranium fire in 1948, urine samples were taken from those who fought the fire or cleaned up afterward, and AEC officials apparently supervised the cleanup by Chapman Valve employees. Brookhaven Medical Group surveyed residues and contaminated materials before they were shipped to the Electro-Met facility. The operations were frequently inspected by AEC officials.

4.3 Is the waste or radioactive material on the site the result of DOE predecessor related operations?

Yes. The 1991 survey indicated that the residual uranium-238 contamination found at Chapman Valve was typical of MED/AEC operations. There are no indications that work with uranium metal was conducted at the site after the AEC operations were terminated.

4.4 Is the site in need of further clean up and was the site left in non-acceptable condition as a result of DOE predecessor related activities?

Yes. Although there were some cleanup activities, there is residual radioactive contamination at the site in excess of current DOE guidelines that is attributable to DOE-predecessor activities.

4.5 Did the present owner accept responsibility for the site with knowledge of its contaminated condition and that additional remedial measures are necessary before the site is acceptable for use without radiological restrictions?

The former Chapman Valve became a part of the current Crane Company in 1959 (Foley and Uziel 1992). An NRC license to Crane Company, for sealed sources or sources used in industrial radiography, was terminated in 1987.

There is no indication that the owner believed residual contamination to be present, because the site had been washed down and decontaminated (Ungerland 1987). A railroad-car full of metal, scrap oxide, contaminated equipment, gravel, dirt, and dust was shipped to the Electro-Met facility, with some part, in turn, reshipped to the Lake Ontario Ordnance Works for storage (Hunter 1949).

5. DISCUSSION AND CONCLUSIONS

Surveys of the former Chapman Valve site indicate residual uranium contamination in excess of DOE guidelines that is attributed to machining of uranium for the AEC.

Based upon the results of the surveys, available records, and interviews with former workers at the site, there is sufficient evidence to indicate authority for remedial action at the former Chapman Valve site under the Atomic Energy Act through FUSRAP.

6. COPIES OF REFERENCES

The following is the list of references that are provided in this section.

a. Wolf, B.S., 1947: Monthly Report. AEC memo to H.B. Fry. December 2.

b. Shugg, C. and D.G. Sturges, 1948: Monthly Shipment. AEC letter to D.H. Lauder, General Electric Company. January 9.

c. Wolf, B.S., 1948a: Weekly Activity Report - Period Jan. 8 to Jan. 14. AEC memo to C. Slesser. January 14.

d. Wolf, B.S., 1948b: Weekly Activity Report - Period Jan. 15 to Jan 21. AEC memo to C. Slesser. January 21.

e. Wolf, B.S., 1948c: Weekly Activity Report - Feb. 26 to March 3. AEC memo to C. Slesser. March 3.

f. Wolf, B.S., 1948d: Weekly Activity Report - April 8 to April 14. AEC memo to C. Slesser. April 14.

g. Wolf, B.S., 1948e: Weekly Activity Report - April 28 to May 5. AEC memo to C. Slesser. May 6.

h. Morgan, J.P., 1948a: Accounting for Source and Fissionable Material. AEC letter to G.E. Fox, Chapman Valve Manufacturing, including monthly accountability report for Chapman Valve. August 24.

i. Morgan, J.P., 1948b: Scrap Metal from Chapman Valve. AEC memo to P.J. Epp. October 7.

j. Fox, G.E., 1948. Letter from Chapman Valve to M. Eisenbud, Radiological Laboratory. November 8.

k. Karl, C.L., 1948: Oxides Scrap from Chapman Valve. AEC memo to J.P. Morgan. November 18.

l. Sargent, E.C., 1948: Study of "Unaccounted for Material" - Uranium. U.S. Government memo to files. November 24. (Chapman may be one of the "rolling" contractors; see Burman 1949.)

m. Morgan, J.P., 1948c: Chapman Valve Contaminated Material. AEC memo to F.J. Epp. December 3.

n. Hunter, D., 1949: Miscellaneous Scrap Metal from Chapman Valve to Lake Ontario Ordnance Works. AEC memo to J. S. Quidor. January 17.

o. Morgan, J.P., 1949: Shipment from Chapman Valve. AEC memo to P.J. Epp. January 17.

p. Fox. G.E., 1949. Letter from Chapman Valve to B.S. Wolf, AEC. January 19.

q. Wolf, B.S., 1949. Letter from AEC to G.E. Fox, Chapman Valve. January 27.

r. Williams, W.J., 1949: SF Materials Lost or Unaccounted For. AEC memo to W.E. Kelly. February 21.

s. Burman, L.C., 1949: Notes on a Program for Study of Uranium Losses. U.S. Government memo to H.M. Chadwell. April 19.

t. Fry, H.B., 1951: Requirements. AEC memo to W.E. Kelly/files. May 14.

u. Ungerland, T.J., 1987: Crane - Indian Orchard. Letter from Crane to J.J. Fiore, DOE. Includes affidavit of December 8, 1987, by Edward Dvorchak. December 14.

v. Foley, R.D. and M.S. Uziel, 1992: Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (C10001). ORNL/RASA-92/1, July 1992.

DOE F 1325.8
(8-89)
EFG (07-90)

098808

United States Government

Department of Energy

memorandum

DATE: DEC 1 5 1992

1993 JAN -4 PM 1:24

REPLY TO ATTN OF: EM-421 (W. A. Williams, 903-8149)

SUBJECT: Authorization for Remedial Action at the Former Chapman Valve Manufacturing Company facility, Indian Orchard, Massachusetts

TO: L. Price, OR

The former Chapman Valve Manufacturing Comapany facility at 203 Hampshire Street, Indian Orchard, Massachusetts, is designated for remedial action under the Formerly Utilized Sites Remedial Action Program (FUSRAP). The site is owned by the Crane Company, 100 Stamford Place, Stamford, Connecticut. This designation is based on the results of a radiological survey and conclusions from an authority review as noted in the attached Designation Summary. Copies of the radiological survey report and the authority review are provided for information.

The site has been assigned a low priority under FUSRAP protocol. The survey concluded that the property contains residual radioactive material in concentrations that exceed current guidelines on the interior surfaces of one building. However, the building is not presently in use; thus, under present conditions, no significant radiation exposures would occur to individuals who access the area. Because of the limited contamination, we recommend that cleanup of the site follow the expedited FUSRAP protocol for removal action.

The effect of this designation on the FUSRAP baseline should be evaluated, documented, and submitted for approval under the baseline change control procedures.

James W. Wagoner II
Director
Division of Off-Site Programs
Office of Eastern Area Programs
Office of Environmental Restoration

Attachments

098808

FORMERLY UTILIZED SITES
REMEDIAL ACTION PROGRAM

DESIGNATION SUMMARY FOR
CHAPMAN VALVE MANUFACTURING
INDIAN ORCHARD, MASSACHUSETTS

December 14, 1992

U.S. DEPARTMENT OF ENERGY
OFFICE OF ENVIRONMENTAL RESTORATION

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INTRODUCTION

The Department of Energy (DOE), Office of Environmental Restoration, has reviewed the past activities of the Atomic Energy Commission (AEC) at the former Chapman Valve Manufacturing Company site in Indian Orchard, Massachusetts, and has completed a radiological survey of the site (Foley and Uziel 1992). DOE has determined that the residual radioactive materials at the site exceed current guidelines (DOE 1990) for use without radiological restrictions.

Based on a review of the available historical documentation and the results of a radiological survey, DOE has concluded that this site shall be designated for remedial action under the Formerly Utilized Sites Remedial Action Program (FUSRAP). The survey results indicate that the residual radioactivity is limited in extent and poses no immediate risk to workers. Therefore, the site has been assigned a low priority. The remainder of this report summarizes the site information and the designation decision.

BACKGROUND

Site Function

The following is based upon the Authority Review (Williams 1992).

The Chapman Valve Manufacturing Company was a principal supplier of regular and special valves and manifolds for the Manhattan Engineer District (MED) and the Atomic Energy Commission (AEC) Y-12 Plant. Contracts identified to date include W-7412-eng-1, W-7401-eng-136, and W-7401-eng-137.

Records also indicate that Chapman Valve, under contract with the Brookhaven Laboratory, machined uranium metal during the period of January through November 1948. By letter dated January 9, 1948, shipment of approximately 26 tons of Brookhaven extruded rods was directed to Chapman Valve's Oak Street facility in Indian Orchard, Massachusetts. However, machining operations did not start until May 1948. Chapman Valve's Monthly Accountability Report for Source and Fissionable (SF) Material, January through July 1948, indicates that 141,223 pounds of SF Material were received and 42,571 pounds were shipped during the period. After the machining work, Chapman Valve may have conducted rolling operations on uranium metal for the AEC.

The only other indication found of the use of radioactive material at the Indian Orchard site is in the Nuclear Regulatory Commission (NRC) License No. 20-00518-02 which was issued to the Crane Company for possession of a sealed source(s) used in industrial radiography. The license was terminated on February 20, 1987, by the Nuclear Materials Safety and Safeguards Branch, NRC Region I, King of Prussia, Pennsylvania.

Site Description

The following is based upon information in the survey report (Foley and Uziel 1992) and Authority Review (Williams 1992).

Indian Orchard is a suburb of Springfield, Massachusetts. In 1948 Chapman Valve set aside approximately one-third of Department 40 (200 feet in length, 60 feet in width) at the Chapman Plant Site, 203 Hampshire Street to engage in this program for Brookhaven Labs. Segregation of the facility was achieved by installing a floor to ceiling wooden partition. The ceiling was more than 50 feet high. Special services provided for the facility included building shields, quenching tanks, suction systems and duct work.

As of 1991, the building was still standing in reasonably good condition. However, the survey team judged the roof unsafe. The space had been vacant since 1987 when Crane discontinued all manufacturing at Indian Orchard.

Owner History

The records indicated that since 1981, the Crane Company has been the occupant of the site which was formerly occupied and operated by the Chapman Valve Manufacturing Company. Transfer of ownership or succession arrangements between the Crane Company and the Chapman Valve Manufacturing Company have not been researched.

Radiological History and Status

Correspondence discovered to date (Williams 1992) pertaining to the radiological history of the site indicates that:

- When the machining program was completed at the end of 1948, all the machinery and equipment used in the program and the wood block floors were removed by the customer's agent. Any uranium cuttings which were still on site were set aside and also removed. After the equipment and residual cuttings were removed, a crew entered the premises and washed down the walls with detergents.

- A health and safety program was set up at the Chapman Valve facility by the AEC during the machining work.

- There was a fire in the restricted area of the AEC project, probably in June 1948. The exact nature of the fire and extent of the damage is unknown. The involvement of uranium metal in the fire is suggested by a Chapman Valve letter dated January 19, 1949, that indicated urine samples were taken from the men who fought and put out the fire and the men who were involved in the cleanup process.

- Termination of uranium operations is indicated in a Chapman Valve letter dated November 8, 1948, that requested termination of AEC film badge services.

- When the contract work was completed, Chapman Valve had in their possession over 27,000 pounds of metal scrap, oxides, sweepings, etc. This material was apparently not removed until several months after the contract was completed.

In August 1991, at the request of the Department of Energy, a site survey was conducted by Oak Ridge National Laboratory personnel (Foley and Uziel 1992). Elevated radioactivity was found within the building, including portions of the floor and walks as well as beams supporting the roof. In some of these locations the levels of radioactivity exceeded DOE guidelines (USDOE 1987, USDOE 1990)

Authority Review

In 1992, the DOE determined that it had authority to conduct remediation at this site (USDOE 1986, Williams 1992).

- Available records indicate that Chapman Valve was directly supervised by the AEC, which also approved the use of the facility. Chapman Valve was subject to accountability controls for SF material. The work area used for the uranium work was separate and restricted. Over 27,000 pounds of metal scrap, oxides, sweepings, etc. were identified for removal several months after the contract was complete.

- AEC established a health and safety program at the facility, including the use of film badges. Following the uranium fire in 1948, urine samples were taken from those who fought the fire or cleaned up afterward. Brookhaven Medical Group surveyed residues and contaminated materials before they were shipped to Electro-Met.

- The 1991 survey indicated that the residual uranium contamination found at Chapman Valve was typical of MED/AEC operations. There are no indications that work with uranium metal was conducted at the site after the AEC operations were terminated.

- Residual uranium (attributable to DOE predecessors) at the site is in excess of current DOE guidelines.

- There is no indication that the owner believed residual uranium to be present, because the site had been washed down and decontaminated by the AEC and other AEC contractors.

DESIGNATION DETERMINATION

Available records indicate a direct AEC involvement in the uranium machining operation conducted at the Chapman Valve facility. A radiological survey indicates that uranium remains on the premises; this residual uranium is the result of the AEC work at the facility. Based on a review of the available historic documents, DOE has authority to perform the needed remedial action at the Chapman Valve site. Accordingly, the site is designated for remedial action under FUSRAP.

REFERENCES

Foley, R.D. and M.S. Uziel, 1992: Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (C10001). ORNL/RASA-92/1, July 1992.

United States Department of Energy (USDOE), 1986: Formerly Utilized Sites Remedial Action Program, Summary Protocol, Identification - Characterization - Designation - Remedial Action - Certification. Office of Nuclear Energy, January.

USDOE, 1987: U.S. Department of Energy Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites. Revision 2, Office of Nuclear Energy, March.

USDOE, 1990: Radiation Protection of the Public and the Environment. Office of Environment, Safety, and Health, February 8.

Williams, W. A., 1992: Authority Determination for Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts. USDOE, (date)

DOE F 1325.8
(8-89)

067327

United States Government

Department of Energy

Oak Ridge Operations

memorandum

DATE March 28, 1990

REPLY TO ATTN OF SE-311 Dillow

SUBJECT ISSUANCE OF DOE ORDER 5400.5, "RADIATION PROTECTION OF THE PUBLIC AND THE ENVIRONMENT"

TO: J. C. Hall, Assistant Manager for Enriching Operations, EO-20
R. L. Egli, Assistant Manager for Energy Research and Development, ER-10
W. R. Bibb, Assistant Manager for Defense Programs, DP-80
W. D. Adams, Assistant Manager for Environmental Restoration and Waste Management, EW-90

The subject order was signed by the Office of Management and Administration (MA) on February 8, 1990, but not received at ORO until March 16, 1990. Paragraph 4 of the General Section of the order (attachment 1) requires that within one month from the date of issuance of the order (now early April because of slippage), Field Office Managers are to provide the appropriate Program Office with a copy to EH-1 for review and comment: (1) a certification for those areas covered by the order for which Site/Operations Offices are in compliance; and/or (2) a request for exemption for areas of non-compliance that includes a plan for achieving compliance. The appropriate Program Office is to submit to EH-1 by May 8, 1990, the certification and/or request for exemption(s). Compliance plans accompanying a request for exemption shall include a schedule of activities which will lead to compliance with the requirements of this order.

The last sentence of paragraph 4 of the introductory part of the order regarding the schedule for achieving compliance has been modified by the Environmental Protection Division (ENVPD) to reflect a page change that will be issued by EH. This change removes the ambiguity regarding schedules for achieving compliance.

Please copy the ENVPD on any correspondence to Program Offices and to EH-1 regarding certification of compliance and requests for exemptions for areas of noncompliance including plans and schedules for achieving compliance.

If you have any questions or require additional information, please contact Weldon Dillow of the Environmental Protection Division at 6-1354.

R. R. Nelson
Assistant Manager
for Environment, Safety and Quality

Attachments:
1. DOE Order 5400.5
2. Memorandum from EH-1

cc w/attachments:
E. W. Gillespie, EO-221
D. R. Allen, EO-222
J. A. Reafsnyder, ER-10
R. J. Spence, DP-81
G. Westerbeck, DP-84
W. M. Seay, EW-93
M. M. Heiskell, DP-85
S. H. McCracken, WSSRAP, EW-94
G. E. Butterworth, K-303-8, MS 7314, ORGDP
R. M. Keyser, 9116, MS 8098, Y-12
F. C. Kornegay, 4500-N, ORNL
S. L. Shell, PAD
R. E. Blake, PORTS
B. Roach, ORAU
M. J. Galper, WMCO
E. P. Marsh, RMI, Astabulah
G. K. Hovey, Bechtel National, Inc.
M. E. Mitchell, K-1001, MS 7155, ORGDP

DOE F 1325.8
(10-88)

United States Government 067327 Department of Energy

EPD 3/19

memorandum

DATE March 12, 1990

REPLY TO ATTN OF EH-1

SUBJECT Compliance Statements/Exemptions and Compliance Plans for DOE 5400.5, "RADIATION PROTECTION OF THE PUBLIC AND ENVIRONMENT"

TO Distribution

The DOE Order for "RADIATION PROTECTION OF THE PUBLIC AND THE ENVIRONMENT," DOE 5400.5, was signed by the Office of Management and Administration (MA) on February 8, 1990. The Order has been distributed, but, it is our understanding that it has not yet reached all field elements within the Department. As a result, Environment, Safety and Health (EH) will be accepting the statements certifying compliance or compliance plans with requests for exemption as required in Section 4 of the Order until May 8, 1990. On this basis, it is recommended that the Program Offices receive input from the Operations Offices early in April 1990. EH-1 should receive copies of these submittals.

If an exemption is requested, it will identify those areas where compliance cannot yet be certified, for example, areas where outfall discharges exceed the Derived Concentration Guides (DCG) and a Best Available Technology (BAT) assessment has not been completed. The request for exemption will be accompanied by a plan for compliance. The primary element in this plan is a schedule for assessing the compliance issues identified in the request for exemption.

This plan may contain actual schedules for achieving compliance; however, where issues are more complex, the plan should identify the schedule for the development of information needed to determine how and when compliance will be achieved. In the latter situation, the plan for compliance will identify the steps that will lead to an actual schedule of implementation and the major milestones associated with the process. For example, if compliance with the BAT requirements of this Order are in question, the compliance plan included with the exemption request will identify, as a milestone, the submission of the BAT analysis Implementation Plan, by August 8, 1990, as required by paragraph II.3.a.(1)(b) of DOE 5400.5.

The BAT analysis Implementation Plan will identify areas requiring BAT analyses and a schedule for the required analyses and associated decisions. The Implementation Plan will identify the schedules for resolving the BAT question such that compliance with paragraph II.3.a(1) can be achieved at the earliest practicable time. Once this plan is issued and approved, the facility is effectively in compliance with the BAT related portions of 5400.5 so long as it is completing assessments in accordance with the schedules in the Implementation Plan.

The Order recognizes that there are no practicable control technologies available for tritium in liquid waste streams.

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Rather, process alternatives and practices that limit the amount of tritium entering the waste stream should be evaluated in accordance with the DOE ALARA Policy. Tritium is, therefore, exempted from the BAT requirements of Section II.3. Instead compliance with the Order is achieved through application of the ALARA process (paragraph II.3.e.(2)). However, all releases must comply with the dose limits provided in this Order.

One element of the draft of this order was identified as a major issue and resulted in several non-concurrences. This relates to the release limits for facilities and equipment having surfaces contaminated with transuranics. The subject limits have been reserved until the issue is resolved under the normal resolution process. Additional guidance will be provided by EH on this and other issues in the near future.

If you have questions on the submission of these plans or the Order in general, contact Mr. Andrew Wallo, EH-231 (FTS-896-4996).

Peter N. Brush
Acting Assistant Secretary
Environment, Safety and Health

DISTRIBUTION: 12/7/89

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Assistant Secretary, Management and Administration, MA-1
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cc:

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Tom Lukow, WIPP Project Office (WIPP-AL)
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cc: Other Organizations

DOE Headquarters Hazardous Waste Regulatory Workgroup
National Low-Level Waste Management Program, EG&G Idaho
Hazardous Waste Remedial Action Program (HAZWRAP)
Pacific Northwest Hazardous Waste Research, Development and Demonstration Center

U.S. Department of Energy

Washington, D.C.

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ORDER

DOE 5400.5

2-8-90

SUBJECT RADIATION PROTECTION OF THE PUBLIC AND THE ENVIRONMENT

1. PURPOSE. To establish standards and requirements for operations of the Department of Energy (DOE) and DOE contractors with respect to protection of member of the public and the environment against undue risk from radiation.

2. SUPERSESSION. DOE 5480.1A, ENVIRONMENTAL PROTECTION, SAFETY, AND HEALTH PROGRAM FOR DOE OPERATIONS, of 8-13-81, Chapter XI that addressed public and environmental radiation protection standards and control practices.

3. SCOPE. The provisions of this Order apply to all Departmental Elements and contractors performing work for the Department as provided by law and or contract and as implemented by the appropriate contracting officer.

4. IMPLEMENTING PROCEDURES AND REQUIREMENTS. This Order becomes effective 3 months from the date of issuance. Within 1 month from the date of issuance of the Order, Heads of Operations Offices shall provide to the appropriate Program Office with copy to EH-1 for review and comment: (1) a certification for those areas covered by the Order for which Site/Operations Offices are in compliance; and/or (2) a request for exemption for areas of non-compliance that includes a Plan for achieving compliance. Within 2 months of issuance, the appropriate Program Office will submit to EH-1 the certification and/or the request for exemption(s). The compliance plans accompanying the request for exemption shall include a schedule of activities which will lead to compliance with the requirements of this order.

5. POLICY. It is the policy of DOE to implement legally applicable radiation protection standards and to consider and adopt, as appropriate, recommendations by authoritative organizations, e.g., the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP). It is also the policy of DOE to adopt and implement standards generally consistent with those of the Nuclear Regulatory Commission (NRC) for DOE facilities and activities not subject to licensing authority.

6. OBJECTIVES.

   a. Protecting the Public. It is DOE's objective to operate its facilities and conduct its activities so that radiation exposures to members of the public are maintained within the limits established in this Order and to control radioactive contamination through the management of real and personal property. It is also a DOE objective that potential exposures to members of the public be as far below the limits as is reasonably achievable (ALARA) and that DOE facilities have the capabilities, consistent with the types of operations conducted, to monitor routine and non-routine releases and to assess doses to members of the public.

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b. Protecting the Environment. In addition to providing protection to members of the public, it is DOE's objective to protect the environment from radioactive contamination to the extent practical.

7. LEGISLATIVE AUTHORITY. The Atomic Energy Act of 1954, as amended, authorizes the Department to protect the health and safety of the public against rad tion in conducting the Department's programs.

8. REFERENCES.

a DOE 1324.2A, RECORDS DISPOSITION, of 9-13-88, which prescribes policies, procedures, standards, and guidelines for the orderly disposition of records of the DOE and its operating contractors.

b DOE 5000.3, UNUSUAL OCCURRENCE REPORTING SYSTEM, of 11-7-84, which establishes DOE policy and provides instructions for reporting, analyzing, and disseminating information on programmatically significant events.

c. DOE 5400.1, GENERAL ENVIRONMENTAL PROTECTION PROGRAM REQUIREMENTS, of 11-9-88, which establishes general environmental protection requirements.

d. DOE 5400.2A ENVIRONMENT COMPLIANCE ISSUE COORDINATION, of 8-18-87, which establishes requirements for coordination of significant environmental compliance issues.

e DOE 5400.4, COMPREHENSIVE ENVIRONMENTAL RESPONSE, COMPENSATION, AND LIABILITY ACT PROGRAM, of 10-6-89, which establishes requirements for hazardous waste cleanup and notification.

f. DOE Orders in the 5400 series dealing with radiological effluent monitoring and environmental surveillance, which describes requirements and provides guidance for monitoring effluents and conducting environmental surveillance.

g. DOE 5440.1C, IMPLEMENTATION OF THE NATIONAL ENVIRONMENTAL POLICY ACT, of 4-9-85, which establishes DOE policy for implementation of the National Environmental Policy Act of 1969.

h. DOE 5480.1B, ENVIRONMENT, SAFETY, AND HEALTH PROGRAM FOR DEPARTMENT OF ENERGY OPERATIONS, of 9-23-86, which outlines environmental, safety, and health protection policies and responsibilities.

i. DOE 5480.4, ENVIRONMENTAL PROTECTION, SAFETY, AND HEALTH PROTECTION STANDARDS, of 5-1-84, which identifies mandatory and reference environmental, safety, and health standards.

j. DOE 5480.5, SAFETY OF NUCLEAR FACILITIES, of 9-23-86, which establishes nuclear facility safety program requirements.

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k. DOE 5480.6, SAFETY OF DEPARTMENT OF ENERGY-OWNED NUCLEAR REACTORS, of 9-23-86, which establishes nuclear reactor safety program requirements.

l. DOE 5480.11, RADIATION PROTECTION FOR OCCUPATIONAL WORKERS, of 12-21-88, which establishes radiation protection standards and program requirements for workers.

m. DOE 5481.1B, SAFETY ANALYSIS AND REVIEW SYSTEM, of 9-23-86, which establishes uniform requirements for the preparation and review of safety analysis documents.

n. DOE 5482.1B, ENVIRONMENT, SAFETY, AND HEALTH APPRAISAL PROGRAM, of 9-23-86, which establishes the DOE environment, safety, and health appraisal program.

o. DOE 5483.1A, OCCUPATIONAL SAFETY AND HEALTH PROGRAM FOR DOE EMPLOYEES AT GOVERNMENT-OWNED, CONTRACTOR-OPERATED FACILITIES, of 6-22-83, which establishes requirements for the protection of the health and safety of employees at DOE contractor-operated facilities.

p. DOE 5484.1, ENVIRONMENTAL PROTECTION, SAFETY, AND HEALTH PROTECTION INFORMATION REPORTING REQUIREMENTS, of 2-24-81, which establishes procedures for the reporting of information having environmental protection, safety, or health protection significance.

q. DOE Orders in the 5500 series that outline responsibilities for emergency preparedness and response.

r. DOE 5820.2A, RADIOACTIVE WASTE MANAGEMENT, of 9-26-88, which establishes policies and guidelines for the management of radioactive waste and contaminated facilities.

s. DOE 6430.1, GENERAL DESIGN CRITERIA, of 12-12-83, which provides general design criteria for use in the acquisition of the Department's facilities and establishes responsibilities and authorities for the development and maintenance of those criteria.

t. DOE publication DOE/EH-0070, "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," of 7-88, which provides conversion factors for use in calculating dose from radionuclides external to the body.

u. DOE publication DOE/EH-0071, "Internal Dose Conversion Factors for Calculation of Dose to the Public," of 7-88, which provides conversion factors for use in calculating dose from radionuclides in the body.

v. DOE publication DOE/EV/1830-T5, "A Guide to Reducing Radiation Exposure to as Low as Reasonably Achievable," of 4-80, which provides contractor personnel with general guidance regarding programs and techniques to reduce radiation exposure to levels as low as is reasonably achievable.

w. DOE publication, "Formerly Utilized Sites Remedial Action Program; Summary Protocol: Identification - Characterization - Designation - Remedial Action - Certification," of 1-86, which provides procedures for conducting remedial actions at formerly utilized sites.

x. DOE/CH-8901, "A Manual for Implementing Residual Radioactive Material Guidelines; A Supplement to U.S. Department of Energy Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites," of 6-89, which provides guidance on the implementation of DOE residual radioactive material limits.

y. EPA Publication EPA-520/1-88-020, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," of 1988, which provides preferred dose conversion factors for use by Federal agencies.

z. Title 10 CFR Part 60, "Disposal of High-Level Wastes in Geologic Repositories," which prescribes rules governing the licensing of DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository operations area.

aa. Title 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," which establishes requirements, procedures, and criteria for licensing ISFSI.

bb. Title 40 CFR Part 61, Subpart H, "National Emission Standard for Radionuclide Emissions from Department of Energy (DOE) Facilities," which regulates radionuclide air emissions from DOE facilities.

cc. Title 40 CFR Part 141, "National Interim Primary Drinking Water Regulations (Safe Drinking Water Act)," which prescribes radionuclide concentration limits for public drinking water.

dd. Title 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," which contains the radiation dose limits for members of the public in the general environment and curie release limits for radioactive materials released into the general environment from operations within the nuclear fuel cycle operations that are associated with the production of electrical power.

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ee. Title 40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Wastes," which establishes requirements for the management and disposal of spent nuclear fuel, high-level, and transuranic wastes.

ff. Title 40 CFR Part 192, "Standards for Remedial Actions at Inactive Uranium Processing Sites," of 1-5-83, which concerns the control of residual radioactive material at designated processing or disposal sites.

gg. Title 42 U.S.C. 300, et seq., Safe Drinking Water Act, as amended, which authorizes EPA to promulgate regulations under two specific programs: the first protects the Nation's public drinking water supplies; the second protects subsurface waters by regulating underground injection of materials.

hh. Title 42 U.S.C. 2011, et seq., Atomic Energy Act of 1954, as amended, which authorizes the conduct of atomic energy activities and establishes authority for protecting the health and safety of the public.

ii. Title 42 U.S.C. 4341, et seq., National Environmental Policy Act of 1969, as amended, which establishes broad national environmental policy.

jj. Title 42 U.S.C. 7401, et seq., Clean Air Act, as amended, which provides requirements to protect and enhance the quality of the Nation's air resources, to promote the public health and welfare.

kk. Title 47 FR 47073, "Food and Drug Administration (FDA) Accidental Radioactive Contamination of Human Food and Animal Feeds; Recommendations for State and Local Agencies," which provides guidance for protecting consumers of produce following a nuclear accident.

ll. Title 48 CFR Part 923.70, "Environmental Conservation and Occupational Safety," which contains the basic provisions of the DOE Environment, Conservation and Occupational Safety Program.

mm. Title 48 CFR Part 970.23, "DOE Management and Operations Contracts, Environmental Conservation, and Occupational Safety," which contains supplemental information to Title 48 CFR Part 923.70 in providing the basic provisions of the DOE Environment, Conservation and Occupational Safety Program.

nn. International Commission on Radiological Protection (ICRP) Publication 23, "Reference Man: Anatomical, Physiological and Metabolic Characteristics," Pergamon Press, Oxford.

oo. ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection."

pp. ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers."

qq. ICRP Publication 45, "Quantitative Bases for Developing a Unified Index of Harm."

rr. ICRP Publication 48, "The Metabolism of Plutonium and Related Elements."

ss. National Council on Radiation Protection and Measurements (NCRP) Report No. 91, "Recommendations on Limits for Exposure to Ionizing Radiation."

tt. Nuclear Regulatory Commission, Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," of 6-74, which establishes limits for surface contamination on materials and equipment.

uu. Nuclear Regulatory Commission publication, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material," of 7-82.

vv. Oak Ridge National Laboratory (ORNL), Publication CCC - 475, 11-86, "CAAC - Code System for Implementation of Atmospheric Dispersion Assessment Required by the Clean Air Act." RSIC Computer Code Collection.

9. RESPONSIBILITIES AND AUTHORITIES.

a. The Secretary. Many provisions in this Order permit and/or necessitate the exercise of discretion and/or judgment in carrying out the requirements of the Order. In those instances, the determination of whether, in the exercise of such discretion and/or judgment, the requirements of this Order were complied with rests initially with the relevant Department authority and, ultimately, with the Secretary. The Secretary retains the sole and final authority to determine what acts are necessary to comply with this Order. Further, the Secretary retains the authority to suspend any and all requirements under this Order whenever the Secretary deems it necessary. This authority may be delegated by the Secretary as appropriate.

b. In addition to those responsibilities and authorities contained in DOE 5480.1B and DOE 5400.1, the following responsibilities and authorities are assigned:

(1) Assistant Secretary for Environment, Safety and Health (EH-1)

(a) Develop DOE public and environmental radiation protection policy and requirements; and

(b) Approve, if warranted, specific exceptions to this Order, pursuant to provisions in DOE 5400.1 and DOE 5820.2A.

(2) Program Assistant Secretaries and Directors of Energy Research (ER) and Civilian Radioactive Waste Management (RW)

(a) Implement DOE public and environmental radiation protection policy and requirements in their respective programs; and

(b) Implement, if warranted, interim control strategies proposed by field organizations pursuant to this Order and DOE 5820.2A.

(3) Managers of Operations Offices

(a) Implement provisions of this Order for their respective activities;

(b) Maintain appropriate capabilities at each operating site for monitoring and assessing routine and unplanned releases of radioactive materials, with respect to the characteristics of radioactive material released and the release modes, consistent with the types of operations conducted;

(c) Process specific requests for exceptions to this Order pursuant to paragraph II.1a(4); and

(d) Temporarily suspend the requirements of this Order when doing so is in their judgement necessary to minimize damage to life or property or to protect public health or safety. Whenever this provision is invoked, such suspension and the reason therefore are to be reported to EH-1 at the earliest practicable time.

(4) Director, Naval Nuclear Propulsion Program: Executive Order 12344, statutorily prescribed by PL 98-525 (42 USC 7158 note) establishes the responsibilities and authority of the Director, Naval Nuclear Propulsion Program (who is also the Deputy Assistant Secretary for Naval Reactors within the Department) over all facilities and activities which comprise the Program, a joint Navy-DOE organization. The policy principle promoted by these executive and legislative actions is cited in the Executive Order as "...preserving the basic structure, policies, and practices developed for this Program in the past...". Accordingly, The Naval Nuclear Propulsion Program is exempt from the provisions of this Order. The Director shall maintain an environmental protection program to ensure compliance with applicable environmental statues and regulations. The Director and EH-1 shall cooperatively develop information exchange and other mutually beneficial programs as appropriate, consistent with PL 98-525.

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10. DEFINITIONS.

a. As Low As Reasonably Achievable (ALARA) is a phrase (acronym) used to describe an approach to radiation protection to control or manage exposures (both individual and collective to the work force and the general public) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. As used in this Order, ALARA is not a dose limit, but rather it is a process that has as its objective the attainment of dose levels as far below the applicable limits of the Order as practicable.

b. Best Available Technology (BAT) means the preferred technology for treating a particular process liquid waste, selected from among others after taking into account factors related to technology, economics, public policy, and other parameters. As used in this Order, BAT is not a specific level of treatment, but the conclusion of a selection process that includes several treatment alternatives.

c. Derived Concentration Guide (DCG) is the concentration of a radionuclide in air or water that, under conditions of continuous exposure for one year by one exposure mode (i.e., ingestion of water, submersion in air, or inhalation), would result in an effective dose equivalent of 100 mrem 0.1 rem (1 mSv). DCGs do not consider decay products when the parent radionuclide is the cause of the exposure (DCG values are presented in Chapter III) (1 rem = 0.01 sievert).

d. Dose Terms.

(1) Absorbed Dose is the energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest in that material. The absorbed dose is expressed in units of rad (or gray). (1 rad = 0.01 gray.)

(2) Collective Dose Equivalent and Collective Effective Dose Equivalent are the sums of the dose equivalents or effective dose equivalents of all individuals in an exposed population within an 80-km radius, for the purposes of this Order, and they are

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expressed in units of person-rem (or person-sievert). When the collective dose equivalent of interest is for a specific organ, the units would be organ-rem (or organ-sievert). For purposes of this Order, the 80-km distance shall be measured from a point located centrally with respect to major facilities or DOE program activities.

(3) Committed Dose Equivalent is the predicted total dose equivalent to a tissue or organ over a 50-year period after a known intake of a radionuclide into the body. It does not include contributions from external dose. Committed dose equivalent is expressed in units of rem (or sievert).

(4) Committed Effective Dose Equivalent is the sum of the committed dose equivalents to various tissues in the body, each multiplied by the appropriate weighting factor. Committed effective dose equivalent is expressed in units of rem (or sievert).

(5) Deep Dose Equivalent, as used in this Order, means the dose equivalent in tissue at a depth of 1 cm deriving from external (penetrating) radiation.

(6) Dose Equivalent is the product of absorbed dose in rad (or gray) in tissue and a quality factor. Dose equivalent is expressed in units of rem (or sievert).

(7) Effective Dose Equivalent is the summation of the products of the dose equivalent received by specified tissues of the body and a tissue-specific weighting factor. This sum is a risk-equivalent value and can be used to estimate the health-effects risk of the exposed individual. The tissue-specific weighting factor represents the fraction of the total health risk resulting from uniform whole-body irradiation that would be contributed by that particular tissue. The effective dose equivalent includes the committed effective dose equivalent from internal deposition of radionuclides and the effective dose equivalent due to penetrating radiation from sources external to the body. Effective dose equivalent is expressed in units of rem (or sievert).

(8) Public Dose means the dose received by member(s) of the public from exposure to radiation and to radioactive material released by a DOE facility or operation, whether the exposure is within a DOE site boundary or off- site. It does not include dose received from occupational exposures, doses received from naturally occurring "background" radiation, doses received as a patient from medical practices, or doses received from consumer products.

(9) Weighting Factor is tissue-specific and represents the fraction of the total health risk resulting from uniform, whole-body irradiation that could be contributed to that particular tissue. The weighting factors recommended by the ICRP (Publication 26) and used here are

Organ or Tissue	Weighting Factor
Gonads	0.25
Breasts	0.15
Red Bone Marrow	0.12
Lungs	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder[1]	0.30

[1] "Remainder means the five other organs with the highest dose (e.g., liver, kidney, spleen, thymus, adrenal, pancreas, stomach, small intestine, or upper and lower large intestine, but excluding skin, lens of the eye, and extremities). The weighting factor for each of these organs is 0.06.

(10) Quality Factor is the principal modifying factor used to ulate the dose equivalent from the absorbed dose. For the purposes of this Order, the following quality factors, which are taken from DOE 5480.11, are to be used.

Radiation Type	Quality Factor
X-rays, gamma rays, positrons, and electrons (including tritium)	1
Neutrons, <10 keV	3
Neutrons, >10 keV	10
Protons and single charged particles of unknown energy with rest mass > one atomic mass unit	
Alpha particles	20
Multiple-charged particles (and particles of unknown charge) of unknown energy	

For neutrons of known energies, the more detailed quality factors given in DOE 5480.11 may be used.

e. Members of the Public means persons who are not occupationally associated with the DOE facility or operations, i.e., persons whose assigned occupational duties do not require them to enter the DOE site. (Also see Dose Terms: Public Dose.)

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f. Monitoring Terms.

(1) Effluent Monitoring is the collection and analysis of samples or measurements of liquid and gaseous effluents for purposes of characterizing and quantifying contaminants, assessing radiation exposures of members of the public, and demonstrating compliance with applicable standards.

(2) Environmental Surveillance is the collection and analysis of samples of air, water, soil, foodstuffs, biota, and other media from DOE sites and their environs and the measurement of external radiation for purposes of demonstrating compliance with applicable standards, assessing radiation exposures of members of the public, and assessing effects, if any, on the local environment.

g. Protective Action Guides (PAG) are projected numerical dose values established by EPA, DOE, or States for individuals in the population. These values may trigger protective actions that would reduce or avoid the projected dose.

h. Radioactivity means the property or characteristic of radioactive material to spontaneously "disintegrate" with the emission of energy in the form of radiation. The unit of radioactivity is the curie (or becquerel).

i. Reference Man means a hypothetical aggregation of human (male and female) physical and physiological characteristics arrived at by international consensus (ICRP Publication 23). These characteristics may be used by researchers and public health workers to standardize results of experiments and to relate biological insult from ionizing radiation to a common base. The "reference man" is assumed to inhale 8400 cubic meters of air in a year and to ingest 730 liters of water in a year.

j. Release of Property, as used in this Order, means the exercising of DOE's authority to release property from its control after confirming that residual radioactive material (over which DOE has authority) on the property has been determined to meet the guidelines for residual radioactive material in Chapter IV or any other applicable radiological requirements. There may be instances in which DOE or other authority will impose restrictions on the management and/or use of the property if the residual radioactive material guidelines of Chapter IV are not met or if other applicable Federal, State, or local requirements cause the imposition of such restrictions.

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k. Remedial Action means those actions consistent with permanent remedy taken instead of, or in addition to, removal action in the event of a release or threatened release of a hazardous substance into the environment, to prevent or minimize the release of hazardous substances so that they do not migrate to cause substantial danger to present or future public health or welfare or the environment.

l. Residual Radioactive Material means any radioactive material which is in or on soil, air, equipment, or structures as a consequence of past operations or activities.

m. Settleable Solids means those solids suspended in waste water that are determined to be settleable using Method 209 E, Settleable Solids pp 98 and 99, 16th edition, Standard Methods for Examination of Water and Waste Water.

n. Sewerage Terms.

(1) Sewage means the waste matter that passes through sewers.

(2) Sewer means the artificial conduit, usually underground, for carrying off waste water and refuse.

(3) Sewerage means the system of sewers.

o. Soil Column is an in situ volume of soil down through which liquid wastes percolate from ponds, cribs, seepage basins, or trenches.

p. Stochastic Effects are biological effects, the probability, rather than the severity, of which is a function of the magnitude of the radiation dose without threshold; i.e., stochastic effects are random in nature. Nonstochastic Effects are biological effects, the severity of which, in affected individuals, varies with the magnitude of the dose above a threshold value.

BY ORDER OF THE SECRETARY OF ENERGY:

DONNA R. FITZPATRICK
Assistant Secretary
Management and Administration

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CHAPTER IV

RESIDUAL RADIOACTIVE MATERIAL

1. PURPOSE. This chapter presents radiological protection requirements and guidelines for cleanup of residual radioactive material and management of the resulting wastes and residues and release of property. These requirements and guidelines are applicable at the time the property is released. Property subject to these criteria includes, but is not limited to sites identified by the Formerly Utilized Sites Remedial Action Program (FUSRAP) and the Surplus Facilities Management Program (SFMP). The topics covered are basic dose limits, guidelines and authorized limits for allowable levels of residual radioactive material, and control of the radioactive wastes and residues. This chapter does not apply to uranium mill tailings or to properties covered by mandatory legal requirements.

2. IMPLEMENTATION. DOE elements shall develop plans and protocols for the implementation of this guidance. FUSRAP sites shall be identified, characterized, and designated, as such, for remedial action and certified for release. Information on applications of the guidelines and requirements presented herein, including procedures for deriving specific property guidelines for allowable levels of residual radioactive material from basic dose limits, is contained in DOE/CH 8901, "A Manual for Implementing Residual Radioactive Material Guidelines, A Supplement to the U.S. Department of Energy Guidelines for Residual Radioactive Material at FUSRAP and SFMP Sites," June 1989.

   a. Residual Radioactive Material This chapter provides guidance on radiation protection of the public and the environment from:

      (1) Residual concentrations of radionuclides in soil (for these purposes, soil is defined as unconsolidated earth material, including rubble and debris that might be present in earth material);

      (2) Concentrations of airborne radon decay products;

      (3) External gamma radiation;

      (4) Surface contamination; and

      (5) Radionuclide concentrations in air or water resulting from or associated with any of the above.

b. Basic Dose Limit. The basic dose limit for doses resulting from exposures to residual radioactive material is a prescribed standard from which limits for quantities that can be monitored and controlled are derived; it is specified in terms of the effective dose equivalent as defined in this Order. The basic dose limits are used for deriving guidelines for residual concentrations of radionuclides in soil. Guidelines for residual concentrations of thorium and radium in soil, concentrations of airborne radon decay products, allowable indoor external gamma radiation levels, and residual surface contamination concentrations are based on existing radiological protection standards (40 CFR Part 192; NRC Regulatory Guide 1.86 and subsequent NRC guidance on residual radioactive material). Derived guidelines or limits based on the basic dose limits for those quantities are used only when the guidelines provided in the existing standards are shown to be inappropriate.

c. Guideline. A guideline for residual radioactive material is a level of radioactive material that is acceptable for use of property without restrictions due to residual radioactive material. Guidelines for residual radioactive material presented herein are of two kinds, generic and specific. The basis for the guidelines is generally a presumed worst-case plausible-use scenario for the property.

(1) Generic guidelines, independent of the property, are taken from existing radiation protection standards. Generic guideline values are presented in this chapter.

(2) Specific property guidelines are derived from basic dose limits using specific property models and data. Procedures and data for deriving specific property guideline values are given by DOE/CH-8901.

d. Authorized Limit. An authorized limit is a level of residual radioactive material that shall not be exceeded if the remedial action is to be considered completed and the property is to be released without restrictions on use due to residual radioactive material.

(1) The authorized limits for a property will include:

(a) Limits for each radionuclide or group of radionuclides, as appropriate, associated with residual radioactive material in soil or in surface contamination of structures and equipment;

(b) Limits for each radionuclide or group of radionuclides, as appropriate, in air or water; and

(c) Where appropriate, a limit on external gamma radiation resulting from the residual material.

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(2) Under normal circumstances expected at most properties, authorized limits for residual radioactive material are set equal to, or below, guideline values. Exceptional conditions for which authorized limits might differ from guideline values are specified in paragraphs IV-5 and IV-7.

(3) A property may be released without restrictions if residual radioactive material does not exceed the authorized limits or approved supplemental limits, as defined in paragraph IV.7a, at the time remedial action is completed. DOE actions in regard to restrictions and controls on use of the property shall be governed by provisions in paragraph IV.7b. The applicable controls and restrictions are specified in paragraph IV.6 and IV.7.c.

e. ALARA Applications. The monitoring, cleanup, and control of residual radioactive material are subject to the ALARA policy of this Order. Applications of ALARA policy shall be documented and filed as a permanent record.

3. BASIC DOSE LIMITS.

a. Defining and Determining Dose Limits. The basic public dose limit for exposure to residual radioactive material, in addition to natural occurring "background" exposures, are 100 mrem (1 mSv) effective dose equivalent in a year, as specified in paragraph II.1a.

b. Unusual Circumstances. If, under unusual circumstances, it is impracticable to meet the basic limit based on realistic exposure scenarios, the respective project and/or program office may, pursuant to paragraph II.1a(4), request from EH-1 for a specific authorization for a temporary dose limit higher than 100 mrem (1 mSv), but not greater than 500 mrem (5 mSv), in a year. Such unusual circumstances may include temporary conditions at a property scheduled for remedial action or following the remedial action. The ALARA process shall apply to the selection of temporary dose limits.

4. GUIDELINES FOR RESIDUAL RADIOACTIVE MATERIAL.

a. Residual Radionuclides in Soil. Generic guidelines for thorium and radium are specified below. Guidelines for residual concentrations of other radionuclides shall be derived from the basic dose limits by means of an environmental pathway analysis using specific property data where available. Procedures for these derivations are given in DOE/CH-8901. Residual concentrations of radioactive material in soil are defined as those in excess of background concentrations averaged over an area of 100 m^2.

(1) Hot Spots. If the average concentration in any surface or below-surface area less than or equal to 25 m^2, exceeds the limit or guideline by a factor of $(100/A)^{0.5}$, [where A is the area (in square meters) of the region in which concentrations are elevated], limits for "hot-spots" shall also be developed and applied. Procedures for calculating these hot-spot limits, which depend on the extent of the elevated local concentrations, are given in DOE/CH-8901. In addition, reasonable efforts shall be made to remove any source of radionuclide that exceeds 30 times the appropriate limit for soil, irrespective of the average concentration in the soil.

(2) Generic Guidelines. The generic guidelines for residual concentrations of Ra-226, Ra-228, Th-230, and Th-232 are:

(a) 5 pCi/g, averaged over the first 15 cm of soil below the surface; and

(b) 15 pCi/g, averaged over 15-cm-thick layers of soil more than 15 cm below the surface.

(3) Ingrowth and Mixtures. These guidelines take into account ingrowth of Ra-226 from Th-230 and of Ra-228 from Th-232, and assume secular equilibrium. If both Th-230 and Ra-226 or both Th-232 and Ra-228 are present and not in secular equilibrium, the appropriate guideline is applied as a limit for the radionuclide with the higher concentration. If other mixtures of radionuclides occur, the concentrations of individual radionuclides shall be reduced so that either the dose for the mixtures will not exceed the basic dose limit or the sum of the ratios of the soil concentration of each radionuclide to the allowable limit for that radionuclide will not exceed 1. Explicit formulas for calculating residual concentration guidelines for mixtures are given in DOE/CH-8901.

b. Airborne Radon Decay Products. Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property that are intended for release without restriction; structures that will be demolished or buried are excluded. The applicable generic guideline (40 CFR Part 192) is: In any occupied or habitable building, the objective of remedial action shall be, and a reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL. [A working level (WL) is any combination of short-lived radon decay products in 1 L of air that will

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result in the ultimate emission of 1.3 x 10^5 MeV of potential alpha energy.] In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions by DOE are not required in order to comply with this guideline when there is reasonable assurance that residual radioactive material is not the source of the radon concentration.

c. External Gamma Radiation. The average level of gamma radiation inside a building or habitable structure on a site to be released without restrictions shall not exceed the background level by more than 20 μR/h and shall comply with the basic dose limit when an "appropriate-use" scenario is considered. This requirement shall not necessarily apply to structures scheduled for demolition or to buried foundations. External gamma radiation levels on open lands shall also comply with the basic limit and the ALARA process, considering appropriate-use scenarios for the area.

d. Surface Contamination. The generic surface contamination guidelines provided in Figure IV-1 are applicable to existing structures and equipment. These guidelines are generally consistent with standards of the NRC (NRC 1982) and functionally equivalent to Section 4, "Decontamination for Release for Unrestricted Use," of Regulatory Guide 1.86, but apply to nonreactor facilities. These limits apply to both interior equipment and building components that are potentially salvageable or recoverable scrap. If a building is demolished, the guidelines in paragraph IV.6a are applicable to the resulting contamination in the ground.

e. Residual Radionuclides in Air and Water. Residual concentrations of radionuclides in air and water shall be controlled to the required levels shown in paragraph II.1a and as required by other applicable Federal and/or State laws.

5. AUTHORIZED LIMITS FOR RESIDUAL RADIOACTIVE MATERIAL.

a. Establishment of Authorized Limits. The authorized limits for each property shall be set equal to the generic or derived guidelines unless it can be established, on the basis of specific property data (including health, safety, practical, programmatic and socioeconomic considerations), that the guidelines are not appropriate for use at the specific property. The authorized limits shall be established to (1) provide that, at a minimum, the basic dose limits of in paragraph IV.3, will not be exceeded under the "worst-case" or "plausible-use" scenarios, consistent with the procedures and guidance provided in DOE/CH-8901, or (2) be consistent with applicable generic guidelines. The authorized limits shall be consistent with limits and guidelines established by other applicable Federal and State laws. The authorized limits are developed through the project offices in the field and are approved by the Headquarters Program Office.

Figure IV-1
Surface Contamination Guidelines

Radionuclides[2/]	Allowable Total Residual Surface Contamination (dpm/100 cm²)[1/] Average[3/,4/]	Maximum[4/,5/]	Removable[4/,6/]
Transuranics, I-125, I-129, Ra-226, Ac-227, Ra-228, Th-228, Th-230, Pa-231.	RESERVED	RESERVED	RESERVED
Th-Natural, Sr-90, I-126, I-131, I-133, Ra-223, Ra-224, U-232, Th-232.	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay product, alpha emitters.	5,000	15,000	1,000
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.[7/]	5,000	15,000	1,000

1/ 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 measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

2/ Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

3/ Measurements of average contamination should not be averaged over an area of more than 1 m². For objects of less surface area, the average should be derived for each such object.

4/ The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

5/ The maximum contamination level applies to an area of not more than 100 cm².

2/ The amount of removable material per 100 cm^2 of surface area should be determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wiping with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

3/ This category of radionuclides includes mixed fission products, including the Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

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b. Application of Authorized Limits. Remedial action shall not be considered complete until the residual radioactive material levels comply with the authorized limits, except as authorized pursuant to paragraph IV.7 for special situations where the supplemental limits and exceptions should be considered and it is demonstrated that it is not appropriate to decontaminate the area to the authorized limit or guideline value.

6. CONTROL OF RESIDUAL RADIOACTIVE MATERIAL. Residual radioactive material above the guidelines shall be managed in accordance with Chapter II and the following requirements.

a. Operational and Control Requirements. The operational and control requirements specified in the following Orders shall apply to interim storage, interim management, and long-term management.

(1) DOE 5000.3, Unusual Occurrence Reporting System

(2) DOE 5440.1C, Implementation of the National Environmental Policy Act

(3) DOE 5480.4, Environmental Protection, Safety, and Health Protection Standards

(4) DOE 5482.1B, Environmental, Safety, and Health Appraisal Program

(5) DOE 5483.1A, Occupational Safety and Health Program for DOE Employees at Government-Owned, Contractor-Operated Facilities

(6) DOE 5484.1, Environmental Protection, Safety, and Health Protection Information Reporting Requirements

(7) DOE 5820.2A, Radioactive Waste Management.

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b. Interim Storage.

(1) Control and stabilization features shall be designed to provide, to the extent reasonably achievable, an effective life of 50 years with a minimum life of at least 25 years.

(2) Controls shall be designed such that Rn-222 concentrations in the atmosphere above facility surfaces or openings in addition to background levels, will not exceed:

(a) 100 pCi/L at any given point;

(b) An annual average concentration of 30 pCi/L over the facility site; and

(c) An annual average concentration of 3 pCi/L at or above any location outside the facility site.

(d) Flux rates from the storage of radon producing wastes shall not exceed 20 pCi/sq.m-sec., as required by 40 CFR Part 61.

(3) Controls shall be designed such that concentrations of radionuclides in the groundwater and quantities of residual radioactive material will not exceed applicable Federal or State standards.

(4) Access to a property and use of onsite material contaminated by residual radioactive material should be controlled through appropriate administrative and physical controls such as those described in 40 CFR Part 192. These control features should be designed to provide, to the extent reasonable, an effective life of at least 25 years.

c. Interim Management.

(1) A property may be maintained under an interim management arrangement when the residual radioactive material exceeds guideline values if the residual radioactive material is in inaccessible locations and would be unreasonably costly to remove. provided that administrative controls are established by the responsible authority (Federal, State, or local) to protect members of the public and that such controls are approved by the appropriate Program Assistant Secretary or Director.

(2) The administrative controls include but are not limited to periodic monitoring as appropriate; appropriate shielding; physical barriers to prevent access; and appropriate radiological safety measures during maintenance, renovation, demolition, or other activities that might disturb the residual radioactive material or cause it to migrate.

(3) The owner of the property should be responsible for implementing the administrative controls and the cognizant Federal, State, or local authorities should be responsible for enforcing them.

d. Long-Term Management.

(1) Uranium, Thorium, and Their Decay Products.

(a) Control and stabilization features shall be designed to provide, to the extent reasonably achievable, an effective life of 1,000 years with a minimum life of at least 200 years.

(b) Control and stabilization features shall be designed to limit Rn-222 emanation to the atmosphere from the wastes to less than an annual average release rate of 20 pCi/m^2/s and prevent increases in the annual average Rn-222 concentration at or above any location outside the boundary of the contaminated area by more than 0.5 pCi/L. Field verification of emanation rates shall be in accordance with the requirements of 40 CFR Part 61.

(c) Before any potentially biodegradable contaminated wastes are placed in a long-term management facility, such wastes shall be properly conditioned so that the generation and escape of biogenic gases will not cause the requirement in paragraph IV.6d(1)(b) to be exceeded and that biodegradation within the facility will not result in premature structural failure in violation of the requirements in paragraph IV.6d(1)(a).

(d) Ground water shall be protected in accordance with legally applicable Federal and State standards.

(e) Access to a property and use of onsite material contaminated by residual radioactive material should be controlled through appropriate administrative and physical controls such as those described in 40 CFR Part 192. These controls should be designed to be effective to the extent reasonable for at least 200 years.

(2) Other Radionuclides. Long-term management of other radionuclides shall be in accordance with Chapters II, III, and IV of DOE 5820.2A, as applicable.

7. SUPPLEMENTAL LIMITS AND EXCEPTIONS. If special specific property circumstances indicate that the guidelines or authorized limits established for a given property are not appropriate for any portion of that property, then the Operations Office may request that supplemental limits or an exception be applied. The responsible Operations Office shall document the decision that the subject guidelines or authorized limits are not appropriate and that the alternative action selected will provide adequate protection,

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giving due consideration to health and safety, the environment, costs, and public policy considerations. The Operations Office shall obtain approval for specific supplemental limits or exceptions from Headquarters as specified in paragraph IV.5, and shall provide to the Headquarters Program Element those materials required by Headquarters for the justification as specified in this paragraph and in the FUSRAP and SFMP protocols and subsequent guidance documents. The Operations Office shall also be responsible for coordination with the State and local government regarding the limits or exceptions and associated restrictions as appropriate. In the case of exceptions, the Operations Office shall be responsible for coordinating with the State and/or local governments to ensure the adequacy of restrictions or conditions of release and that mechanisms are in place for their enforcement.

a. Supplemental Limits. Any supplemental limits shall achieve the basic dose limits set forth in Chapter II of this Order for both current and potential unrestricted uses of a property. Supplemental limits may be applied to any portion of a property if, on the basis of a specific property analysis, it is demonstrated that

(1) Certain aspects of the property were not considered in the development of the established authorized limits for that property; and

(2) As a result of these certain aspects, the established limits either do not provide adequate protection or are unnecessarily restrictive and costly.

b. Exceptions to the authorized limits defined for a property may be applied to any portion of the property when it is established that the authorized limits cannot reasonably be achieved and that restrictions on use of the property are necessary. It shall be demonstrated that the exception is justified and that the restrictions will protect members of the public within the basic dose limits of this Order and will comply with the requirements for control of residual radioactive material as set forth in paragraph IV.6.

c. Justification for Supplemental Limits and Exceptions. The need for supplemental limits and exceptions shall be documented by the Operations Office on a case-by-case basis using specific property data. Every reasonable effort should be made to minimize the use of supplemental limits and exceptions. Examples of specific situations that warrant DOE use of supplemental standards and exceptions are

(1) Where remedial action would pose a clear and present risk of injury to workers or members of the public, notwithstanding reasonable measures to avoid or reduce risk.

(2) Where remedial action, even after all reasonable mitigative measures have been taken, would produce environmental harm that is clearly excessive compared to the health benefits to persons living on or near affected properties, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that may reasonably be anticipated.

(3) Where it is determined that the scenarios or assumptions used to establish the authorized limits do not apply to the property or portion of the property identified, or where more appropriate scenarios or assumptions indicate that other limits are applicable or appropriate for protection of the public and the environment.

(4) Where the cost of remedial action for contaminated soil is unreasonably high relative to long-term benefits and where the residual material does not pose a clear present or future risk after taking necessary control measure. The likelihood that buildings will be erected or that people will spend long periods of time at such a property should be considered in evaluating this risk. Remedial action will generally not be necessary where only minor quantities of residual radioactive material are involved or where residual radioactive material occurs in an inaccessible location at which specific property factors limit its hazard and from which it is difficult or costly to remove. Examples include residual radioactive material under hard-surfaced public roads and sidewalks, around public sewer lines, or in fence-post foundations. A specific property analysis shall be provided to establish that the residual radioactive material would not cause an individual to receive a radiation dose in excess of the basic dose limits stated in paragraph IV.3, and a statement specifying the level of residual radioactive material shall be provided to the appropriate State and/or local agencies for appropriate action, e.g., for inclusion in local land records.

(5) Where there is no feasible remedial action.

8. SOURCES.

a. Basic Dose Limits. Dosimetry model and dose limits are defined in Chapter II of this Order.

b. Generic Guidelines for Residual Radioactive Material. Residual concentrations of radium and thorium in soil are defined in 40 CFR Part 192. Airborne radon decay products are also defined in 40 CFR Part 192, as are guidelines for external gamma radiation. The surface contamination definition is adapted from NRC (1982).

c. Control of Radioactive Wastes and Residues. Interim storage is guided by this Order and DOE 5820.2A. Long-term management is guided by this Order, 40 CFR Part 192, and DOE 5820.2A.

14501-00-DC-01

REV. 1

DESIGN CRITERIA FOR FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM [FUSRAP] AND SURPLUS FACILITIES MANAGEMENT PROGRAM [SFMP]

FEBRUARY 1986

Prepared by
U.S. DEPARTMENT OF ENERGY
OAK RIDGE OPERATIONS OFFICE

14501-00-DC-01-15

Rev. 1

DESIGN CRITERIA
FOR
FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM (FUSRAP)
AND
SURPLUS FACILITIES MANAGEMENT PROGRAM (SFMP)

(ISSUED FOR CLIENT APPROVAL)

FEBRUARY 1986

Approved by: E. L. Keller, Director
Technical Services Division
Oak Ridge Operations Office

2-24-86
Date

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Assistant Manager for
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Date

PREFACE TO DESIGN CRITERIA

These design criteria have been written in a generic form that summarizes criteria applicable for remedial action and long-term management activities associated with the radioactive wastes at the FUSRAP and SFMP sites. Site-specific information is provided in the appendices to this generic document. As a specific scope of work for a site is determined, design bases and work plans for each of the sites will be developed.

Appendix A contains definitions of terms used in these design criteria and referenced documents. Appendix B provides a listing of FUSRAP and SFMP sites by WBS number and contains estimated waste quantities at the sites. Appendix C contains the residual contamination and waste control criteria. Appendix D lists site information for specific sites which will be required as a remedial action for the specific site is developed. This information will be included in the work plan for each site.

The design criteria will be referenced by the designation 14501-00-DC-01.

These design criteria will be periodically revised, as appropriate, to reflect new practices, additional information, revisions of applicable regulations, and standard revisions.

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1.0 INTRODUCTION

1.1 SCOPE

This document defines the design criteria for the identification of materials, evaluation of remedial action alternatives, selection of design parameters for site cleanup remedial actions and interim storage, and long-term management methods for handling FUSRAP and SFMP radioactive wastes.

1.2 OBJECTIVE

The primary objective of the Formerly Utilized Sites Remedial Action Program (FUSRAP) and Surplus Facilities Management Program (SFMP) projects is to stabilize, decontaminate, and/or dispose of FUSRAP and SFMP derived wastes in such a manner as to minimize the radiological risks posed by these wastes and to enable certification of the cleaned up FUSRAP and SFMP sites for unrestricted future use. At some sites, remedial action may be in situ long-term management with monitoring as necessary to detect any contaminant migration from the site in excess of radiological design criteria. At other sites, an interim storage program may be established until a decision for final disposition is made.

1.3 DEFINITIONS

Appendix A contains definitions of terms that are used in these design criteria as well as in the referenced documents.

1.4 CHANGES TO CRITERIA

The criteria for FUSRAP and SFMP remedial actions set forth in this document are based on elements of various federal orders, regulations, and standards that may be subject to change. This document will be revised to reflect changed criteria as authorized and approved by DOE.

2.0 APPLICABLE DOCUMENTS

2.1 GENERAL

The intent of these design criteria is to use DOE Orders where applicable. Applicable orders, regulations and standards, and sections thereof, as well as industry standards, will be investigated on a site-specific basis to formulate the design bases for the specific site.

2.2 FEDERAL ORDERS, REGULATIONS, AND STANDARDS

The following federal orders, regulations, and standards contain elements that are generally applicable to the FUSRAP and SFMP projects, and are summarized for these criteria.

2.2.1 Quality Assurance

DOE Order 5700.6A--Quality Assurance and DOE/OR-FUSRAP-82-001 Plan for Quality Assurance. The Project Quality Assurance Program complies with DOE Order 5700.6A, and the FUSRAP Plan for Quality Assurance (DOE/OR-FUSRAP-82-001).

For each remedial action site, and interconnecting activities (such as transportation), a formal evaluation (Quality Assurance Assessment) will be made of the consequences of failure of equipment and facilities to perform satisfactorily in service. This Assessment, which will be an adjunct to design engineering with subsequent modifications as may be required, will give full consideration to safety, environment, costs, schedule delays, programmatic goals, public reaction, or any other factor important to achieving project objectives.

When the formal evaluation indicates that consequences of failure may be unacceptable, significant, or unknown and the probability of failure is high or unknown, additional deliberate actions to find

and prevent quality problems are mandatory. The additional actions to assure quality of design and engineering, and particularly to assure implementation of that design and engineering, will be documented using a Quality Action Plan.

2.2.2 Radiation Protection

DOE Order 5480.1A. This order establishes control over the environmental protection, safety, and health protection programs. Chapter XI, Requirements for Radiation Protection, Attachment XI-1, defines radiation protection guides for concentration in air and water above natural background which will be used as criteria for releases from DOE's FUSRAP and SFMP operations. Chapter XII, Prevention, Control, and Abatement of Environmental Pollution, provides requirements for the control of sources of environmental pollution in accordance with the substantive and procedural aspects of all applicable federal, state, and local pollution control standards.

DOE Order 5480.2--Hazardous and Radioactive Mixed Waste Management. This order establishes hazardous waste management procedures for facilities operated under authority of the Atomic Energy Act of 1954, as amended (AEA). The procedures will follow, to the extent practicable, regulations issued by the Environmental Protection Agency (EPA) pursuant to the Resource Conservation and Recovery Act of 1976 (RCRA).

DOE Order 5481.1--Safety Analysis and Review System. This DOE Order establishes requirements for the preparation and review of safety analyses for each DOE operation, including: identification of hazards and their elimination or control; assessment of risk; documented management authorization of operation; and transportation of hazardous materials.

2.2.3 Land Disposal of Radioactive Wastes

Elements of the DOE Orders and federal regulations mentioned in the following sections provide technical guidelines for long-term, near-surface land burial facilities and ancillary facilities.

DOE Order 6430.1--General Design Criteria Manual. This order contains basic architectural and engineering design requirements for new DOE facilities; provides technical specification requirements; and outlines planning and design requirements for new facilities, facility additions, facility alterations, and building acquisitions to achieve economy of construction, operation, and maintenance.

40 CFR 192--Standards for Remedial Action at Inactive Uranium Processing Sites. This regulation defines remedial action criteria for inactive uranium processing sites. Some elements of these standards are applicable to the FUSRAP and SFMP programs. Service life of a mill tailings disposal site is defined in this regulation and has been adopted for FUSRAP and SFMP projects. Specific service life and release control requirements for interim storage sites and long-term management sites are noted in Section 3.2 of these Design Criteria.

2.2.4 Handling, Transportation, and Storage

DOE Order 1540.1--Materials Transportation and Traffic Management. Hazardous materials at FUSRAP and SFMP sites shall be shipped in accordance with DOE Order 1540.1. This document outlines DOE's policies and procedures for the management of materials transportation to ensure that it is accomplished in a manner commensurate with:

(1) Operational requirements for transportation services

(2) Established practices and procedures for transportation safety, economy, efficiency, and cargo security

(3) The National Transportation Policy as established by Congress and cognizant federal agencies

(4) Applicable federal, state, local, and international transportation regulations.

Intra-building and intra-site transfers are excluded from the provisions of this order.

DOE Order 5480.1A--Environmental Protection, Safety, and Health Protection Program for DOE Operations. Chapter 3 of this Order contains safety requirements for packaging of fissile and radioactive material. It also defines the requirements for design, evaluation, and testing of containers used for the transport of DOE's fissile and radioactive materials.

49 CFR 171-179--Transportation of Hazardous Materials. These regulations specify requirements for bulk shipments of uranium or thorium ores and physical or chemical concentrations of those ores and uranium metal or natural thorium metal, or alloys of these materials.

2.2.5 Health and Safety

Occupational Safety and Health Administration (OSHA) 29 CFR 1910. This section contains the health and safety regulations for general industry.

Occupational Safety and Health Administration (OSHA) 29 CFR 1926. This section establishes the general health and safety regulations for construction.

2.2.6 Surveys

Surveys for characterization and remedial action will be performed in accordance with the following specifications.

National Oceanic and Atmospheric Administration (NOAA).

- "Classification, Standards of Accuracy, and General Specifications of Geodetic Control Surveys"
- "Specification to Support Classification, Standards of Accuracy, and General Specifications of Geodetic Control Surveys"
- "Manual of Geodetic Triangulation," "Specification Publication No. 247

U.S. Department of Interior (USDI) "Manual of Instructions for the Survey of Public Lands of the United States," 1973, Bulletin 6.

2.2.7 Weather

National Oceanic and Atmospheric Administration. "Comparative Climatic Data for the United States through 1982," 1983.

2.3 STATE AND LOCAL REGULATIONS

State and local regulations governing handling, transportation, and storage of radioactive materials generally follow federal orders and regulations, but may vary depending on whether the particular state is an "Agreement State" under the Atomic Energy Act of 1954, as amended. DOE regulations will be followed, and state and local regulations will be reviewed on a site-specific basis.

2.4 DESIGN CODES, GUIDES, AND STANDARDS

The following industry and national codes, standards, and guides, as applicable, will also serve as guidelines for the Design Criteria for FUSRAP and SFMP:

- American Association of State Highway and Transportation Officials (AASHTO)
- American Concrete Institute (ACI)

- American Conference of Government Industrial Hygienists (ACGIH)
- American Institute of Steel Construction (AISC)
- American National Standards Institute (ANSI)
- American Nuclear Society (ANS)
- American Petroleum Institute (API)
- American Railway Engineering Association (AREA)
- American Society for Testing and Materials (ASTM)
- American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE)
- American Society of Mechanical Engineers (ASME)
- American Water Works Association (AWWA)
- American Welding Society (AWS)
- Institute of Electrical and Electronic Engineers (IEEE)
- Interstate Commerce Commission (ICC)
- Illuminating Engineering Society (IES)
- National Electrical Code (NEC)
- National Electrical Manufacturers' Association (NEMA)
- National Electrical Safety Code (NESC)
- National Fire Protection Association (NFPA) "National Fire Code"
- National Geodetic Survey (NGS)
- National Standard Plumbing Code (NSPC)
- Occupational Safety and Health Standards (OSHA)
- Underwriters' Laboratory (UL)
- Uniform Building Code (UBC)
- U.S. Army Corps of Engineers Dredging Documents
- U.S. Geological Survey (USGS)

3.0 DESIGN REQUIREMENTS

3.1 GENERAL

FUSRAP work may involve remedial action at a number of sites. The currently designated FUSRAP and SFMP sites are listed in Appendix B; waste characteristics and estimated volumes at each site are also given.

Additional sites may be added or deleted with passage of federal legislation; therefore, the list of sites may be subject to revision. The specific type and quantity of contaminated material at each site, as well as geologic, meteorologic, and other site conditions affecting the design and design approach, differ from site to site.

3.2 RADIOLOGICAL DESIGN CRITERIA

The proposed DOE Interim Residual Contamination and Waste Control Guidelines for FUSRAP and SFMP sites are summarized in Appendix C. This criteria should be followed in defining cleanup requirements, developing remedial action plans, and performing and verifying field remedial actions.

3.3 SPECIFIC SITE CONDITIONS

The following information is required for each site and will be completed before or during detailed design and engineering of disposal facilities.

3.3.1 Scope of Work

The Scope of Work for the needed remedial actions must be clearly defined. This may be initiated with the preparation of the Preliminary Engineering Evaluation Report for each site with a

Design Basis, or as a separate document. It will be in accordance with the waste management plan outlined in Section 3.3.4 of these Design Criteria.

3.3.2 State and Local Regulations

In consultation with appropriate DOE-ORO personnel, applicable state and local regulations and ordinances will be reviewed to determine requirements to achieve compliance with health, safety, and environmental regulations. Construction permits and local property access agreements will be obtained as required. Any permits, licenses, or other authorization required by federal, state, or local environmental protection statutes, or any other legal authorizations required by DOE, will be obtained by DOE, Oak Ridge Operations.

3.3.3 Site Information

Define the site conditions for each site as necessary for design decisions. Parameters that may be needed include the following (see Appendix D for detailed requirements):

- Property surveys, easements, and datum
- Water levels
- Precipitation
- Humidity
- Groundwater table
- Frost penetration
- Ice conditions
- Air temperature
- Noise levels
- Winds
- Seismology

- Soil and foundation conditions
- Site historical information (including past and current use; as-built design drawings of buried utilities, structures, and systems; and existing monitoring systems).

3.3.4 Waste Characterization

Complete information on the type, quantity, and existing disposition of the radioactive wastes at any given site will usually be required prior to initiation of the Preliminary Engineering Evaluation Report or detailed design. If data and information in existing reports is not complete, or possibly out of date, additional characterization survey work may be required. Examples of additional characterization, to be planned by Bechtel and approved by DOE on a site specific basis and according to a predetermined need, include the following:

- Location and depth of buried wastes.
- Radiological, physical, and chemical characteristics of wastes in ponds, under surface water, and/or in groundwater.
- Extent of radiological migration, groundwater flow patterns, and seasonal variations.
- Wastes/contamination in building structures that may be scheduled for dismantlement or demolition.

3.3.5 Support Facilities

The identification of the needed temporary and/or permanent support facilities will be made and may include the following:

- Security
- Contamination control
- Structures
- Equipment
- Water treatment and control

- Utilities
- Access routes
- Monitoring system
- Document control
- Administration

3.3.6 Waste Transportation

The following facets for transporting the waste materials will be investigated as applicable:

- Waste form and quantity to be transported
- Mode of transportation
- Packaging and control
- Transportation routes
- Local traffic patterns and impact on community.

APPENDIX A

DEFINITIONS

Abbreviations/Terms	Definitions
AEC	Atomic Energy Commission
alpha particle	A positively charged particle emitted from certain radioactive material. It consists of two protons and two neutrons, hence is identical with the nucleus of the helium atom. It is the least penetrating of common radiation, hence is not dangerous unless alpha-emitting substances have entered the body.
background radiation	Naturally occurring low-level radiation to which all life is exposed. Background radiation levels vary from place to place on the earth.
beta particle	A particle emitted from some atoms undergoing radioactive decay. A negatively charged beta particle is identical to an electron. A positively charged beta particle is called a position. Beta radiation can cause skin damage, and beta emitters are harmful if they enter the body.
BNI	Bechtel National, Inc.

buffer zone	A portion of the land disposal site that is controlled by the licensee and that lies between the disposal unit and the boundary of the site.
CFR	Code of Federal Regulations
Ci	Curie (the unit of radioactivity of any nuclide, which decays at a rate of 3.7×10^{10} disintegrations/second)
contamination	The radioactive substance which is not a portion of the material into and onto which it is now dispersed.
daughter product	The nuclide remaining after a radioactive atom (parent) has undergone radioactive decay. A daughter atom also may be radioactive, producing further daughter products.
decontamination	The removal of radioactive material by chemical or mechanical means from an undesirable location and placement of the removed radioactive material in an acceptable form and location.
dismantlement	The organized manner by which a system or structure is segmented into component pieces which can be managed.

disposal	Isolation of waste from the biosphere with no intent of retrieval in a manner which does not permit easy access to the waste after its emplacement, and does not require perpetual maintenance and monitoring.
disposal site	A portion of a land disposal facility which is used for disposal of waste. It consists of disposal units and a buffer zone.
disposal unit	For near-surface disposal, a "disposal unit" means a discrete portion of the disposal site into which waste is placed for disposal.
DOE	Department of Energy
dpm	Disintegrations per minute
egr	External gamma radiation (gamma radiation emitted from a source(s) external to the body, as opposed to internal gamma radiation emitted from ingested or inhaled sources)
engineered barrier	Man-made structures or devices that are intended to prevent an intruder from inadvertent exposure to radiation from certain waste or to prevent escape of radionuclides to the environment.
EPA	Environmental Protection Agency
exposure	Magnitude of radiation. It is defined and measured in terms of electrical charge produced per unit mass of air.

FUSRAP	Formerly Utilized (MED/AEC) Sites Remedial Action Program
gamma background	Natural gamma ray activity everywhere present, originating from two sources: (1) cosmic radiation bombarding the earth's atmosphere continually, and (2) terrestrial radiation. Whole body absorbed dose equivalent in the U.S. due to natural gamma background ranges from about 60 to 125 mrem/yr.
gamma ray	High energy electromagnetic radiation emitted from the nucleus of a radioactive atom, with specific energies for the atoms of different elements and having high penetrating power.
ground water	Subsurface water in the zone of full saturation.
half-life	The period of time required for one-half of the original amount of a radioisotope to decay into a daughter product.
health effect	An adverse physiological response to environmental pollutants. While physiological responses include sickness, genetic defects, and death, for FUSRAP/SFMP one health effect is defined as one death resulting from cancer caused by exposure to radiation.

hydrogeologic unit	Any soil or rock unit or zone which, by virtue of its porosity or permeability or lack thereof, has a distinct influence on the storage or movement of ground water.
inadvertent intruder	A person who might occupy the disposal site unknowingly after closure and engage in normal activities, such as agriculture, dwelling construction, and other pursuits in which the person might be exposed to radiation from the waste.
interim storage	A short-term disposal having control and stabilization features designed to ensure, to the extent reasonably achievable, an effective life of 50 years and, in any case, at least 25 years at which time ultimate disposal will be made.
intruder barrier	A sufficient depth of cover over the waste that exposure to radiation by an inadvertent intruder will meet the standards for protection against radiation specified in DOE Manual 5820.1 and in 10 CFR 61, or engineered structures that provide equivalent protection to the inadvertent intruder.
land disposal facility	The land, buildings, and equipment which are intended to be used for the disposal of radioactive wastes beneath the surface of the land.

long-term management	A form of ultimate disposal and storage involving near-surface burial of FUSRAP and SFMP radioactive wastes. Includes monitoring and corrective action, as necessary, to ensure that contaminants are not migrating from the site in excess of design criteria, and an institutional control period not less than that specified in 40 CFR 192. Control and stabilization features are designed to ensure to the extent reasonably achievable, an effective life of 1,000 years and, in any case, at least 200 years.
LSA	Low Specific Activity - A class of radioactive material as defined in 49 CFR 173.389(c).
umhos/cm	Micromhos per centimeter (10^{-6} mho/cm)
uR/hr	Microroentgens per hour (10^{-6} R/hr)
mR/hr	Milliroentgens per hour (10^{-3} R/hr)
mrad/hr	Millirads per hour (10^{-3} rad/hr)
MED	Manhattan Engineer District
mho	A unit of electrical conductance, the reciprocal of electrical resistance.
MPC	Maximum permissible concentration as defined per 10 CFR 20.103.

near-surface disposal facility	A land disposal facility in which radioactive waste is disposed within the upper 15-20 meters of the earth's surface.
NEPA	National Environmental Policy Act
NRC	Nuclear Regulatory Commission
nuclide	A general term applicable to all atomic forms of the elements; nuclides comprise all the isotopic forms of all the elements. Nuclides are distinguished by their atomic number, atomic mass, and energy state.
pCi/l	Picocurie per liter (10^{-12} Ci/l)
R	Roentgen (a unit of exposure to ionizing radiation). It is that amount of gamma or x-rays required to produce an electrical charge that is numerically equal to 2.58 x 10^{-4} coulombs/kg.
rad	The basic unit of absorbed dose of ionizing radiation. A dose of one rad means the absorption of 100 ergs of radiation energy per gram of absorbing material.
radioactivity	The spontaneous decay or disintegration of an unstable atomic nucleus, usually accompanied by the emission of ionizing radiation.

radioactive decay chain	A succession of nuclides, each of which transforms by radioactive disintegration into the next, until a stable nuclide results. The first member is called the parent, the intermediate members are called daughters, and the final stable member is called the end product.
radon	A radioactive, chemically inert gas having a half-life of 3.8 days (radium-222); formed as a daughter product of radium (radium-226).
radon background	Low levels of radon gas found in an area due to the presence of uranium or radium in soil and building materials.
radon daughter	One of the several short-lived radioactive daughter products of radon. (Several of the daughters emit alpha particles.)
remedial action	Steps and processes that are undertaken to physically identify, decontaminate, stabilize, or otherwise provide long-term management of radioactive materials to permit certification for unrestricted public use of the area or site.
rdc	Radon daughter concentration (the concentration in air of short-lived radon daughters, usually expressed in pCi/l; also measured in terms of working level (WL).

rem	Roentgen equivalent man. The unit of dose equivalence for all types of ionizing radiation which expresses the effectiveness of the absorbed dose on a common scale. The rem is the basic unit used to record the accumulated dose equivalent to personnel.
site closure and stabilization	Those actions that are taken upon completion of operations that prepare the disposal site for custodial care and that assure that the disposal site will remain stable and will not need ongoing, active maintenance.
SFMP	Surplus Facilities Management Program
surveillance	Observation of the disposal site for purposes of visual detection of need for maintenance, custodial care, evidence of intrusion, and compliance with other license and regulatory requirements.
WL	Working level. A unit of radon daughter exposure, equal to any combination of short-lived radon daughters in 1 liter of air, that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. This level is equivalent to the energy produced in the decay of the daughter products that are present under equilibrium conditions in a liter of air containing 100 pCi of radium-222. It does not include decay of lead-210 (22-year half-life) and subsequent daughter products.

WLM	Working Level Month - An exposure to a one-WL concentration for 170 hours per month.
WBS NO.	Work Breakdown Structure identification sequence number designated by DOE. (See Appendix B for list of identification numbers for the specific sites.)

APPENDIX B

LIST OF FUSRAP AND SFMP SITES

AND

ESTIMATED WASTE QUANTITIES(7/82)

FUSRAP SITES

MRS No.	Site Name	State	Radioisotope	Estimated Volume (yd^3)	Concentration (pCi/g)	Radioactivity (Ci)	References	Remarks
101	Acid/Pueblo	NM	Plutonium-239	390*	110	4.8×10^{-2}	Remedial Action Completed	Soils
102	Albany Research	OR	Natural Uranium Uranium-238 Natural Uranium Radium-226	3,000[b]	a	a	b	Radiological characterization not complete
104	Mayo Canyon	NM	Strontium-90	1,520*	-	-	FB6DU 409-317	No Excavation Required - Marker Placed
105	Chupadera	NM	-	N/A	-	-	Remedial Action Not Required	
108	E. I. duPont	NJ	Uranium-238 Uranium-238 Uranium-238	7,000	1,400 1,100 6,600	2.7 0.5 0.9	DOE/EV-0005/8	Contam. Soils & Drainage Ditch
114	Kellex	NJ	-	175*	-	-		Completed
115	Niagara Falls Vicinity Properties	NY	Radium-226	48,000	a	a	BMI-2045, 2061, 2074	
117	Middlesex Landfill	NJ	Radium-226	33,000	11	1.4	DOE/EV-0005/20	
118	Middlesex Sampling Plant	NJ	Radium-226 Radium-226	- 57,000	- -	3.5 10.5 -	DOE/EV-0005/1	
121	Palos Park	IL	Hydrogen-3	4,030	-	3.0×10^3	DOE/EV-0005/7	
151	St. Louis Airport	MO	Radium-226	86,000	30	3.8	DOE/EV-0005/16	

APPENDIX B (Cont'd)

FUSRAP SITES

HRS No.	Site Name	State	Radioisotope	Estimated Volume (yd^3)	Concentration (pCi/g)	Radioactivity (Ci)	References	Remarks
125	Shpack	MA	Radium-226 Uranium-238	400	109 125	5.5 6.3	ORNL-5799 DOE/EV-0005/31	
126	Universal Cyclops	PA	-	30	-	-	ORO-777	
129	Linde Air Products	NY	Radium-226 Uranium-238 Actinium-227	26,000	9.3 349 2.3	8.0×10^{-2} 2.80 2.0×10^{-2}	DOE/EV-0005/5 FB&DU 409-329	
130	Univ. of Calif.	CA	-	30*	-	-	Completed 9/82	
131	Univ. of Chicago	IL	-	75*	-	-		
134	SLAPSS (Vic. Prop)	MO	-	13,000	-	-		
137	Wayne/Pequannock	NJ	Uranium-238 Thorium-232 Thorium-228 Radium-226	50,000	a	a		
138	Maywood	NJ	Uranium-238 Uranium-235 Uranium-234 Thorium-230 Thorium-232 Thorium-228 Radium-226	210,000	a	a		
139	Colonie	NY	Uranium-238	30,000	a	a		Radiological Characterization Not Complete
140	Hazelwood	MO	Thorium-230 Radium-226 Uranium-238 Uranium-235 Uranium-234	61,000	a	a		Radiological Survey Not Available
			Total Volume FUSRAP	630,650				

APPENDIX B (Cont'd)

SFMP SITES

WBS No.	Site Name	State	Radioisotope	Estimated Volume (yd^3)	Concentration (pCi/g)	Radioactivity (Ci)	References	Remarks
201	Weldon Spring Storage Site	MO						
	o Raffinate Pits		Natural Uranium and Thorium, Radium-226	220,500	-	824	DOE/OR/20722-5	Sludge volume
	o Quarry		Natural Uranium and Thorium, Radium-226	130,800	-	-	-	
	o Vicinity Properties		Natural Uranium Radium-226	102,00	-	-	-	
202	Niagara Falls Storage Site	NY	Radium-226 Uranium-238	218,000	-	940	DOE/OR/20722-1	(Includes contaminated residues, soils and rubble)
		Total Volumes SFMP		671,300				

[a]Actual waste volume

[b]Volumes are from Project Plan

[c]Information is unknown at this time.

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APPENDIX C

U.S. DEPARTMENT OF ENERGY GUIDELINES FOR RESIDUAL RADIOACTIVITY AT FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM AND REMOTE SURPLUS FACILITIES MANAGEMENT PROGRAM SITES

(Rev. 1, July 1985)

A. INTRODUCTION

This document presents U.S. Department of Energy (DOE) radiological protection guidelines for cleanup of residual radioactive materials and management of the resulting wastes and residues. It is applicable to sites identified by the Formerly Utilized Sites Remedial Action Program (FUSRAP) and remote sites identified by the Surplus Facilities Management Program (SFMP).* The topics covered are basic dose limits, guidelines and authorized limits for allowable levels of residual radioactivity, and requirements for control of the radioactive wastes and residues.

Protocols for identification, characterization, and designation of FUSRAP sites for remedial action; for implementation of the remedial action; and for certification of a FUSRAP site for release for unrestricted use are given in a separate document (U.S. Dept. Energy 1984). More detailed information on applications of the guidelines presented herein, including procedures for deriving site-specific guidelines for allowable levels of residual radioactivity from basic dose limits, is contained in a supplementary document--referred to herein as the "supplement" (U.S. Dept. Energy 1985).

"Residual radioactivity" includes: (1) residual concentrations of radionuclides in soil material,** (2) concentrations of airborne radon decay products, (3) external gamma radiation level, and (4) surface contamination. A "basic dose limit" is a prescribed standard from which limits for quantities that can be monitored and controlled are derived; it is specified in terms of the effective dose equivalent as defined by the International Commission on Radiological Protection (ICRP 1977, 1978). Basic dose limits are used explicitly for deriving guidelines for residual concentrations of radionuclides in soil material, except for thorium and radium. Guidelines for

*A remote SFMP site is one that is excess to DOE programmatic needs and is located outside a major operating DOE research and development or production area.

**The term "soil material" refers to all material below grade level after remedial action is completed.

residual concentrations of thorium and radium and for the other three quantities (airborne radon decay products, external gamma radiation level, and surface contamination) are based on existing radiological protection standards (U.S. Environ. Prot. Agency 1983; U.S. Nucl. Reg. Comm. 1982). These standards are assumed to be consistent with basic dose limits within the uncertainty of derivations of levels of residual radioactivity from basic limits.

A "guideline" for residual radioactivity is a level of residual radioactivity that is acceptable if the use of the site is to be unrestricted. Guidelines for residual radioactivity presented herein are of two kinds: (1) generic, site-independent guidelines taken from existing radiation protection standards, and (2) site-specific guidelines derived from basic dose limits using site-specific models and data. Generic guideline values are presented in this document. Procedures and data for deriving site-specific guideline values are given in the supplement.

An "authorized limit" is a level of residual radioactivity that must not be exceeded if the remedial action is to be considered completed. Under normal circumstances, expected to occur at most sites, authorized limits for residual radioactivity are set equal to guideline values. Exceptional conditions for which authorized limits might differ from guideline values are specified in Sections D and F. A site may be released for unrestricted use only if the residual radioactivity does not exceed guideline values at the time remedial action is completed. Restrictions and controls on use of the site must be established and enforced if the residual radioactivity exceeds guideline values. The applicable controls and restrictions are specified in Section E.

DOE policy requires that all exposures to radiation be limited to levels that are as low as reasonably achievable (ALARA). Implementation of ALARA policy is specified as procedures to be applied after authorized limits have been set. For sites to be released for unrestricted use, the intent is to reduce residual radioactivity to levels that are as far below authorized limits as reasonable considering technical, economic, and social factors. At sites where the residual radioactivity is not reduced to levels that permit release for unrestricted use, ALARA policy is implemented by establishing controls to reduce exposure to levels that are as low as is reasonably achievable. Procedures for implementing ALARA policy are described in the supplement. ALARA policies, procedures, and actions must be documented and filed as a permanent record upon completion of remedial action at a site.

B. BASIC DOSE LIMITS

The basic limit for the annual radiation dose received by an individual member of the general public is 500 mrem/yr for a period of exposure not to exceed 5 years and an average of 100 mrem/yr over a lifetime. The committed effective dose equivalent, as defined in ICRP Publication 26 (ICRP 1977) and calculated by dosimetry models described in ICRP Publication 30 (ICRP 1978), shall be used for determining the dose.

C. GUIDELINES FOR RESIDUAL RADIOACTIVITY

C.1 Residual Radionuclides in Soil Material

Residual concentrations of radionuclides in soil material shall be specified as above-background concentrations averaged over an area of 100 m^2. If the concentration in any area is found to exceed the average by a factor greater than 3, guidelines for local concentrations shall also be applicable. These "hot spot" guidelines depend on the extent of the elevated local concentrations and are given in the supplement.

The generic guidelines for residual concentrations of Th-232, Th-230, Ra-228, and Ra-226 are:

- 5 pCi/g, averaged over the first 15 cm of soil below the surface
- 15 pCi/g, averaged over 15-cm-thick layers of soil more than 15 cm below the surface

These guidelines take into account ingrowth of Ra-226 from Th-230 and of Ra-228 from Th-232, and assume secular equilibrium. If either Th-230 and Ra-226 or Th-232 and Ra-228 are both present, not in secular equilibrium, the guidelines apply to the higher concentration. If other mixtures of radionuclides occur, the concentrations of individual radionuclides shall be reduced so that the dose for the mixtures will not exceed the basic dose limit. Explicit formulas for calculating residual concentration guidelines for mixtures are given in the supplement.

The guidelines for residual concentrations in soil material of all other radionuclides shall be derived from basic dose limits by means of an environmental pathway analysis using site-specific data. Procedures for deriving these guidelines are given in the supplement.

C.2 Airborne Radon Decay Products

Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property that are intended for unrestricted use; structures that will be demolished or buried are excluded. The applicable generic guideline (40 CFR 192) is: In any occupied or habitable building, the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL.* In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions are not required in order to comply with this guideline when there is reasonable assurance that residual radioactive materials are not the cause.

C.3 External Gamma Radiation

The average level of gamma radiation inside a building or habitable structure on a site to be released for unrestricted use shall not exceed the background level by more than 20 μR/h.

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*A working level (WL) is any combination of short-lived radon decay products in one liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

C.4 Surface Contamination

The following generic guidelines, adapted from standards of the U.S. Nuclear Regulatory Commission (1982), are applicable only to existing structures and equipment that will not be demolished and buried. They apply to both interior and exterior surfaces. If a building is demolished and buried, the guidelines in Section C.1 are applicable to the resulting contamination in the ground.

Radionuclides†[2]	Allowable Total Residual Surface Contamination (dpm/100 cm²)†[1]		
	Average†[3],†[4]	Maximum†[4],†[5]	Removable†[4],†[6]
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000α	15,000α	1,000α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5,000β-γ	15,000β-γ	1,000β-γ

†[1] 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 measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

†[2] Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

†[3] Measurements of average contamination should not be averaged over an area of more than 1 m². For objects of less surface area, the average should be derived for each such object.

†[4] The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

†[5] The maximum contamination level applies to an area of not more than 100 cm².

†[6] The amount of removable radioactive material per 100 cm² of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm² is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. The numbers in this column are maximum amounts.

D. AUTHORIZED LIMITS FOR RESIDUAL RADIOACTIVITY

The remedial action shall not be considered complete unless the residual radioactivity is below authorized limits. Authorized limits shall be set equal to guidelines for residual radioactivity unless: (1) exceptions specified in Section F of this document are applicable, in which case an authorized limit may be set above the guideline value for the specific location or condition to which the exception is applicable; or (2) on the basis of site-specific data not used in establishing the guidelines, it can be clearly established that limits below the guidelines are reasonable and can be achieved without appreciable increase in cost of the remedial action. Authorized limits that differ from guidelines must be justified and established on a site-specific basis, with documentation that must be filed as a permanent record upon completion of remedial action at a site. Authorized limits differing from the guidelines must be approved by the Director, Oak Ridge Technical Services Division, for FUSRAP and by the Director, Richland Surplus Facilities Management Program Office, for remote SFMP--with concurrence by the Director of Remedial Action Projects for both programs.

E. CONTROL OF RESIDUAL RADIOACTIVITY AT FUSRAP AND REMOTE SFMP SITES

Residual radioactivity above the guidelines at FUSRAP and remote SFMP sites must be managed in accordance with applicable DOE Orders. The DOE Order 5480.1A requires compliance with applicable federal, state, and local environmental protection standards.

The operational and control requirements specified in the following DOE Orders shall apply to interim storage, interim management, and long-term management.

a. 5440.1B, Implementation of the National Environmental Policy Act

b. 5480.1A, Environmental Protection, Safety, and Health Protection Program for DOE Operations

c. 5480.2, Hazardous and Radioactive Mixed Waste Management

d. 5480.4, Environmental Protection, Safety, and Health Protection Standards

e. 5482.1A, Environmental, Safety, and Health Appraisal Program

f. 5483.1, Occupational Safety and Health Program for Government-Owned Contractor-Operated Facilities

g. 5484.1, Environmental Protection, Safety, and Health Protection Information Reporting Requirements

h. 5484.2, Unusual Occurrence Reporting System

i. 5820.2, Radioactive Waste Management

E.1 Interim Storage

a. Control and stabilization features shall be designed to ensure, to the extent reasonably achievable, an effective life of 50 years and, in any case, at least 25 years.

b. Above-Background Rn-222 concentrations in the atmosphere above facility surfaces or openings shall not exceed: (1) 100 pCi/L at any given point, (2) an annual average concentration of 30 pCi/L over the facility site, and (3) an annual average concentration of 3 pCi/L at or above any location outside the facility site (DOE Order 5480.1A, Attachment XI-1).

c. Concentrations of radionuclides in the groundwater or quantities of residual radioactive materials shall not exceed existing federal, state, or local standards.

d. Access to a site shall be controlled and misuse of onsite material contaminated by residual radioactivity shall be prevented through appropriate administrative controls and physical barriers--active and passive controls as described by the U.S. Environmental Protection Agency (1983--p. 595). These control features should be designed to ensure, to the extent reasonable, an effective life of at least 25 years. The federal government shall have title to the property.

E.2 Interim Management

a. A site may be released under interim management when the residual radioactivity exceeds guideline values if the residual radioactivity is in inaccessible locations and would be unreasonably costly to remove, provided that administrative controls are established to ensure that no member of the public shall receive a radiation dose exceeding the basic dose limit.

b. The administrative controls, as approved by DOE, shall include but not be limited to periodic monitoring, appropriate shielding, physical barriers to prevent access, and appropriate radiological safety measures during maintenance, renovation, demolition, or other activities that might disturb the residual radioactivity or cause it to migrate.

c. The owner of the site or appropriate federal, state, or local authorities shall be responsible for enforcing the administrative controls.

E.3 Long-Term Management

Uranium, Thorium, and Their Decay Products

a. Control and stabilization features shall be designed to ensure, to the extent reasonably achievable, an effective life of 1,000 years and, in any case, at least 200 years.

b. Control and stabilization features shall be designed to ensure that Rn-222 emanation to the atmosphere from the waste shall not: (1) exceed an annual average release rate of 20 pCi/m^2/s, and (2) increase the annual average Rn-222 concentration at or above any location outside the boundary of the contaminated area by more than 0.5 pCi/L. Field verification of emanation rates is not required.

c. Prior to placement of any potentially biodegradable contaminated wastes in a long-term management facility, such wastes shall be properly conditioned to ensure that (1) the generation and escape of biogenic gases will not cause the requirement in paragraph b of this section (E.3) to be exceeded, and (2) biodegradation within the facility will not result in premature structural failure in violation of the requirements in paragraph a of this section (E.3).

d. Groundwater shall be protected in accordance with 40 CFR 192.20(a)(2) and 192.20(a)(3), as applicable to FUSRAP and remote SFMP sites.

e. Access to a site should be controlled and misuse of onsite material contaminated by residual radioactivity should be prevented through appropriate administrative controls and physical barriers--active and passive controls as described by the U.S. Environmental Protection Agency (1983--p. 595). These controls should be designed to be effective to the extent reasonable for at least 200 years. The federal government shall have title to the property.

Other Radionuclides

f. Long-term management of other radionuclides shall be in accordance with Chapters 2, 3, and 5 of DOE Order 5820.2, as applicable.

F. EXCEPTIONS

Exceptions to the requirement that authorized limits be set equal to the guidelines may be made on the basis of an analysis of site-specific aspects of a designated site that were not taken into account in deriving the guidelines. Exceptions require approvals as stated in Section D. Specific situations that warrant exceptions are:

a. Where remedial actions would pose a clear and present risk of injury to workers or members of the general public, notwithstanding reasonable measures to avoid or reduce risk.

b. Where remedial actions--even after all reasonable mitigative measures have been taken--would produce environmental harm that is clearly excessive compared to the health benefits to persons living on or near affected sites, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that may reasonably be anticipated.

c. Where the cost of remedial actions for contaminated soil is unreasonably high relative to long-term benefits and where the residual radioactive materials do not pose a clear present or future risk after taking necessary control measures. The likelihood that buildings will be erected or that people will spend long periods of time at such a site should be considered in evaluating this risk. Remedial actions will generally not

be necessary where only minor quantities of residual radioactive materials are involved or where residual radioactive materials occur in an inaccessible location at which site-specific factors limit their hazard and from which they are costly or difficult to remove. Examples are residual radioactive materials under hard-surface public roads and sidewalks, around public sewer lines, or in fence-post foundations. In order to invoke this exception, a site-specific analysis must be provided to establish that it would not cause an individual to receive a radiation dose in excess of the basic dose limits stated in Section B, and a statement specifying the residual radioactivity must be included in the appropriate state and local records.

d. Where the cost of cleanup of a contaminated building is clearly unreasonably high relative to the benefits. Factors that shall be included in this judgment are the anticipated period of occupancy, the incremental radiation level that would be effected by remedial action, the residual useful lifetime of the building, the potential for future construction at the site, and the applicability of remedial actions that would be less costly than removal of the residual radioactive materials. A statement specifying the residual radioactivity must be included in the appropriate state and local records.

e. Where there is no feasible remedial action.

G. SOURCES

Limit or Guideline	Source
Basic Dose Limits	
Dosimetry Model and Dose Limits	International Commission on Radiological Protection (1977, 1978)
Generic Guidelines for Residual Radioactivity	
Residual Concentrations of Radium and Thorium in Soil Material	40 CFR 192
Airborne Radon Decay Products	40 CFR 192
External Gamma Radiation	40 CFR 192
Surface Contamination	Adapted from U.S. Nuclear Regulatory Commission (1982)
Control of Radioactive Wastes and Residues	
Interim Storage	DOE Order 5480.1A
Long-Term Management	DOE Order 5480.1A; 40 CFR 192

H. REFERENCES

International Commission on Radiological Protection. 1977. Recommendations of the International Commission on Radiological Protection (Adopted January 17, 1977). ICRP Publication 26. Pergamon Press, Oxford. [As modified by "Statement from the 1978 Stockholm Meeting of the ICRP." Annals of the ICRP, Vol. 2, No. 1, 1978.]

International Commission on Radiological Protection. 1978. Limits for Intakes of Radionuclides by Workers. A Report of Committee 2 of the International Commission on Radiological Protection. Adopted by the Commission in July 1978. ICRP Publication 30. Part 1 (and Supplement), Part 2 (and Supplement), Part 3 (and Supplements A and B), and Index. Pergamon Press, Oxford.

U.S. Environmental Protection Agency. 1983. Standards for Remedial Actions at Inactive Uranium Processing Sites; Final Rule (40 CFR Part 192). Fed. Regist. 48(3):590-604 (January 5, 1983).

U.S. Department of Energy. 1984. Formerly Utilized Sites Remedial Action Program. Summary Protocol: Identification - Characterization - Designation - Remedial Action - Certification. Office of Nuclear Energy, Office of Terminal Waste Disposal and Remedial Action, Division of Remedial Action Projects. April 1984.

U.S. Department of Energy. 1985. Supplement to U.S. Department of Energy Guidelines for Residual Radioactivity at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites. A Manual for Implementing Residual Radioactivity Guidelines. Prepared by Argonne National Laboratory, Los Alamos National Laboratory, Oak Ridge National Laboratory, and Pacific Northwest Laboratory for the U.S. Department of Energy. (In preparation.)

U.S. Nuclear Regulatory Commission. 1982. Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material. Division of Fuel Cycle and Material Safety, Washington, DC. July 1982.

APPENDIX D

SITE INFORMATION FOR SPECIFIC SITES
(See Design Criteria, Section 3.3.3)

1.0 GENERAL

This appendix is a general outline of the information that will be obtained for a FUSRAP/SFMP site through historical research and/or field investigation activities during site characterization. This information will be used as a starting point for preparation of Design Bases for the sites. The data unique to a particular site are enclosed between single asterisks (*..*).

2.0 SURVEYS AND DATUM

Information on site description, surveys, plant coordinates, plant datum, plant grade, horizontal and vertical survey control points, plant grid north, site boundary, access roads, railroads, etc., will be obtained.

3.0 WATER LEVELS

For sites located on rivers, lakes, or at the ocean, the probable maximum and minimum water levels and their fluctuations will be obtained. The design maximum flood elevations, as noted below, will be investigated and recorded for the site:

	Elevation Above Mean Sea Level (MSL) (*..*)
Maximum recorded high water	ft
100-year projected flood	ft
Probable maximum flood	ft
Maximum projected water level for plant safety	ft
Design high water	ft
Design low water	ft

(In general, the 100-year flood shall be used for design.)

4.0 PRECIPITATION (*..*)

Rainfall

Average annual	in.
Daily maximum	in.
Design hourly maximum (100-year storm)	in.
Probable maximum precipitation (PMP) per hour	in.

Flash floods caused by thunderstorm may occur and are to be considered in the design. (Note value to be used in flood design as *..* in. per hour.)

SNOWFALL (*..*)

Average annual	in.
Season maximum	in.
Maximum for month of *..*	in.
Daily maximum	in.
Design snow load	lb/sq. ft.

5.0 GROUNDWATER TABLE

The high water table to be used in design will be stated.

For the design of all underground structures, the high water table will be assumed as elevation *..* ft.

Average groundwater level is approximately at *..* ft.

6.0 FROST PENETRATION

Depth below grade *..* in.

7.0 ICE

If applicable, ice pack formation will be described giving appropriate design loads.

8.0 AIR TEMPERATURE (*..*)

Maximum design	°F
Minimum design	°F
Average annual	°F
Average wet bulb	°F
Average dry bulb	°F

9.0 NOISE LEVELS

Noise level measurement and monitoring during construction will be maintained for sites as required by local authorities.

10.0 WINDS

Based on 100-year recurrence interval, the design wind velocity shall be *..* mph at *..* feet above grade in accordance with the Uniform Building Code (UBC). The prevailing wind is in *..* direction. Wind velocity will be adjusted as appropriate for structure height and gust factors. The effects of tornadoes will be investigated as required by site conditions.

11.0 SEISMOLOGY

The site is in UBC Zone *..*. Seismic loads shall be considered in accordance with Section 2312 of UBC criteria.

Verification of whether a higher zoning than that required by UBC may be more appropriate for the particular site will be made.

12.0 GEOTECHNICAL INVESTIGATIONS

Subsurface investigations will provide a description of the soil and geological and hydrological conditions and other data for the preparation of "Soil and Geological Investigation Report". The design basis will list from the report the hydraulic gradient of ground water, soil profile, location of bedrock, determination of confined and unconfined aquifers, establishment of monitoring wells, test results of soil and rock properties, allowable bearing and/or pile capacities (as applicable) for foundation design, active and passive lateral earth pressure, etc. Compaction criteria and maximum slopes for excavation will also be specified.

13.0 GUIDELINES FOR RESIDUAL RADIOACTIVITY

To be developed for each site. Refer to Appendix C.

DOE F 1325.8
(08-93)

132753

United States Government

Department of Energy

memorandum

DATE: JUL 27 1995

JUL 31 1 41 PM '95

REPLY TO ATTN OF: EM-421 (J. Wagoner, 903-8147)

SUBJECT: Supplemental Limits for Residual Uranium at the Chapman Valve Site, Indian Orchard, Massachusetts

TO: L. Price, Director
Former Sites Restoration Division, OR

This is to approve the general approach to remediation of the Chapman Valve Site in Indian Orchard, Massachusetts, which contemplates the use of Supplemental Limits in the decontamination of this site.

Proposal:

The site contains residual uranium from uranium metal machining activities in the late 1940s. Cleanup performed at the end of the production removed most uranium from the floor and from the walls at floor level. Residual uranium remains on a number of overhead, horizontal surfaces, such as roof support trusses and walls. The residual uranium is concentrated in the west end of the building. Residual uranium is also present in the roof, where it is imbedded in the asphaltic roof material and on roof planks.

The proposed remediation approach would be as follows:

- Remove all detectable contamination at ground level to the surface limits established in DOE Order 5400.5, Chapter IV;
- Perform gross decontamination of all horizontal surfaces above the ground level; and
- Perform decontamination on all horizontal truss surfaces to meet an average surface level of no more than 15,000 dpm per 100 cm^2 of uranium activity for the first eight trusses.

Independent verification is proposed for the floor area and the trusses. In addition, after the property owner demolishes the building, the independent verification contractor would verify the building footprint to confirm the remediated status of the premises.

This proposal would leave residual uranium in the roof, wall, and, possibly, on roof support trusses. Residual uranium in the roof is fixed in place by the roof material, which also contains asbestos. The residual uranium in the wall is fixed and, in general, averages well below the applicable limits in DOE 5400.5, Chapter IV. The horizontal surfaces of the first eight roof support trusses in the west end of the building would be cleaned to meet an average of 15,000 dpm per 100cm^2.

Printed on recycled paper

Regulatory Issues:

The total inventory of uranium at the site (before remediation) is less than 100 pounds, the reportable quantity (RQ) established by the U.S. Environmental Protection Agency at 40 CFR 302.2. A higher RQ has also been established based on radioactivity (See 40 CFR 302.4, Appendix B). However, the higher limit is superseded by the lower one (See footnote "&" to Appendix B of 40 CFR 302.4).

In addition, using the proposed remediation approach, any uranium remaining at the site would be less than the general license quantity established by the U.S. Nuclear Regulatory Commission at 10 CFR 40.22.

As a result, the proposed approach is generally consistent with the *de minimis* regulatory limits established by other agencies regulating radioactive materials.

Cost Benefit Issues:

Your staff has provided dose and cost estimates for additional decontamination of the facility to meet the specific requirements of DOE Order 5400.5, Chapter IV. Specific compliance would involve significant incremental costs in excess of $500,000, with avoided dose to a hypothetical maximum exposed dose of 19 millirem per year. The cost of additional cleanup is not justified, based on the analysis furnished. In the event of new information or changed circumstances, Departmental policy permits you, your staff, and your supporting contractors to conduct additional decontamination to achieve radiation exposures "as low as reasonably achievable."

Approval:

This is to approve the use of supplemental limits as a "gross decontamination" approach to cleanup of this site, pursuant to DOE Order 5400.5, Paragraphs IV.7 and IV.5. The approval is based on the small uranium inventory present at the site, the high cost of additional remediation, and the insignificant dose reduction that would be achieved from additional remediation.

Please prepare a formal hazard assessment to document the calculations and recommendations concerning the supplemental limits. The hazard assessment will be used as a basis to document the approach and to ratify and confirm the supplemental limits.

Please call me if you have any questions.

James W. Wagoner II
Director
Off-Site/Savannah River Program Division
Office of Eastern Area Programs
Office of Environmental Restoration

cc: J. Kopotic, OR
R. Rodriguez, ORNL

DOE F 1325.8
(8-89)
EFG (07-90)

144918

United States Government **Department of Energy**

memorandum

DATE: JUL 31 1996 AUG 5 1 53 PM '96

REPLY TO ATTN OF: EM-42 (W. Williams, 301-903-8149)

SUBJECT: Supplemental Limits for Residual Uranium at the Chapman Valve Site, Indian Orchard, Massachusetts

TO: L. Price, OR

This is to ratify and confirm the prior approval of Supplemental Limits in the cleanup of the Chapman Valve Site in Indian Orchard, Massachusetts.

Background:

The general approach to the cleanup of this site and the contemplated use of Supplemental Standards were approved by memorandum dated July 27, 1995. The site was decontaminated using an approved "gross decontamination" approach which focused the cleanup on the building areas with the highest levels of residual uranium. In this approach, it was intended and expected that residual uranium would remain in the roof, upper wall, and, possibly, on the roof support trusses.

A "Formerly Utilized Sites Remedial Action Program Technical Memorandum" (CCN: 143419) has been prepared to reflect the cleanup in an "as accomplished" basis and to reflect the actual cleanup levels achieved. This document has been prepared in lieu of a hazard assessment and is adequate for approving Supplemental Limits. The calculations indicate that all potential radiation doses are well below the applicable dose limits in Department of Energy Order 5400.5, Chapter IV. In the unlikely event of reuse of building materials, the radiation doses from residual materials are a small fraction of the dose from naturally occurring radionuclides in the building materials.

It is anticipated that the technical memorandum will be published as the Post Remedial Action Report (PRAR), after inclusion of comments.

Approval:

We recommend that the technical memorandum be published as the PRAR after resolution of comments.

A number of regulatory, cost-benefit, and technical issues were addressed in the July 27, 1995, memorandum. The technical memorandum demonstrates

that remedial action was accomplished as intended. Accordingly, the use of Supplemental Standards for this site is ratified and confirmed.

Please call me if you have any questions.

Albert S. Johnson
Team Leader
FUSRAP Team
Office of Eastern Area Programs
Office of Environmental Restoration

cc: S. Oldham, DOE/OR
J. Kopotic, DOE/OR
R. Rodriguez, Oak Ridge National Laboratory

143419

Bechtel

Oak Ridge Corporate Center
151 Lafayette Drive
P.O. Box 350
Oak Ridge, Tennessee 37831-0350

Telephone: (423) 220-2000

Job No. 14501, FUSRAP Project
DOE Contract No. DE-AC05-91OR21949
Code: 7330/WBS: 133

JUN 20 1996

U.S. Department of Energy
Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, TN 37831-8723

Attention: James D. Kopotic, Site Manager
Former Sites Restoration Division

Subject: Chapman Valve Site - Technical Memorandum Post Remedial Action Report

Dear Mr. Kopotic:

Enclosed is a copy of the subject document. All comments received from ANL and FSRD in the package transmitted by Steve Oldham to me on June 6, 1996 (CCN 143043) have been addressed; a comment resolution package is enclosed.

In order to expedite this review, the document is being transmitted at this time as a Technical Memorandum. However, based on the amount of review and comment that the document has already received and the fact the document follows the annotated outline for a Post Remedial Action Report (PRAR), we recommend the document be published as a PRAR following incorporation of any comments you have on this version. This would eliminate any misconceptions the public might have concerning the level of effort that went into the Chapman Valve cleanup. During your review of this document, we will be putting it in the PRAR document format.

This document and all attachments were prepared under my direction or supervision in accordance with a system designed to ensure that the information submitted was properly gathered and evaluated. To the best of my knowledge and belief, they are true, accurate, and complete.

143419

If you have any questions, please call me at 576-4718 or Bob Robbins at 576-4886.

Sincerely,

Karen C. Noey

for M. E. Redmon
Project Manager - FUSRAP

NJ_0896.Doc

Enclosure: Technical Memorandum Post Remedial Action Report - Chapman Valve

Concurrence: T.M. King @ TMK M. A. Kucera MAK

ACTION REQ'D X YES ☐ NO DUE DATE 6/28/96
RESPONSE TO CHRON NO.
☐ FFA ☐ Permit ☐ Milestone ☐ OcR ☐ CCN ☐ CAR ☐ Mid-Yr ☐ Yr-End ☐ Periodic Rpt

143419

FUSRAP Project
Job 14501

NO. _____ REV. NO. 0
DATE: _____

FUSRAP TECHNICAL MEMORANDUM

TO: Mike Redmon

FROM: Robert Robbins

DATE: 13 May, 1996

SUBJECT: Chapman Valve Final Remedial Action Report

Prepared by	Team Lead Approval	Project Engineer Approval	Project Manager Approval
Tom Kay for Bob Robbins	Melissa A Kucera	[illegible]	M Redmon

Background

This technical memorandum documents the remedial action conducted at the Chapman Valve Site (CHV) in Indian Orchard, Massachusetts from July to September 1995 (Figure 1). The remedial action at CHV was an efficient, cost-effective, and environmentally acceptable approach for this small site; the approach, which followed the expedited protocols, complied with state and local regulations.

Remedial activities at the CHV were performed as part of the U.S. Department of Energy's (DOE) Formerly Utilized Sites Remedial Action Program (FUSRAP). FUSRAP was established to identify and clean up, or otherwise control, sites where residual radioactive contamination remains from the early years of the nation's atomic energy program or from commercial operations causing conditions that Congress has authorized DOE to remedy. FUSRAP was established in 1974 and currently includes 46 sites in 14 states. The CHV was designated for remedial action under FUSRAP in 1992.

FUSRAP objectives for the CHV were to

- remove or otherwise control radioactive contamination above current DOE guidelines, and
- achieve and maintain compliance with applicable criteria for the protection of human health and the environment.

Bechtel National, Inc. (BNI), the project management contractor (PMC), assisted DOE's Oak Ridge Operations Office in the planning, management, and implementation of the cleanup of the CHV. DOE Headquarters utilized Oak Ridge National Laboratory (ORNL) as an independent verification contractor (IVC) to verify that the remedial action met the cleanup criteria established for the site.

History

The CHV is located in Indian Orchard, a suburb of Springfield, Massachusetts at 203 Hampshire Street. CHV was formerly owned and operated by the Chapman Valve Manufacturing Company. Since 1981, The Crane Company has owned and occupied the site. The Crane Company vacated the building in 1987. In 1991, the building was still standing and in reasonably good condition; however, harsh winter conditions have resulted in the deterioration of the building and a structural inspection performed by Alderman and MacNeish Architects and Engineers of Springfield, MA (Ref. 1) indicated the roof was unsafe. During remedial action areas under failing roof sections were identified and appropriate precautions taken (i.e., posting, barricading, limiting access, etc.) to ensure the safety of workers.

In 1948, Chapman Valve set aside approximately one-third of Department 40 in the western end of building 23 (see Figure 2) for the machining of uranium rods for the Atomic Energy Commission's (AEC) Brookhaven National Laboratory. Uranium operations were terminated on November 8, 1948, in a letter that requested termination of AEC film badge services. After the contract was completed, November 1948, Chapman Valve had in its possession over 27,000 pounds of metal scrap, oxides, and sweepings. This material was identified and removed several months after the contract was complete.

Extent of Contamination

ORNL personnel indicated in a 1992 survey report (Ref. 2) that the residual uranium contamination found at the CHV was typical of Manhattan Engineering District/AEC operations. The contamination was limited to interior areas and included floors, walls, and overhead beams. Following a review of files it was concluded there are no indications that work with uranium metal was conducted at the site after the AEC operations were terminated.

The PMC performed additional radiological surveys in November and December 1994 to supplement and refine existing survey information (Ref. 3). Characterization findings indicated the presence of contamination in the western end of building 23 at CHV (see Figures 2, 3, 4, and 5). These findings are in agreement with historical process information obtained during interviews conducted with a former Chapman Valve supervisor. According to the supervisor, a temporary wall was constructed across the center portion of the building from floor to ceiling between columns A-7 and B-7, this is also the location in the building where contamination on the walls begins to decrease significantly.

Additional contamination was identified where a chip burner was located in the southwest corner of Grid A-1 that exhausted to the atmosphere out a nearby window. The exhaust location and the shape of the roof of the building would lead to the deposition of more contamination on the south roof than the north roof as indicated by characterization measurements. The uranium storage area was located in room B4 (south side of building between columns B4 and B7). Characterization results did not indicate any subsurface soil contamination below the base slab of the floor.

Levels of contamination decreased from west to east in the building and survey and sampling results from the east end indicated near background radiological conditions. The exterior of the building except for the roof and several locations on the exterior west and south walls indicated near background radioactivity.

Chemical sampling results indicated the presence of lead paint on the cranes, electrical boxes, and structural steel. Bulk asbestos sampling was positive for the composite roof material and electrical box insulation.

Results from decontamination tests indicated that the majority of the contamination on the walls was removed with the paint. Contamination on the trusses was significantly decreased after light brushing.

Remedial Action Guidelines

Radioactive contamination at the CHV consisted primarily of natural uranium. Table 1 lists the DOE residual contamination guidelines for release of formerly contaminated properties for use without radiological restrictions. The guidelines listed in Table 1 were applied to the crane, floor, and drainlines. These guidelines were adopted by DOE based on their compatibility with U.S. Environmental Protection Agency (EPA) criteria for remedial action found in 40 CFR 192, "Uranium Mill Tailings

Remedial Action Program" (Ref. 4) and are contained in DOE Order 5400.5, "Radiation Protection of the Public and the Environment" (Ref. 5).

Supplemental guidelines were also developed based on information contained in a technical study and preliminary hazard assessment (Ref. 6). A summary of final hazard assessment calculations is included at the end of this technical memorandum utilizing the post-remedial action survey data. Supplemental limits were approved (Ref. 7) for the horizontal truss surfaces that were decontaminated in the first eight bays. The limit for fixed surface contamination was 15,000 dpm/100 cm^2 averaged over the truss and 1,000 dpm/100 cm^2 removable surface contamination. This limit applies to both alpha and beta/gamma contamination.

Remedial Action Design

The approach to remedial action at CHV was determined by a multi-disciplinary core team. The core team looked at all potential decontamination techniques and determined the technique that was the most technically feasible and implementable. Because of potential salvage and reuse of the crane and the possibility that the base slab would remain intact following future building demolition it was determined the crane and floor would be decontaminated to DOE residual contamination guidelines listed in Table 1. A supplemental guideline of 15,000 dpm/100 cm^2 average for the horizontal surfaces of an entire truss was determined to be protective of a future demolition worker and that the dose contribution of contamination on the walls and roof were negligible to the same demolition worker. Due to this no decontamination of the wall and roof surfaces was necessary; however, to comply with DOE's As Low As Reasonably Achievable (ALARA) policy it was decided to decontaminate horizontal wall surfaces of removable contamination. The approach to remedial action was approved by the Massachusetts Department of Environmental Protection (Ref. 8) and the Massachusetts Department of Public Health (Ref. 9).

Decontamination Activities

Decontamination activities at CHV lasted approximately 8 weeks from July to September 1995. All remediation efforts were confined to the interior of building 23. Designation and characterization surveys revealed contamination on interior building surfaces in the western third of building 23 from column 1 to column 8 and on a bridge crane located in grid A-23 (Figures 3, 4, and 5). During floor removal activities it was discovered that contamination above criteria existed under a concrete ramp just inside the west equipment door and in drain lines located in room B4. Techniques used in the remedial action are summarized in Table 2.

Volume reduction, waste minimization, and cost savings techniques employed during the remedial action included segregation, sampling, and surveying of the wastes produced. Washable protective clothing and a personnel contamination monitor were also used as cost savings and waste minimization techniques. Total costs for remedial action are shown in Table 3. The following are specific examples of waste minimization and cost savings measures at CHV:

- Concrete and debris were surveyed to determine if contamination levels were above criteria. The material below criteria was placed in a clean pile in the building and, with property owner concurrence, was left onsite.

- Paint removed from the crane was treated by acid dissolution and solidification with concrete to reduce the content of leachable lead to the extent that TCLP results were below RCRA limits. The concrete monoliths created following acid treatment were determined to be below radiological guidelines and with concurrence of the property owner and the State of Massachusetts were left onsite.

- Interstate Nuclear Services was contracted to provide laundry services for washable protective clothing. The Protech-2000 coveralls used provided for cost savings associated with recycling and reuse of protective clothing and also reduced waste volumes by eliminating the disposal of standard Tyvek protective clothing. An additional added benefit to the Protech-2000 was its ability to breath and the comfort level of the workers as opposed to Tyvek.

- An Eberline PCM-2 personnel contamination monitor was utilized at the access control point instead of the standard manual frisk. The time saved when considering the standard hand frisk takes approximately five minutes when compared to the one minute time required for the PCM-2 was substantial over the life of the remedial action project at CHV. The PCM-2 also assured a consistent exit survey over the conventional hand frisk being performed by the individual.

The following sections contain descriptions of decontamination techniques for each area remediated.

10-Ton Bridge Crane

The bridge crane located at grid A23 (Figure 2) was decontaminated of residual surface contamination utilizing a hand scraping technique with putty knifes and vacuuming. The crane was successfully decontaminated without disturbing painted surfaces except in several small areas requiring paint removal to achieve a successful decontamination. Prior to paint removal the HEPA vacuums were emptied to minimize the amount of lead paint requiring treatment. Very small quantity generator status permitted treatment and disposal onsite following successful treatment. The paint was dissolved in an acid solution prior to being mixed with concrete. Following the treatment of the paint the concrete monoliths were surveyed and determined to be below DOE residual surface contamination guidelines listed in Table 1 and were left onsite at the concurrence of the property owner and the State of Massachusetts. The wooden decking of the crane was removed and decontaminated using the same technique that was used on the crane. The decontaminated boards were also left onsite.

Overhead Trusses

The upper horizontal surfaces of trusses in the main bay of building 23 from truss 1 to truss 8 were decontaminated by vacuuming using HEPA vacuums to remove contaminated dust. In one area of truss 2 abrasive decontamination was required to meet the supplemental limits set for CHV. The radioactive dust generated from the operation was set in concrete to eliminate the potential for fugitive dust during transport to and waste handling at the disposal facility. The concrete/dust was then placed in an intermodal container for disposal at Envirocare of Utah.

Horizontal Wall Surfaces

Horizontal surfaces (crane rail, window sills, radiators, and a large pipe on the north wall) on the walls of the main bay from column 1 to column 8 were decontaminated by vacuuming with a HEPA vacuum to remove loose contaminated dust. The decontamination of these horizontal surfaces was performed in order to comply with DOE's As Low As Reasonably Achievable (ALARA) policy. Preliminary hazard assessment calculations indicated that the wall contamination did not contribute significant dose to a future demolition worker, however, because the dose to the demolition worker was primarily from removable contamination it was decided to vacuum the horizontal surfaces to eliminate it. The radioactive dust generated from the operation was set in concrete to eliminate the potential for fugitive dust during transport to and waste handling at the disposal facility. The concrete/dust was then placed in an intermodal container for disposal at Envirocare of Utah.

Floor

A backhoe and skid steer loader equipped with a hoe-ram attachment were used to remove the wooden block and concrete pads above the base slab in the main bay and the B-series rooms on the south side of the building from column 1 to column 8. The wooden blocks and concrete pads were considered clean material based on volumetric sampling results from the PMC characterization (Ref. 3) and were placed in a clean pile and left onsite with concurrence of the property owner. The floor was vacuumed with HEPA vacuums and the dust collected. Self tapping steel floor anchors were removed with a jackhammer and surveyed. Any floor anchors containing above background contamination were placed in the intermodal for disposal at Envirocare of Utah. Areas of the floor that required further decontamination were mechanically cleaned with side grinders equipped with vacuum shrouds connected to HEPA vacuums. During the survey of the floor it was determined that the ramp at

the west equipment door of building 23 required partial removal to gain access to contamination under the ramp. The portions of the ramp requiring removal were broken up with a jackhammer and a skid steer loader equipped with a hoe-ram attachment.

Shovels were then used to remove the contaminated soil below the ramp. All radioactive dust collected during the decontamination of the floor was mixed with concrete to eliminate the potential for fugitive dust during transport to and waste handling at the disposal facility. The concrete/dust was then placed in an intermodal container for disposal at Envirocare of Utah.

Drain Lines

A ductile iron drain line discovered in room B4 (Figure 3) after removal of the wooden blocks was determined to be contaminated above criteria. Radioactive drain lines identified following the removal of the wooden blocks from the floor were removed by breaking the concrete with jackhammers and a skid steer loader equipped with a hoe-ram attachment. Lead seals in the joints of the pipe were segregated by breaking the pipe away from the seal. A total of 145 feet of 4 inch drain line was removed and placed in the intermodal for shipment to Envirocare of Utah. The lead seals were surveyed and free released then taken to a local lead recycling company.

Contamination Control During Remedial Action

During the remedial action, engineering and administrative controls (such as dust control and hazardous work permits) and personal protective equipment (PPE) were used to protect remediation workers and members of the public from exposure to radiation in excess of applicable standards and in accordance with a site specific safety and health plan for CHV (Ref. 10).

All personnel working in contaminated areas were required to wear Protech coveralls, hard hats, safety glasses, hearing protection, boots, and gloves. If conditions warranted, additional PPE such as face shields were used. Site conditions did not necessitate the use of personnel respiratory protection except in the HEPA vacuum changeout tent.

Workers leaving radioactively restricted work areas were scanned at the control point by a personnel contamination monitor followed by a hand and foot frisk to ensure that they were not contaminated and to prevent the spread of contamination.

The potential primary exposure pathways during remedial action of persons onsite and offsite were inhalation and ingestion of radioactively contaminated airborne dust from mechanical decontamination and excavation activities. HEPA filtration units were used to control the spread of dust and minimize the potential for contaminants to become airborne. In addition a fine water mist was sprayed to control dust during floor removal and during transport of material to the intermodal container. All equipment used in the controlled area was surveyed before being released from the site.

During remediation, particulate air monitoring devices were placed in the areas being remediated. Monitoring locations were selected to provide data for the worst-case scenario. Concentrations of uranium-238 ranging from 1.5×10^{-14} to 5.8×10^{-13} μCi/ml were conservatively derived by collecting air particulate samples daily from lapel air samplers worn by workers. After gross activity per volume of air passed through the filter was determined, the source of all activity on the filter was assumed to be uranium-238. The derived air concentrations (DACs) were then compared with the applicable DOE guideline, which is a DAC of 2.0×10^{-11} μCi/ml for occupational exposures to airborne uranium-238 (Ref. 11).

Area air particulate sampling was also performed adjacent to areas being remediated to ensure that no member of the general public was exposed to radioactivity above DOE guidelines (DOE Order 5400.5). Because all remediation activities were inside of building 23 and there were no active vents in the building it was determined that exterior area air particulate monitoring was not required on the exterior of the building. An Eberline RAS-1 high-volume monitor was used and collected daily and counted after four days to allow for radon decay. The limits in DOE Order 5400.5 are derived concentrations guidelines (DCGs); a DCG is the concentration of a particular radionuclide that would provide an effective dose equivalent of 100 mrem/y, the DOE basic dose limit, to an individual continuously exposed to the radionuclide by one pathway for an entire year.

Concentrations of uranium-238 measured by area particulate monitors ranged from 1.5×10^{-15} to 3.1×10^{-13} μCi/ml. The DCG is 2.0×10^{-12} μCi/ml for uranium-238.

Post-Remedial Action Sampling/Surveying Results

After each portion of the property was decontaminated, a radiological survey of the area was conducted to confirm that all radioactive contamination above the cleanup criteria had been removed. Initial post-remediation surveys were conducted by ThermoNuclear Services (TNS) on behalf of the PMC. Survey techniques used during post-remediation and verification surveys included direct surface contamination measurements, removable contamination measurements, external gamma radiation exposure rate measurements, and soil sampling. The initial post-remediation surveys were conducted in accordance with TNS procedures (Ref. 12) and PMC instruction guides (Ref. 13 and 14). The IVC performed independent verification surveys of the floor, trusses, and bridge crane using similar or identical survey techniques. The IVC survey data will be issued in a separate verification report for CHV.

Discussion of post-remedial action survey data for each decontaminated area are included in the following sections:

10-Ton Bridge Crane

Post-remedial action activities on the crane included direct and removable contamination surveys and sampling of paint and dust residues to determine if the waste was a RCRA mixed waste. Figure 6 shows the survey locations and Table 4 contains the survey data. A total of 169 locations were surveyed and direct alpha and beta/gamma average surface contamination readings were 44 and 520 dpm/100 cm^2, respectively. Average removable alpha and beta/gamma readings were 3 and 42 dpm/100 cm^2, respectively. The crane was successfully decontaminated to DOE residual contamination guidelines listed in Table 1.

Table 9 contains sampling results used in bench scale tests to determine if treatment of the paint and dust residues was a possible waste minimization technique. After acid dissolution and stabilization in a concrete monolith, a sample was collected to determine if the treated material met RCRA standards. The final sample result for TCLP lead and cadmium was 338 and 3.0 μg/l, well below the RCRA limits of 5,000 and 1,000 μg/l. Because the work at CHV met the criteria for very small quantity generation, and the treatment technique was successful, the concrete monoliths were left onsite for disposal by The Crane Company. This was done with the concurrence of the State of Massachusetts and the property owner.

Overhead Trusses

Fixed and removable contamination surveys were conducted on the trusses to determine the effectiveness of the decontamination effort. The results of this survey are included in Table 5. The maximum fixed averages per truss for alpha and beta/gamma contamination were 2,114 and 12,261 dpm/100 cm^2, respectively. Both of these readings were from truss 2 and were below the supplemental limit of 15,000 dpm/100 cm^2 average per truss. All removable contamination readings were below 1,000 dpm/100 cm^2. An additional survey conducted on non-horizontal surfaces and between welded angles of the trusses confirmed that that the supplemental limits had not been exceeded and these areas did not require decontamination.

Horizontal Wall Surfaces

The ALARA approach implemented on the horizontal wall surfaces was performed to decontaminate those surfaces with removable contamination that exhibited a potential for resuspension or migration. Survey locations for the west, north, and south walls are shown in Figures 7, 8, and 9. Survey results for the west, north, and south walls are included in Tables 6, 7, and 8. The maximum and average removable alpha measurements for the west, north, and south walls were 164 and 17 dpm/100 cm^2, 850 and 35 dpm/100 cm^2, and 111 and 17 dpm/100 cm^2, respectively. The maximum and average removable beta/gamma measurements for the west, north, and south walls were 191 and 70 dpm/100 cm^2, 3,197 and 58 dpm/100 cm^2, and 319 and 45 dpm/100 cm^2, respectively. Except for the single reading on the north wall all measurements were below the

removable criteria of 1,000 dpm/100 cm^2. The decontamination effort was considered successful in lowering the levels of removable contamination. The survey data is used in the final hazard assessment (Ref. 15) which is summarized in this technical memorandum.

Floor

Figure 10 summarizes survey results for the decontamination effort on the floor. A table of the survey results is not shown in this technical memorandum because of the large quantity of data (1,598 locations) and because all data results were below DOE Order 5400.5 criteria. Fixed beta/gamma contamination was the contaminant of concern on the floor; therefore only fixed beta/gamma measurements are summarized in Figure 10. Average, minimum, and maximum fixed beta/gamma results were 914, -872, and 4,934 dpm/100 cm^2, respectively (Note: negative numbers indicate that the measurement was less than the minimum detectable activity and that after background was subtracted, the numerical value was negative). A soil sample (Table 9) was also collected from under the removed section of the west equipment door ramp to confirm successful decontamination of the soil under the ramp. Sample results for ^{238}U, ^{226}Ra, and ^{232}Th were 2.0, 0.47, and 0.41 pCi/g. A site specific uranium guideline for CHV was not determined; however these results were well below the typical ^{238}U guideline for a FUSRAP site of 35 to 50 pCi/g. The ^{226}Ra and ^{232}Th results are below the DOE Order 5400.5 criteria of 5 pCi/g for surface soils.

Drain Lines

Following the removal of 145 feet of the 4 inch drain lines in room B4 and prior to restoration, three composite soil samples (Table 9) were collected in the trench to determine if the decontamination effort was successful. The maximum sample results for ^{238}U, ^{226}Ra, and ^{232}Th were 0.62, 0.47, and 0.50 pCi/g. A site specific uranium guideline for CHV was not determined; however the results are well below the typical ^{238}U guideline for a FUSRAP site of 35 to 50 pCi/g. The ^{226}Ra and ^{232}Th results are below the DOE Order 5400.5 criteria of 5 pCi/g for surface soils. Contaminated drainlines were loaded into the intermodal and sent to Envirocare of Utah for disposal.

Hazard Assessment

A hazard assessment was prepared (Ref. 15) to document the post remediation condition for the CHV. A summary of the exposure scenarios is included in this technical memorandum and a summary of the results of the hazard assessment is listed in Table 10. The hazard assessment also calculates the total mass of uranium at CHV and the volumetric activity of the rubble resulting from building demolition. The volumetric activity was compared to activities of material commonly placed in a landfill (Table 11).

The bases for this assessment are:

- Contamination at the CHV is found only on limited portions of Building 23. The total curie content of all affected building components is approximately 6 mCi.
- Building 23 is deteriorating, abandoned, and virtually certain to be demolished.

Exposure Scenarios

The hypothetical demolition phases representing each of the building components, potential reuse/recycling, and significant conditions/assumptions are as follows.

Roof Demolition

Contamination is consolidated into built-up roofing material. The primary exposure pathway is inhalation of contaminated dust. The primary dust producing activities would be: destroying the roof with heavy equipment (allowing it fall to the ground); and, loading the roof rubble into trucks using heavy equipment. This scenario is plausible but conservative in that the roof contains asbestos and the likely scenario would involve appropriate asbestos controls - those controls would also mitigate release of airborne radioactivity and worker exposure in general.

Wall Demolition

The maximum dust producing activity would involve the use of a wrecking ball (or other forceful demolition technique) where the impact of the ball occurs at approximately 20 feet above ground and dislodges (and makes airborne) some surface contamination on the walls (from brick and glass surfaces) as well as producing dust as clean portions of the brick, mortar, and glass are pulverized. To maximize the potential exposure, the calculation approach assumes demolition is performed one wall at a time, and the rubble from one wall is removed before the next wall is demolished. In addition to the initial plume, contaminated dust would be resuspended as rubble is removed.

Structural Steel Demolition

Decontamination was sufficient so that no airborne contamination would result from the exposed steel surfaces during demolition. Some areas, such as intersections of steel members, gusset plates, etc., are more difficult to access and could be the source of airborne contamination as the trusses are removed.

Following decontamination it is assumed that some of the contamination from inaccessible areas will be dislodged as the steel building framework is dismantled. The contamination on the inaccessible surfaces is assumed to be the same as the exposed surfaces surveyed during characterization. The contamination is from deposited dust which is assumed to be 0.1 cm thick.

The primary exposure pathways are inhalation and ingestion. The activity generating the most airborne contamination will entail toppling the supporting beams and allowing the trusses to fall to the ground where they will subsequently be salvaged for scrap. It is assumed that torch cutting will be performed to size the metal for transport. Airborne contamination will be released as: (1) the trusses fall to the ground; (2) the trusses are cut for transport; and, (3) the trusses are removed. Assuming all trusses are felled in a very short time frame, the contaminated plume from any one truss falling will not have dissipated when the next truss is dropped; thus, the plume of airborne contamination will include all the contamination released from the trusses.

Personal contamination (and subsequent ingestion) will occur as the trusses are handled during cutting and removal.

Steel Recycling

Potential exposures from reuse or recycling of the steel were considered for three activities:

- Torch cutting of the steel so that it can be of the size necessary for acceptance for melting;
- Sand blasting of the steel to remove lead based paint so the smelter will accept the steel
- Exposure from the discarded slag after melting

Floor Block Removal

The floor of building 23 is composed separately of 3 different building materials; concrete, fire brick, or wood block, each overlaid on a concrete base slab. Over 90% of the floor area is composed of wood blocks. Isolated contamination has been found on the surface of the

floor, in adhesive between wood blocks and gaps in fire brick and possibly on the concrete base slab. The dose to a potential future worker removing the blocks left onsite during remedial action was calculated.

Reuse of Building Materials

The dose that a person could receive if he were to live in a small building constructed with the contaminated materials from CHV was calculated. The building walls are from the west wall, ceiling from the roofing material, and floor from the wood blocks.

Conclusions of Hazard Assessment

Residual contamination on the roof and walls of Building 23 will not result in a radiation dose above 100 mrem to any member of the general public. The maximum dose to a hypothetical demolition worker is estimated to be 5.6 mrem, and offsite exposure to the general public would be much less. Brick comprises approximately 64% of the mass of the contaminated portion of Building 23. The natural (background) radioactivity found in brick at the site is approximately 6.4 pCi per gram. The amount of radioactivity added by the surface contamination ranged from 0.18 to 1.5 pCi/g; thus, the contamination has only increased the radioactivity naturally found in brick less than 23 percent. Given the low radiation dose calculated to result from roof and wall contamination (and the fact that these calculations were very conservative and likely over-estimated the dose) supplemental limits for these building components should be the same as the contamination levels there at present. No remedial action need be performed in these areas because of the low doses calculated in the hazard assessment.

Post-Remedial Action Status

Analytical results from post-remedial action surveys indicate that the levels of radioactivity in the remediated areas meet applicable DOE cleanup guidelines. A summary of remedial action at CHV is provided in Table 12. The IVC has reviewed the post-remedial action surveys and results and determined that the measurements obtained verify that the remediated areas comply with the established DOE guidelines for the site. No areas of contamination above DOE guidelines or supplemental limits remain at the site.

After completing the verification survey, the IVC will report its findings and recommendations to DOE Headquarters and the DOE Oak Ridge Operations Office. DOE will review the report to verify that the remedial action was successful, and a certification docket will then be prepared. The certification docket officially certifies that the site has been successfully remediated to established criteria. The issuance of the certification docket will be documented through publication of a notice in the *Federal Register*.

References

Ref. 1 Letter from Alderman and MacNeish to Con-Test, inc, *Structural Inspection, Crane Company,* November, 1992.

Ref. 2 *Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts,* ORNL/RASA-92/1, Oak Ridge National Laboratory, Oak Ridge, TN, July 1992.

Ref. 3 *FUSRAP Technical Memorandum - Chapman Valve Characterization Results,* TM 133-45-003 Bechtel National, Inc., Oak Ridge, TN, May 4, 1995.

Ref. 4 *Uranium Mill Tailings Remedial Action Program,* Department of Energy, 1986.

Ref. 5 *Radiation Protection of the Public and the Environment,* DOE Order 5400.5, Department of Energy, Washington DC, June 1990.

Ref. 6 *Technical Study for the Remedial Action at the Chapman Valve Site, Indian Orchard, Massachusetts,* Department of Energy, May 1995.

Ref. 7 Memorandum from Wagoner to Price, *Supplemental Limits for Residual Uranium at the Chapman Valve Site, Indian Orchard Massachusetts,* July 27, 1995.

Ref. 8 Letter from Weinberg to Pantaleoni, Springfield 1-0170, *Remedial Action Plan Concurrence,* June 15, 1995.

Ref. 9 Letter from O'Connell to Kopotic, *Technical Study Concurrence,* June 14, 1995.

Ref. 10 *Health and Safety Work Instruction for Chapman Valve,* WI-95-126, Bechtel National, Inc., June 21, 1995.

Ref. 11 Radiation Protection for Occupational Workers, DOE Order 5480.11, Department of Energy, Washington DC, July 1989.

Ref. 12 *Health Physics Operational Procedures Manual,* Thermo Nuclear Services.

Ref. 13 *Instruction Guide for Post-Remediation Survey of Soil,* 191-IG-032, Rev. 0, Bechtel National, Inc., 1993.

Ref. 14 *Instruction Guide for Post-Remediation Survey of Structures,* 191-IG-031, Rev. 0, Bechtel National, Inc., 1992.

Ref. 15 *Hazard Assessment for Chapman Valve Site,* Calculation 133-CV-001, Rev. 1, Bechtel National, Inc.

Table 1
Summary of Residual Contamination Guidelines[a]

Basic Dose Limits

The basic limit for the annual radiation dose received by an individual member of the general public is 100 mrem/yr. In implementing this limit DOE applies as low as reasonably achievable (ALARA) principles to set site-specific guidelines.

Soil Guidelines[b,c,d,e]

Radium-226 Radium-228 Thorium-230 Thorium-232	5 pCi/g when averaged over the first 15 cm (6 in.) of soil below the surface: 15 pCi/g when averaged over any 15-cm (6 in.)-thick soil layer below the surface layer.
Uranium[f]	Guideline being developed on a site-specific basis.

Water Guidelines

DOE Derived Concentration Guidelines[g]

Radium-226	100 pCi/L (100 x 10^{-9} µCi/ml)
Thorium-230	300 pCi/L (300 x 10^{-9} µCi/ml)
Thorium-232	50 pCi/L (50 x 10^{-9} µCi/ml)
Uranium-238	600 pCi/L (600 x 10^{-9} µCi/ml)
Uranium-235	600 pCi/L (600 x 10^{-9} µCi/ml)
Uranium-234	500 pCi/L (500 x 10^{-9} µCi/ml)
Total Uranium[h]	500 pCi/L (500 x 10^{-9} µCi/ml)

Doe Drinking Water Guidelines[i]

Radium-226 + Radium-228	5 pCi/L (5 x 10^{-9} µCi/ml)
Gross Alpha	15 pCi/L (15 x 10^{-9} µCi/ml)

Structural Guidelines

Airborne Radon Decay Products

Generic guidelines for concentration of airborne radon decay products shall apply to existing occupied or habitable structures on private property that has no radiological restrictions on its use; structures that will demolished or buried are excluded. The applicable generic guideline (40 CFR 192) is: in any occupied or habitable building, the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) shall not exceed 0.02 WL[j]. In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions are not required in order to comply with this guideline when there is reasonable assurance that residual radioactive materials are not the cause.

External Gamma Radiation

The average level of gamma radiation inside a building or habitable structure on a site that has no radiological restriction on its use shall not exceed the background level by more the 20 µR/h and will comply with the basic dose limits when an appropriate-use scenario is considered.

Indoor/Outdoor Structure Surface Contamination

	Allowable Surface Residual Contamination[k] (dpm/100 cm^2)		
Radionuclide[l]	Average[m,n]	Maximum[m,o]	Removable[m,p]
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-124, I-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000 α	15,000 α	1,000 α
Beta-gamma emitters (radionuclides with decay modes other that alpha emission or spontaneous fission except Sr-90 and other noted above)	5,000 β–γ	15,000 β–γ	1,000 β–γ

Table 2
Decontamination Techniques Used at the CHV

Type	Description
HEPA Vacuuming	High-efficiency particulate air- (HEPA-) filtered vacuum cleaners were used to remove loose contamination and dust. HEPA vacuums were also used in conjunction with other techniques (grinding, wire brushing, etc.) to eliminate the airborne contamination associated with these techniques.
Jackhammering	Conventional jackhammers were used on small areas to remove anchor bolts from the concrete slab. Skid steer loaders equipped with hoe-ram attachments were used to remove the wooden blocks from the floor and to break up the concrete pads to expose the base slab.
Excavation	Contaminated concrete and debris was removed from the building with a skid steer loader. Contaminated soil from the west ramp and the pipe excavation was performed with shovels.
Wire Brushing	Small areas on the overhead trusses requiring rework were wire brushed to remove contamination.
Scraping	Scraping with putty knifes was utilized to remove contamination from the surface of the 10-ton crane and from wooden planks removed from the crane deck.

Table 4

Chapman Valve Post-Remedial Action Crane Survey Data[a]

Location Number	Direct (DPM/100 cm^2)		Removable (DPM/100 cm^2)[b]		Location Number	Direct (DPM/100 cm^2)		Removable (DPM/100 cm^2)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	16	201	2	84	33	8	-177	-[b]-	-[b]-
2	1	513	2	110	34	15	25	-[b]-	-[b]-
3	-6	313	-1	-19	35	15	-354	-[b]-	-[b]-
4	9	223	2	20	36	15	-76	-[b]-	-[b]-
5	1	335	-1	-53	37	1	354	-[b]-	-[b]-
6	-6	112	-1	-45	38	-5	-101	-[b]-	-[b]-
7	-6	201	-1	15	39	8	-380	-[b]-	-[b]-
8	-6	402	-1	41	40	21	708	-[b]-	-[b]-
9	-6	380	-1	110	41	26	101	-[b]-	-[b]-
10	16	491	2	3	42	12	557	-[b]-	-[b]-
11	9	313	5	80	43	19	-127	-[b]-	-[b]-
12	1	134	-1	50	44	-3	177	-[b]-	-[b]-
13	9	357	2	114	45	4	633	-[b]-	-[b]-
14	9	201	2	11	46	4	481	-[b]-	-[b]-
15	1	290	-1	-15	47	134	1493	8	73
16	37	0	2	33	48	220	2378	5	87
17	9	112	2	20	49	163	2606	-2	77
18	-6	290	-1	24	50	55	708	-[b]-	-[b]-
19	15	1670	-2	7	51	4	936	-[b]-	-[b]-
20	-5	-329	-[b]-	-[b]-	52	12	354	-[b]-	-[b]-
21	48	1771	4	15	53	19	734	-[b]-	-[b]-
22	15	177	-[b]-	-[b]-	54	19	911	-[b]-	-[b]-
23	21	607	-[b]-	-[b]-	55	19	405	-[b]-	-[b]-
24	1	633	-[b]-	-[b]-	56	-3	607	-[b]-	-[b]-
25	21	253	-[b]-	-[b]-	57	12	531	-[b]-	-[b]-
26	55	3416	1	45	58	148	1695	8	77
27	15	329	-[b]-	-[b]-	59	91	2075	12	39
28	8	2049	1	-53	60	26	253	-[b]-	-[b]-
29	21	-278	-[b]-	-[b]-	61	19	0	-[b]-	-[b]-
30	35	1619	10	63	62	62	354	-[b]-	-[b]-
31	28	1569	4	11	63	12	177	-[b]-	-[b]-
32	48	531	-[b]-	-[b]-	64	-3	633	-[b]-	-[b]-

14 3 7 1 9

Table 4 - Continued

Chapman Valve Post-Remedial Action Crane Survey Data[a]

Location Number	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Removable (DPM/100 cm^2)[b] Alpha	Removable (DPM/100 cm^2)[b] Beta/Gamma
127	-24	-786	-[b]-	-[b]-
128	-33	-647	-[b]-	-[b]-
129	689	-324	-[b]-	-[b]-
130	198	-69	-[b]-	-[b]-
131	143	23	-[b]-	-[b]-
132	319	-92	-[b]-	-[b]-
133	87	-370	-[b]-	-[b]-
134	235	3237	-1	-3
135	161	671	-[b]-	-[b]-
136	13	1156	-1	-10
137	4	324	-[b]-	-[b]-
138	-43	69	-[b]-	-[b]-
139	-33	-254	-[b]-	-[b]-
140	-43	-925	-[b]-	-[b]-
141	-43	-416	-[b]-	-[b]-
142	13	-879	-[b]-	-[b]-
143	172	651	-[b]-	-[b]-
144	24	311	-[b]-	-[b]-
145	43	481	-[b]-	-[b]-
146	6	1471	12	17
147	89	396	-[b]-	-[b]-
148	302	283	-[b]-	-[b]-
149	459	736	-[b]-	-[b]-
150	552	5348	9	-17
151	274	4641	1	58
152	98	368	-[b]-	-[b]-
153	43	-57	-[b]-	-[b]-
154	15	85	-[b]-	-[b]-
155	-4	-170	-[b]-	-[b]-
156	24	538	-[b]-	-[b]-
157	24	-311	-[b]-	-[b]-
158	24	-226	-[b]-	-[b]-
159	70	1585	1	31
160	256	538	-[b]-	-[b]-
161	89	198	-[b]-	-[b]-
162	61	-113	-[b]-	-[b]-
163	98	311	-[b]-	-[b]-
164	172	1330	12	20
165	80	-170	-[b]-	-[b]-
166	43	-85	-[b]-	-[b]-
167	33	283	-[b]-	-[b]-
168	70	-198	-[b]-	-[b]-
169	61	-198	-[b]-	-[b]-
			-[b]-	-[b]-
AVERAGE	44	520	3	42
DOE Guideline	5000	5000	1000	1000

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Missing removable surface data was not taken because direct readings were less than the removable criteria of 1000 dpm/100 cm^2 for that location.

14 34 19

Table 6

Chapman Valve Post-Remedial Action West Wall Survey Data[a]

Location Number	Direct (DPM/100 cm^2)		Removable (DPM/100 cm^2)[b]		Location Number	Direct (DPM/100 cm^2)		Removable (DPM/100 cm^2)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	100	4075	-1	18	22	989	8263	19	187
2	82	3792	-1	24	23	878	80985	9	119
3	193	8489	-1	4	24	322	5490	-1	-6
4	137	14375	5	45	25	626	23147	19	52
5	1304	37295	12	119	26	774	11149	2	65
6	878	61743	42	401	27	4470	14544	108	170
7	2508	15903	9	123	28	6	2999	2	-30
8	693	22977	5	106	29	6	6112	12	18
9	600	59479	75	221	30	70	4980	2	18
10	2230	130787	164	919	31	43	1471	-1	18
11	4824	60045	9	113	32	61	170	-[b]-	-[b]-
12	3731	5433	138	594	33	6	1075	9	18
13	44	5999	5	38	34	154	30730	2	-37
14	248	4301	-1	28	35	302	12620	2	-16
15	359	34692	2	89	36	395	26033	15	4
16	433	10922	2	14	37	376	10413	9	31
17	174	14997	9	31	38	24	4075	9	4
18	63	16921	5	92	39	43	3679	5	-9
19	137	22354	5	102	40	43	1585	2	-64
20	433	6508	5	18	41	61	6452	-1	-57
21	878	15167	2	85	42	24	2999	2	-30
					Average	708	19410	17	70
					Doe Guideline	-[c]-	-[c]-	-[c]-	-[c]-

a Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

b Missing removable surface data was not taken because direct readings were less than the removable criteria of 1000 dpm/100 cm^2 for that location.

c The walls were decontaminated to comply with DOE's ALARA policy and the data collected is intended to be used in the final hazard assessment calculation; therefore there is no specific surface criteria that applies to the walls.

14 3 4 1 6

Table 7 - continued

Chapman Valve Post-Remedial Action North Wall Survey Data[a]

Location Number	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Removable (DPM/100 cm^2)[b] Alpha	Removable (DPM/100 cm^2)[b] Beta/Gamma	Location Number	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Removable (DPM/100 cm^2)[b] Alpha	Removable (DPM/100 cm^2)[b] Beta/Gamma
33	919	1503	22	79	66	104	856	-[b]-	-[b]-
67	363	740	-[b]-	-[b]-	82	169	1064	18	-35
68	132	763	-[b]-	-[b]-	83	39	509	-[b]-	-[b]-
69	539	1434	-2	33	84	548	6544	31	185
70	11	2127	5	-5	85	132	1572	21	29
71	178	1064	8	22	86	48	23	-[b]-	-[b]-
72	280	1202	18	39	87	1021	14268	31	155
73	243	2428	21	9	88	48	3645	15	39
74	94	555	-[b]-	-[b]-	89	576	3076	15	-62
75	30	1804	11	-59	90	20	-416	-[b]-	-[b]-
76	169	532	-[b]-	-[b]-	91	1437	27749	25	243
77	85	2405	18	-28	92	2595	4648	213	266
78	94	-185	-[b]-	-[b]-	93	345	-46	-[b]-	-[b]-
79	910	2012	38	-45	94	372	2359	34	100
80	11	1526	-2	39	95	446	7307	101	209
81	335	1896	11	46	96	48	4116	5	-32
					Average	1001	2823	35	58
					Doe Guideline	-[c]-	-[c]-	-[c]-	-[c]-

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Missing removable surface data was not taken because direct readings were less than the removable criteria of 1000 dpm/100 cm^2 for that location.

[c] The walls were decontaminated to comply with DOE's ALARA policy and the data collected is intended to be used in the final hazard assessment calculation; therefore there is no specific surface criteria that applies to the walls.

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Table 8 - Continued

Chapman Valve Post-Remedial Action South Wall Survey Data[a]

Location Number	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Removable (DPM/100 cm^2)[b] Alpha	Removable (DPM/100 cm^2)[b] Beta/Gamma	Location Number	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Removable (DPM/100 cm^2)[b] Alpha	Removable (DPM/100 cm^2)[b] Beta/Gamma
33	567	4278	11	121	66	104	578	--[b]--	--[b]--
67	283	1188	32	106	73	450	22864	2	11
68	6	4358	2	-50	74	43	3679	2	-23
69	1469	3452	12	45	75	98	2320	15	-6
70	580	1075	9	-30	76	43	2037	5	-9
71	265	1981	9	85	77	6	1358	-1	-37
72	450	10696	12	75					
					Average	642	6387	17	45
					Doe Guideline	--[c]--	--[c]--	--[c]--	--[c]--

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Missing removable surface data was not taken because direct readings were less than the removable criteria of 1000 dpm/100 cm^2 for that location.

[c] The walls were decontaminated to comply with DOE's ALARA policy and the data collected is intended to be used in the final hazard assessment calculation; therefore there is no specific surface criteria that applies to the walls.

Table 10
Hazard Assessment Summary

Dose Scenario	Calculated Number	Units
Building Demolition		
Roof Demolition	4.9	mrem
Wall Demolition	0.02	mrem
Structural Steel Demolition	0.76	mrem
Removal of Floor Blocks	0.0007	mrem
Total	5.6	mrem
Structural Steel Recycling		
Torch Cutting of Trusses	5.3	mrem
Sand Blasting of Trusses	10.7	mrem
Slag From Melting Trusses	0.29	mrem
Total	16.3	mrem
Reuse of Building Materials	17.3	mrem/yr
Activity of Building Rubble	4.4	pCi/g
Mass of Uranium in Rubble	20	lbs

143419

Table 12
REMEDIAL ACTION SUMMARY

WBS	133
SITE	CHAPMAN VALVE
OWNER	CRANE COMPANY
ADDRESS	203 HAMPSHIRE STREET
CITY,STATE	INDIAN ORCHARD, MA 05515

REMEDIATION AUTHORITY
- [x] NEPA/CERCLA
- [ ] SUPERFUND
- [ ] RCRA

ACTION	DATE	RESPONSIBLE ENTITY	DOCUMENT
DESIGNATION	12/15/92	DOE	AUTHORIZATION FOR RA
CHARACTERIZATION	Jul-92	ORNL	RESULTS OF RADIOLOGICAL SURVEY AT THE FORMER CHAPMAN VALVE MANUFACTURING COMPANY, INDIAN ORCHARD, MASSACHUSSETTS
CHARACTERIZATION	5/4/95	BNI	FUSRAP TECHNICAL MEMORANDUM - CHAPMAN VALVE CHARACTERIZATION RESULTS
FINAL RA	TBD (PROJECTED COMPLETION MAY/96)	BNI/DOE/ORNL	FUSRAP TECHNICAL MEMORANDUM - CHAPMAN VALVE POST-REMEDIAL ACTION REPORT

TOTAL VOLUME	750 CY
To Remain In Situ	731 CY
Volume Reduction	
Net Disposal	19 CY

DOCUMENTATION USED: HAZARD ASSESSMENT IN FUSRAP TECHNICAL MEMORANDUM - CHAPMAN VALVE POST-REMEDIAL ACTION REPORT

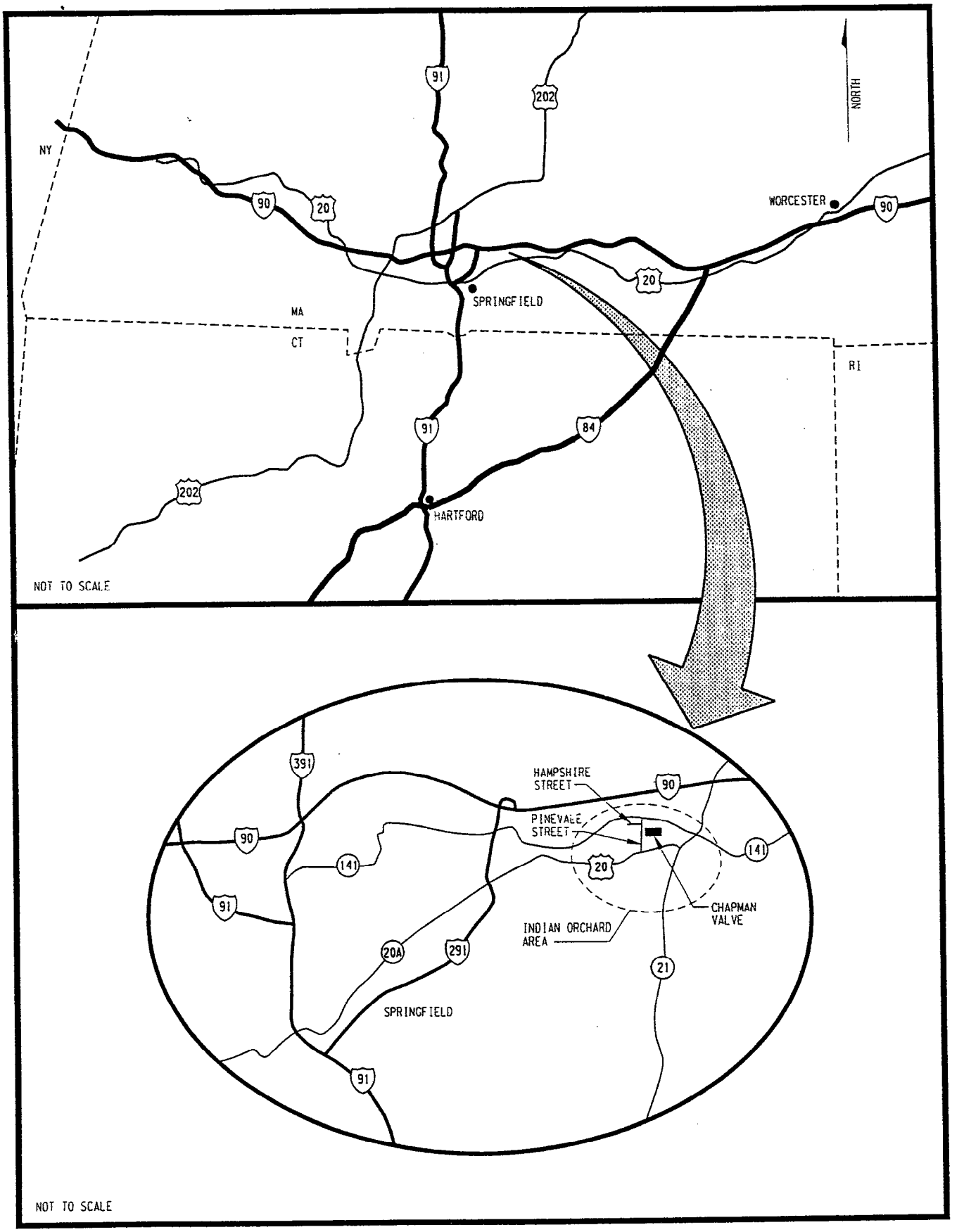
TYPE OF WASTE FOR NET DISPOSAL

REGULATORY		VOLUME	DISPOSAL SITE
[x] LLRW		19 CY	CLIVE, UTAH
[ ] 11(E)2			
[ ] MIXED			
[ ] CHEMICAL			
PHYSICAL			
[x] BUILDING RUBBLE		6 CY	CLIVE, UTAH
[x] SOIL		3 CY	CLIVE, UTAH
[ ] LIQUID			
[x] OTHER	DUST MIXED WITH CONCRETE	10 CY	CLIVE, UTAH

TREATMENT TECHNOLOGIES APPLIED AT THE SITE

CHEMICAL STABILIZATION

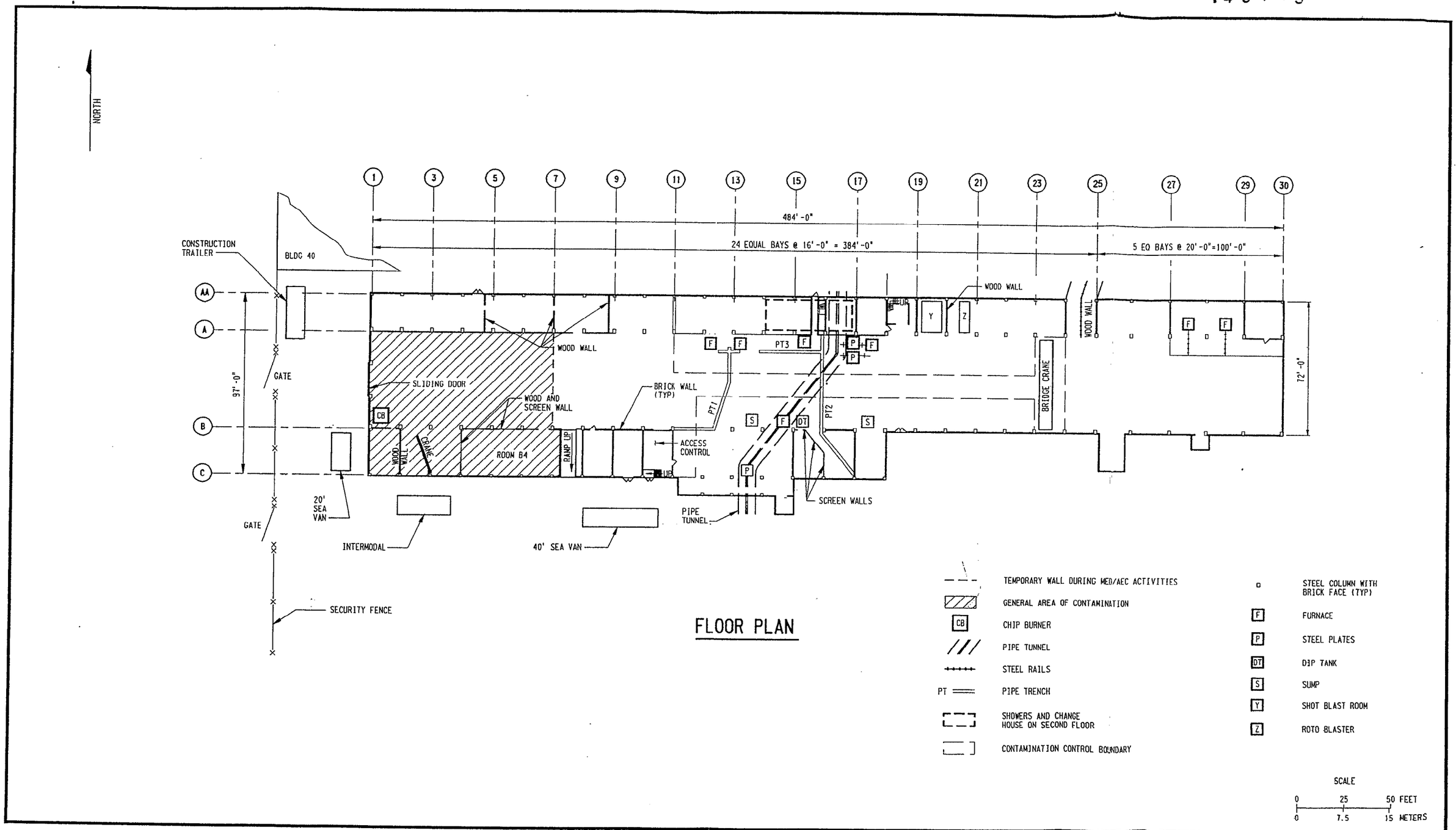
143419



R76F010.DGN

Figure 1
Approximate Location of Chapman Valve Site

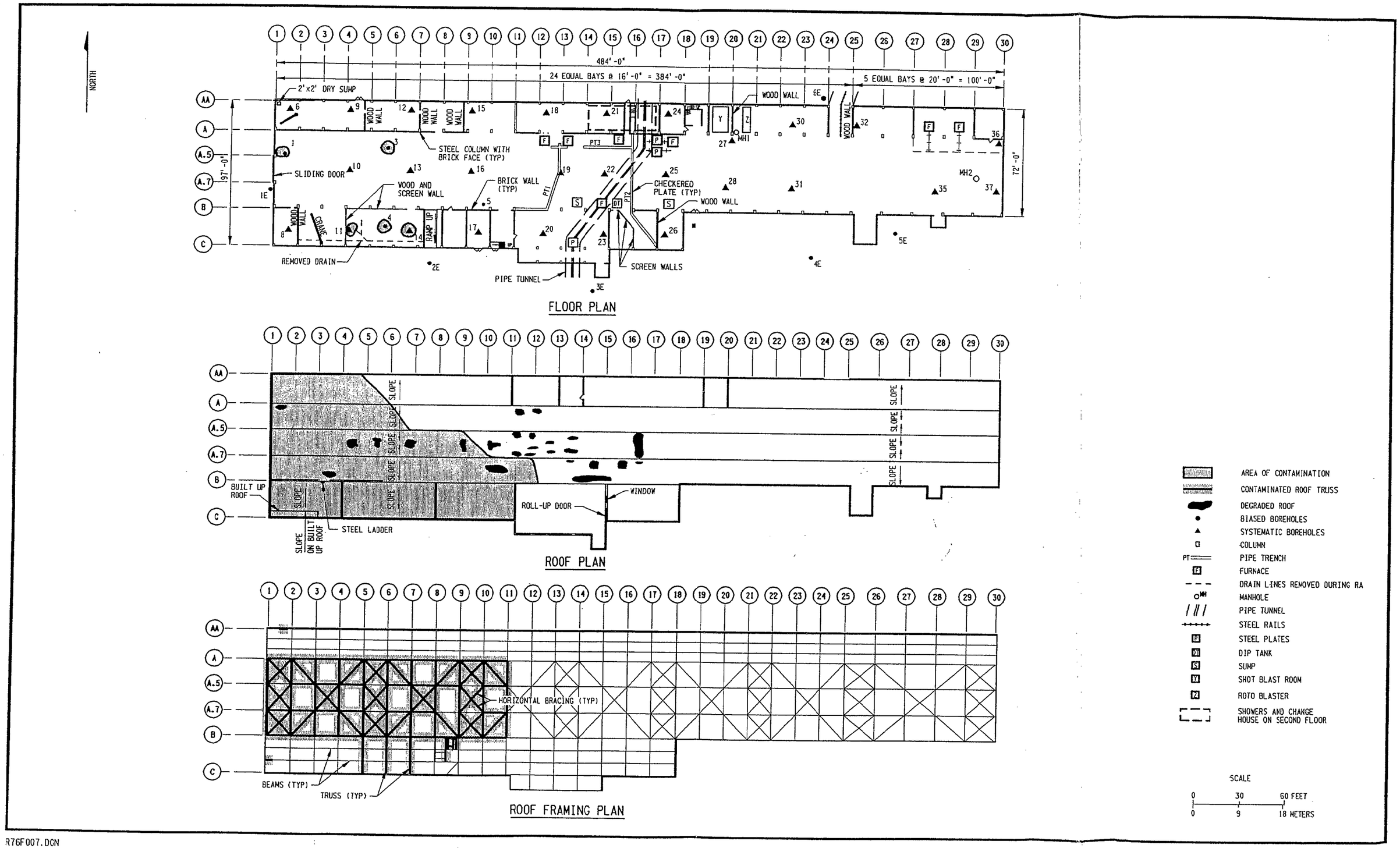
143419



R76F009.DGN

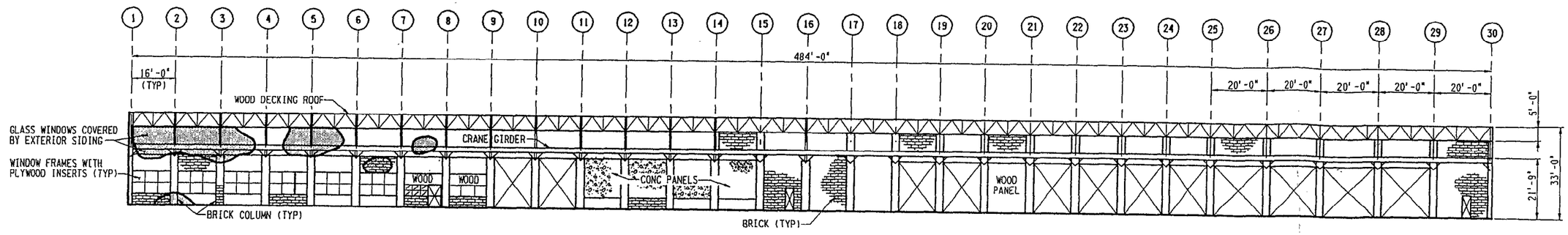
Figure 2
Chapman Valve
Building 23 - Plan View

143419

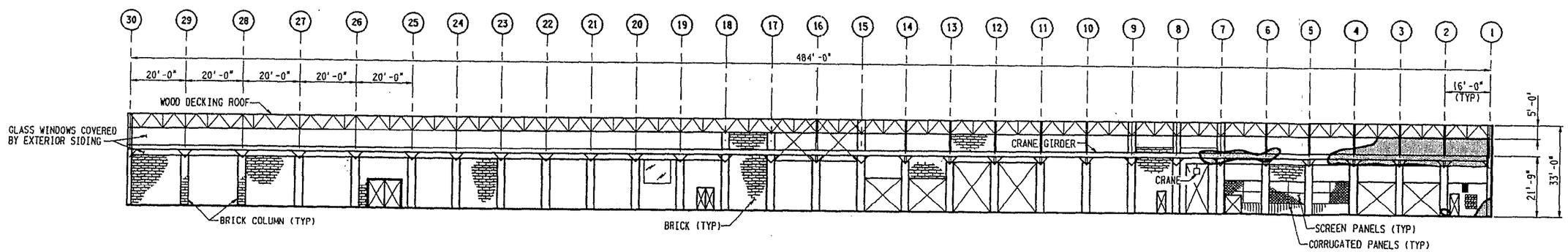


R76F007.DGN

Figure 3
Chapman Valve - 94 Characterization Results
Plan Views



WALL A - ELEVATION LOOKING NORTH



WALL B - ELEVATION LOOKING SOUTH

CONTAMINATED SURFACE

INDICATES OPENING OR DOORWAY

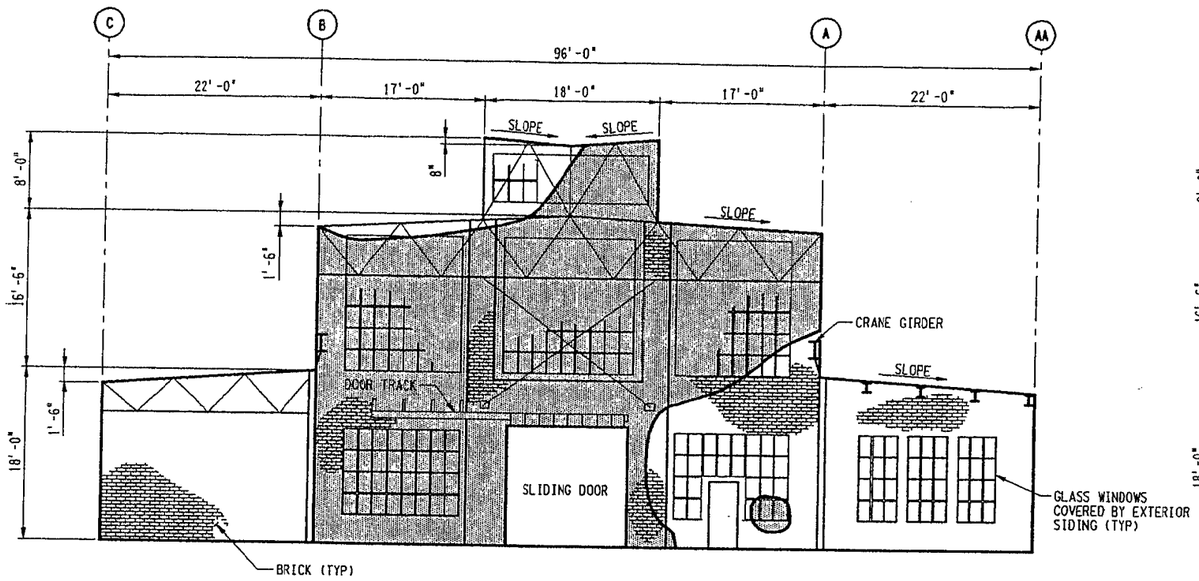
SCALE

0 20 40 FEET

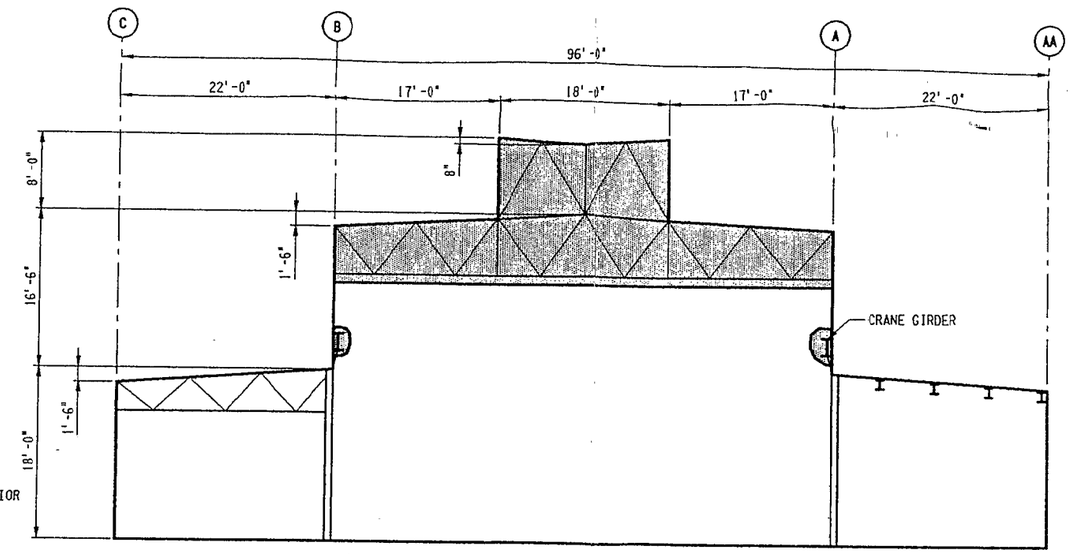
0 6 12 METERS

R76F006.DGN

Figure 4
Chapman Valve - 94 Characterization Results
Interior North and South Wall



WALL 1 - ELEVATION LOOKING WEST



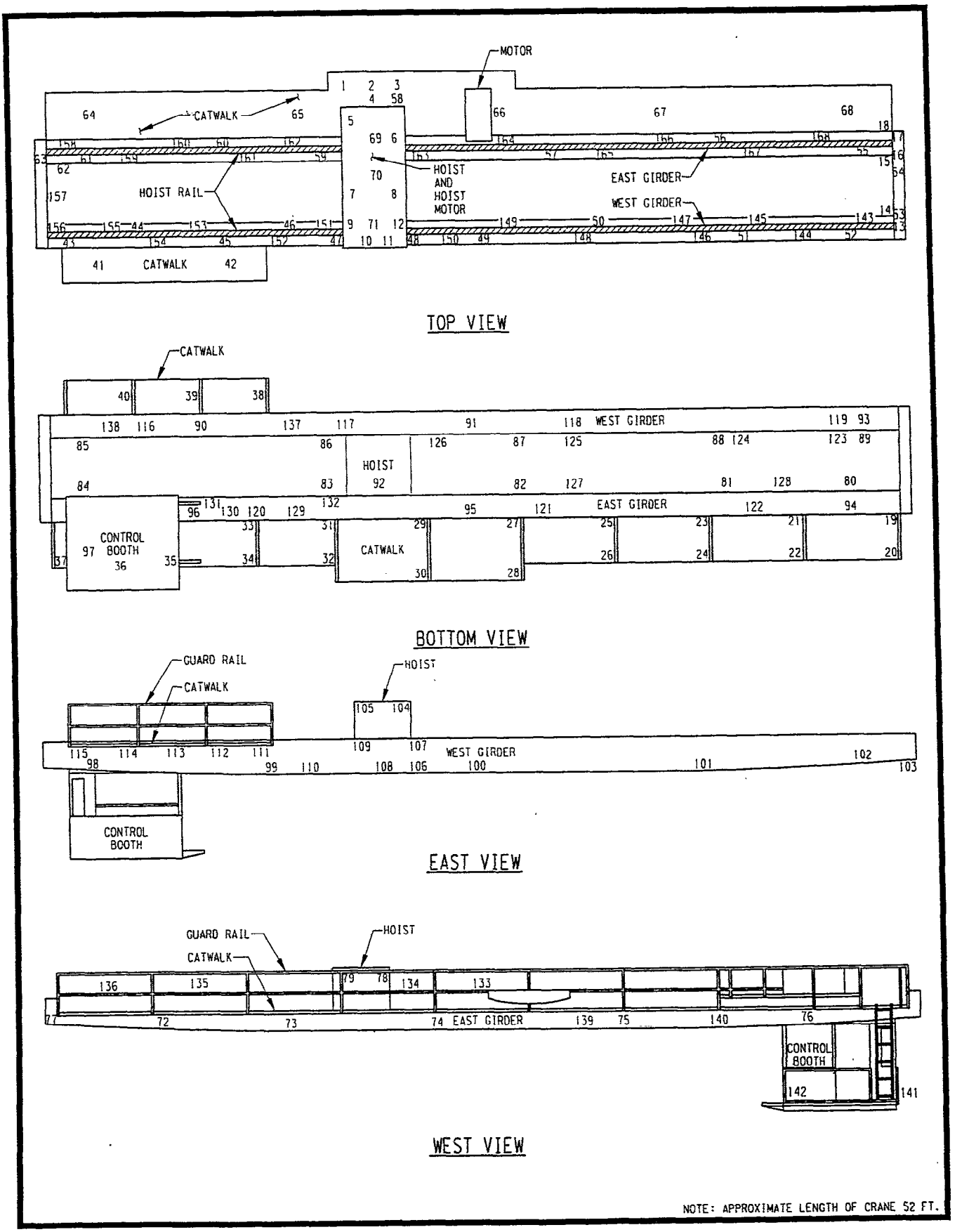
COLUMN 3 - ELEVATION LOOKING WEST

CONTAMINATED SURFACE

0 8 16 FEET
SCALE

R76F008.DGN

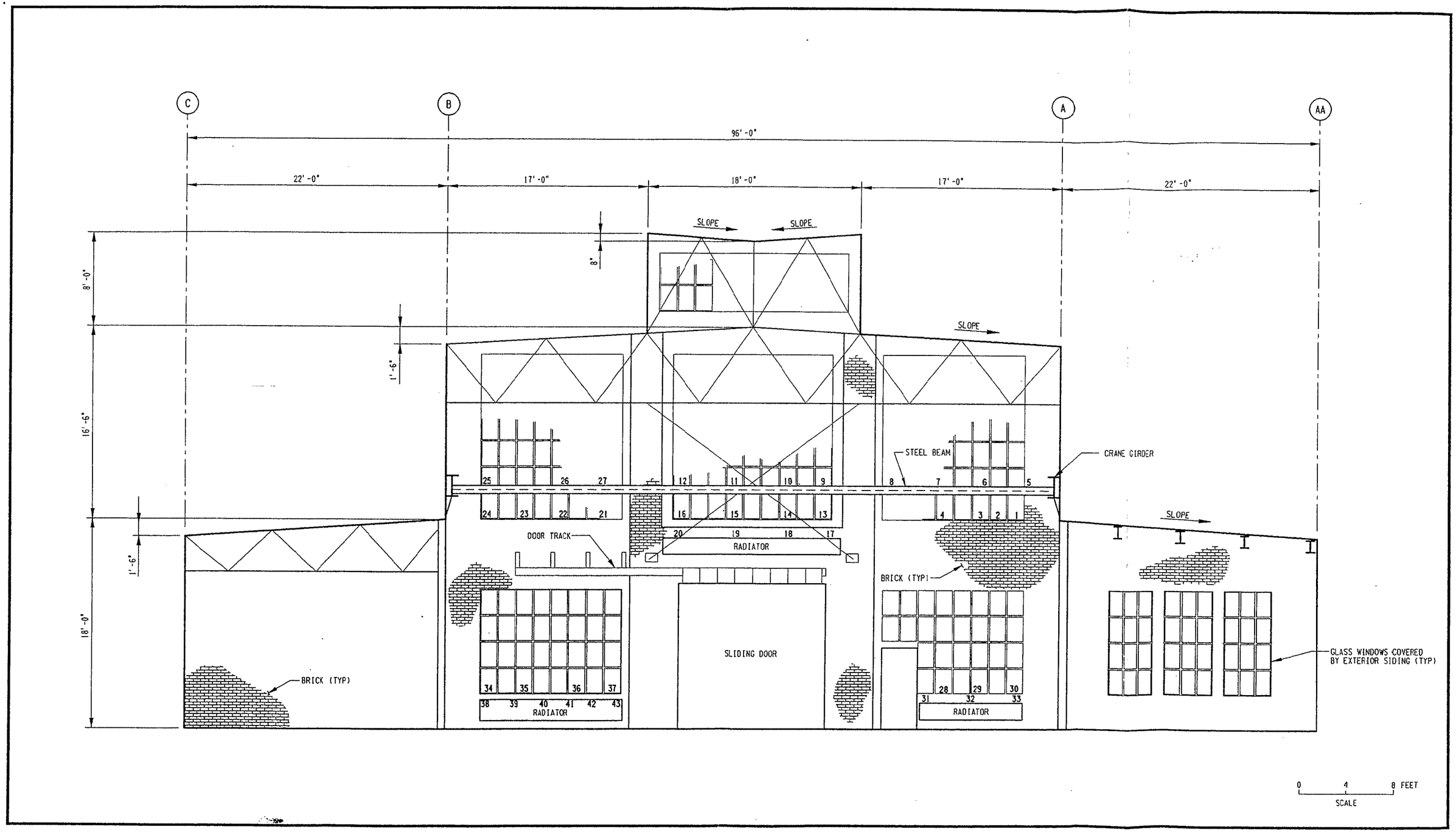
Figure 5
Chapman Valve - 94 Characterization Results
Interior West Wall



R76F011.DGN

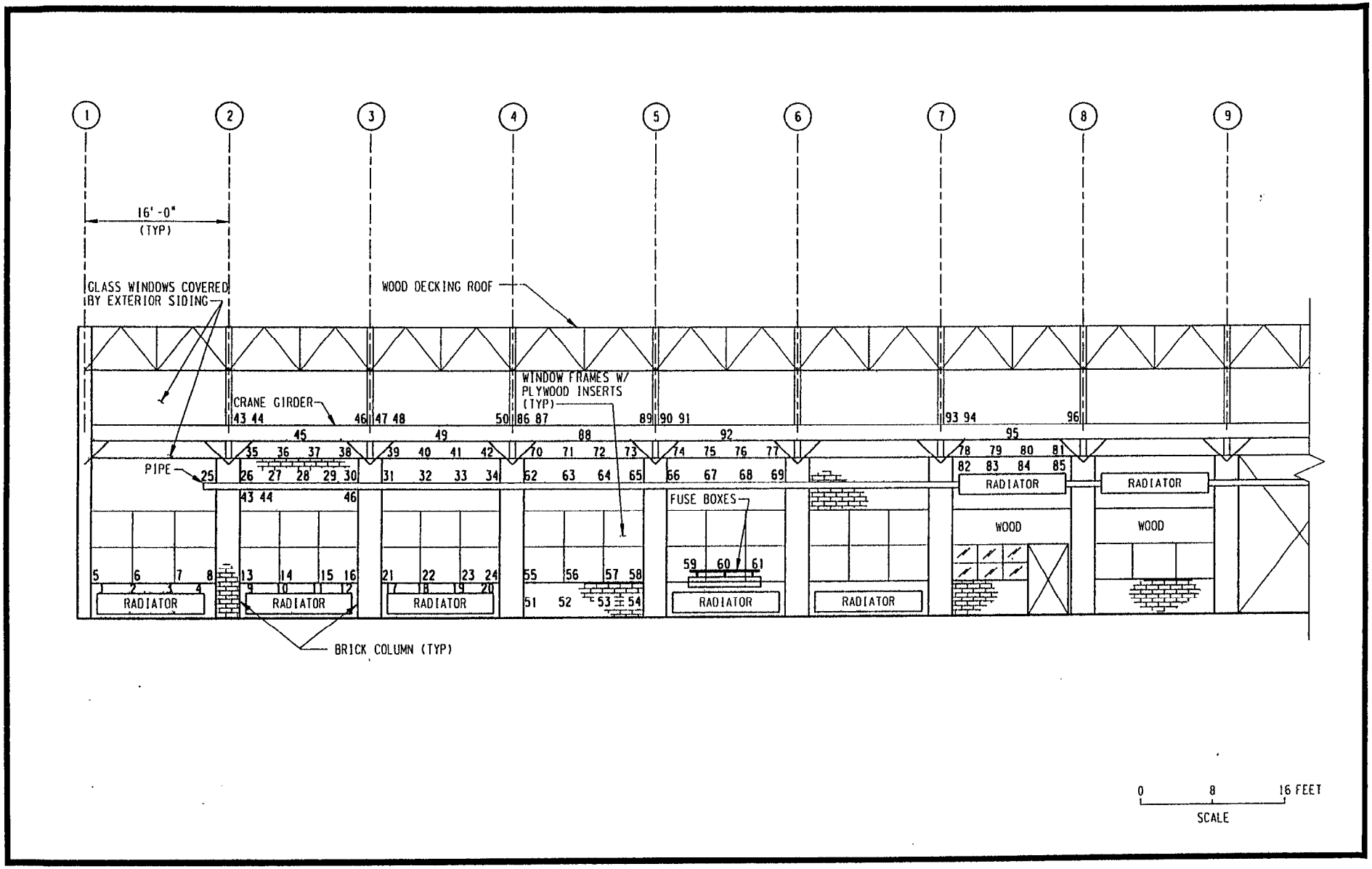
Figure 6
Post Remedial Action Survey Locations
on Over Head Crane

143419



R76F005.DGN

Figure 7
Post Remedial Action Survey Locations
Interior West Wall

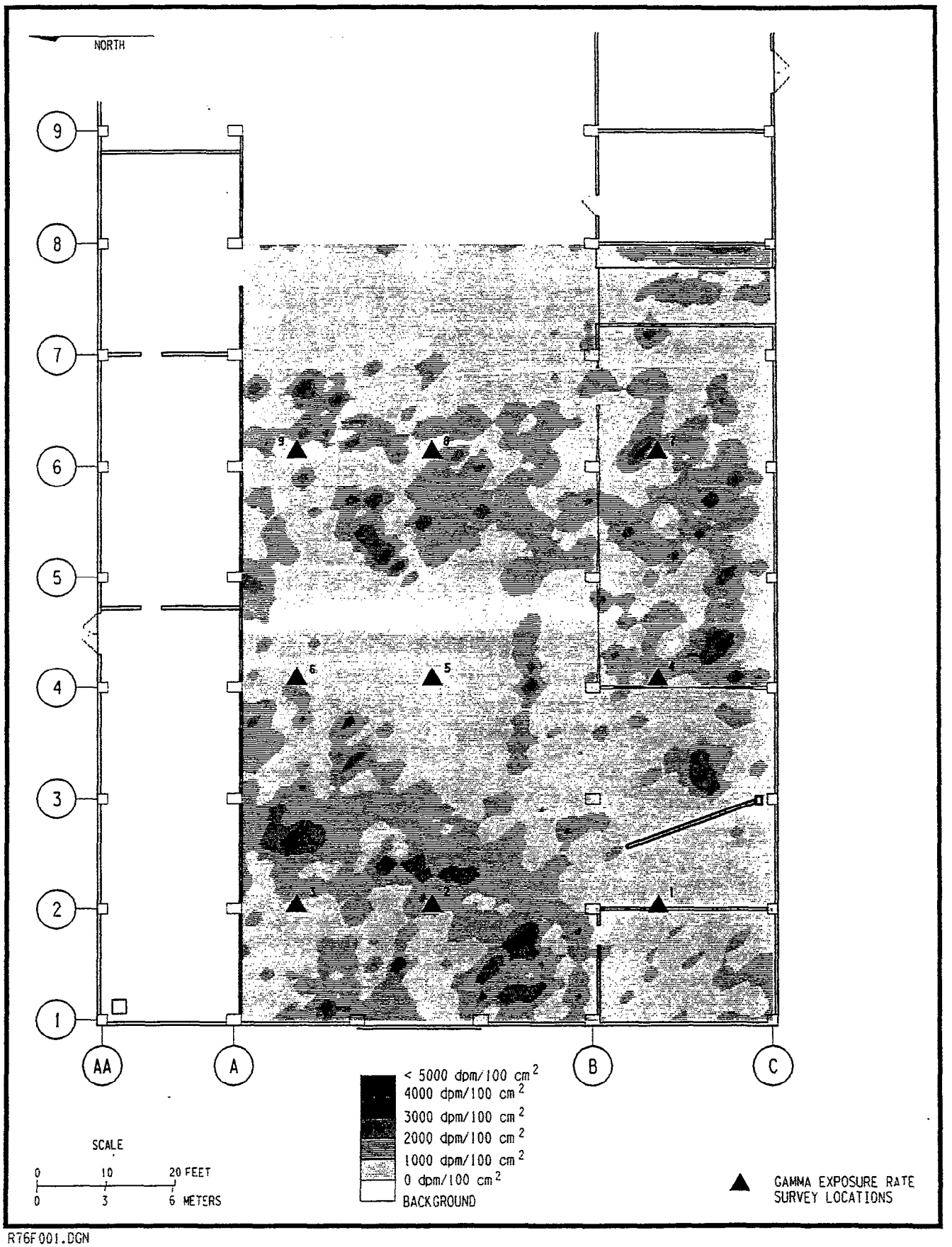


R76F003.DGN

Figure 8
Post Remedial Action Survey Locations
Interior North Wall

14319

143419



R76F001.DGN

Figure 10
Chapman Valve Post RA Floor Survey

Editor's Note:

BEST

AVAILABLE

COPY

Statement of Limitation

Use of the following calculations (Chapman Valve Hazard Assessment No. 133-CV-001) by persons without access to the pertinent factors and without proper regard for their purpose could lead to erroneous conclusion.

Should it become necessary to use this calculation in your work in the future, it is suggested that the calculation be reviewed with authorized Bechtel personnel to ensure that the purposes, assumptions, judgments and limitations are thoroughly understood. Bechtel cannot assume responsibility for use of this calculation not under our direct control.

CALCULATION COVER SHEET

Bechtel National, Inc.

Project: FUSRAP

Job No.: ~~14105~~ 14501 AKG 12/15/95

Discipline: ET

Calc No.: 133-CV-001

Subject: Hazard Assessment for Chapman Valve

Computer Program: N/A

Program No.:

Committed Calculation ☒ Preliminary ☐ Superseded ☐

Rev	Shts	Originator	Checker	Reviewer	Approval	Date
0	14	Carl Efird CRE	[illegible]	DMK	[illegible]	6-2-95
1	26	Andrea George AKG	[illegible]	[illegible]	[illegible]	12/19/95

Summary of Revision

Rev. 1 incorporates post-remedial action data and includes four new scenarios.

Microfilmed:	Rev	Date	Spool Number	Rev	Date	Spool Number

Logged 18W

Calculation Sheet

Originator Andrea George AG Date 12/15/95 Calc. No. 133-CV-001 Rev. No. 1
Project FUSRAP Job No. 14501 Checked 12/18/95 Date 12/18/95
Subject Hazard Assessment for Chapman Valve Site Sheet No. 1 of 26

PURPOSE

This calculation documents the post remediation condition of the Chapman Valve site.

The bases for this assessment are:

- Contamination at the Chapman Valve site is found only on limited portions of Building 23. The total curie content of all affected building components is approximately 6 mCi.
- Building 23 is derelict and virtually certain to be demolished.

REFERENCES

BNI Calculation 133-CV-001, Rev.0, February 1995

DOE, 1983, Pathways Analysis and Radiation Dose Estimates for Radioactive Residues at Formerly Utilized MED/AEC Sites, ORO-832, Oak Ridge, Tenn. (March)

DOE, 1989, A Manual for Implementing Residual Radioactive Material Guidelines, DOE/CH/8901 (June)

DOE, 1990, Order 5400.5, Radiation Protection of the Public and Environment, (June)

FUSRAP Technical Memorandum, No. 133-45-003, Chapman Valve Characterization Results

Schleien, Bernard, ed., 1992. The Health Physics and Radiological Health Handbook, Scinta, Inc. Silver Spring, Maryland.

Sehmel, George A., 1984. "Deposition and Resuspension", Chapter 12 in Atmospheric Science and Power Production edited by Darryl Randerson, DOE/TIC-27601

Waier, Philip R., Senior. Ed., Means Building Construction Cost Data, R.S. Means Company, 1996.

Yu et.al., 1993. Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil, Argonne National Laboratory ANL/EAIS-8 (April)

RESRAD computer code, Version 5.61, Released 8/28/95.

RES-BUILD computer code, Version 1.5, Released 12/15/94.

ASSUMPTIONS

1. Bldg. 23 at the Chapman Valve site is contaminated with processed uranium metal. U-238, U-234, and U-235 are present in their natural activity ratio of 1:1:0.046.

2. Among the principle exposure pathways - inhalation, ingestion, and external radiation exposure - the dose from inhalation far exceeds the contribution from the other pathways. Therefore, except for structural steel, only the dose from inhalation is calculated in determining the supplemental limit for each building component. In the case of the structural steel, the highest amount of hand contact with contaminated material is expected and the dose has been calculated to illustrate that, in the worse case, the dose from subsequent ingestion is well below 1 mrem.

3. Dose conversion factors for inhaled and ingested radioactivity are taken from the RESRAD computer code version 5.61. Dose calculations incorporate the dose from short-lived decay products of U-238 and U-235.

4. If left in place, the likely disposition of Bldg. 23 is demolition by a civilian demolition firm. The maximally exposed individual as a result of residual radioactivity would be a hypothetical demolition worker who participates in all phases of the demolition.

5. The breathing rate for the exposed worker is conservatively assumed to be 20 m^3 per 8 hr shift. (Yu et.al. 1993)

6. Mass loading assumptions are based on the following:

   When dust is produced during normal construction activities, such as handling building rubble, a mass loading factor of 600 $\mu g\ m^{-3}$ is employed (DOE 1983).

   When a high, instantaneous mass loading factor is required - as when the roof or walls are dropped - 10 mg m^{-3} is employed (OSHA PEL for nuisance dust).

7. The airborne contaminant particulates are assumed to have a size of 1 µm activity median aerodynamic diameter (AMAD). This is a conservative assumption designed to maximize the dose to the worker by presuming that all airborne contamination is respirable and will remain suspended a significant amount of time (the settling velocity is taken to be 0.0075 cm^{-1} (Schmel 1984), the high specific gravity of U would probably result in a faster settling velocity).

8. The dose calculations are presented for each phase of a hypothetical demolition in order to highlight the maximum exposure associated with each building component. This approach is conservative in that the likely scenario would entail rapid, mechanical, destruction of the structure (i.e. without individual phases) and subsequent removal of the debris. The latter tactic would, to some extent, quench potential airborne contamination as the structure is dropped.

   The hypothetical demolition phases representing each of the building components and significant conditions/assumptions are as follows.

Roof Demolition: Contamination is consolidated into built-up roofing material. The principle exposure pathway is inhalation of contaminated dust. The primary dust producing activities would be: destroying the roof with heavy equipment (allowing it fall to the ground); and, loading the roof rubble into trucks using heavy equipment. This scenario is plausible but conservative in that the roof contains asbestos and the likely scenario would involve appropriate asbestos controls - those controls would also mitigate release of airborne radioactivity and worker exposure in general.

The contaminated portion of the roof covers an area of 12,567 ft^2, the contaminated roofing material is 0.75 in. (0.065 ft) thick. The density of the roofing material is taken to be 1.5 g cm^{-3} (asphaltic tar).

The average total U concentration in the built-up roofing material is estimated at 171 pCi g^{-1}.

Wall Demolition: Contamination is on the surface of the walls. Post remediation smear samples were all below DOE 5400.5 guidelines. Walls are constructed of 16 inch brick and mortar with large bays of windows. The maximum dust producing activity would involve the use of a wrecking ball (or other forceful demolition technique) where the impact of the ball occurs at approximately 20 feet above ground and dislodges (and makes airborne) some surface contamination on the walls (from brick and glass surfaces) as well as producing dust as clean portions of the brick, mortar, and glass are pulverized. To highlight the maximum exposure, the calculation approach assumes demolition is performed one wall at a time, and the rubble from one wall is removed before the next wall is demolished. The instantaneous mass loading during wall demolition is assumed to be 10 mg m^{-3}. This is a very conservative assumption in that the dust loading is so high that the air could not suspend this mass of material for a significant amount of time. In addition, standards for fugitive dust limit the average concentration to 50 $\mu g\ m^{-3}$. In addition to the initial plume, contaminated dust would be resuspended as rubble is removed. ⚠1

Post remediation surveys indicated the following average fixed contamination levels in Bay 1 through Bay 8. ⚠1

West Wall	19,410 dpm/100 cm^2
North Wall	3,480 dpm/100 cm^2
South Wall	5,922 dpm/100 cm^2

Structural Steel Demolition: Contamination is on the upper horizontal surfaces of the overhead structural steel (trusses) in bays 1-8 starting at the west wall.

Demolition of Building 23 would likely include salvage of the steel. Because of the likelihood of significant contact with contamination by workers as the steel is recovered, decontamination of trusses was performed.

Decontamination was sufficiently rigorous so that no airborne contamination would result from the exposed steel surfaces during demolition. Some areas, such as intersections of steel

members, gusset plates, etc., are more difficult to access and could be the source of airborne contamination as the trusses are removed.

Following decontamination it is assumed that some of the contamination from inaccessible areas will be dislodged as the steel building framework is dismantled. The contamination on the inaccessible surfaces is assumed to be the same as the exposed surfaces surveyed during characterization, which was 3298 dpm/100 cm^2. The contamination is from deposited dust which is assumed to be 0.1 cm thick. The surface area of the contaminated steel is approximately 1542 ft^2. However, an engineering review of the structural layout of Building 23 has disclosed that the surface area of the inaccessible portions of the steel is approximately 44.3 ft^2.

The primary exposure pathways are inhalation and ingestion. The activity generating the most airborne contamination will entail toppling the supporting beams and allowing the trusses to fall to the ground where they will subsequently be salvaged for scrap. It is assumed that torch cutting will be performed to size the metal for transport. Airborne contamination will be released as: (1) the trusses fall to the ground; (2) the trusses are cut for transport; and, (3) the trusses are removed. Assuming all trusses are felled in a very short time frame, the contaminated plume from any one truss falling will not have dissipated when the next truss is dropped; thus, the plume of airborne contamination will embody all the contamination present on the trusses.

Personal contamination (and subsequent ingestion) will occur as the trusses are handled during cutting and removal.

Steel Recycling: Potential exposures from reuse or recycling of the steel were considered for three activities:

- torch cutting of the steel so that it can be of the size necessary for acceptance for melting;
- sand blasting of the steel to remove lead based paint so the smelter will accept the steel
- exposure from the discarded slag after melting

Floor Demolition: The floor of building 23 is composed separately of 3 different building materials; concrete, fire brick, or wood block, each overlaid on a concrete base slab. Over 90% of the floor area is composed of wood blocks. Isolated contamination has been found on the surface of the floor, in adhesive between wood blocks and gaps in fire brick and possibly on the concrete base slab. Remedial activities removed the top layer (wood, brick, concrete) to expose the base slab for remediation to DOE criteria. However, a composite sample made from 26 wood blocks systematically collected had an activity concentration of 1 pCi per gram. This concentration is near background levels and the wood blocks were left at the site after they were removed.

Reuse of Building Materials: The dose that a person could receive if he were to live in a small building constructed with the contaminated materials from CHV was calculated. The building walls are from the west wall, ceiling from the roofing material, and floor from the wood blocks.

CALCULATION

Dose From Roof Demolition

1. The volume of contaminated built-up roofing material is:

$$(12{,}576\ ft^2)(0.0625\ ft) = 7.86 \times 10^2\ ft^3 = 2.23 \times 10^7\ cm^3$$

2. The mass of built-up roofing material is:

$$(2.23 \times 10^7\ cm^3)(1.5\ g\ cm^{-3}) = 3.35 \times 10^7\ g$$

3. The curie content of the contaminated built-up roofing material is:

$$(171\ pCi\ g^{-1})(3.35 \times 10^7\ g) = 5.73 \times 10^9\ pCi$$

4. There is no recognized value to the instantaneous mass loading that would occur as the roof is destroyed and allowed to drop to the floor of the building. An airborne dust concentration of 10 mg m^{-3} depicts severe conditions and would represent a conservative value.

The airborne radioactivity concentration is thus:

$$(0.01\ g\ m\text{-}3)(171\ pCi\ g^{-1}) = 1.71\ pCi\ m^{-3}$$

If it is conservatively assumed that all airborne particulates have an AMAD of 1 μm then the settling of the contaminated plume will be very slow, and a worker who enters the building and begins removal of the contaminated roofing debris soon after the roof is dropped will be exposed to the concentration calculated above.

The magnitude of the exposure will depend on the duration of the work. Because the roof is constructed of 3 in. thick plank in addition to the contaminated built-up roofing material, the volume of rubble associated with the contaminated are of the roof is:

$$(7.86 \times 10^2\ ft^3\ \text{contaminated built-up material}) + (12{,}576\ ft^2 \times 0.25\ ft\ \text{wood}) = 3.93 \times 10^3\ ft^3 = 1.11 \times 10^2\ m^3$$

If a front end loader with a 1.25 m^3 capacity is used to remove the roof rubble, the time required to remove the rubble (during which a worker is exposed) is approximately two 8 hour shifts. 32 hrs elapse, therefore, from the start to finish of the removal of rubble (two 8 hour shifts separated by 16 hours).

It is assumed that all dust would settle out between shifts, therefore, there would be no exposure to the initial plume after the first shift. If it is conservatively assumed that there would be no airborne reduction during the 8 hr shift, the exposure to the worker would be:

$(171\ pCi\ g^{-1})(20\ m^3$ inhaled per 8 hr shift$) = 34.2$ pCi inhaled

In addition to the initial plume of contaminated dust produced as the roof is demolished, there would also be contaminated dust generated during removal of the roof rubble. This activity is comparable to normal construction activities and a standard (although very conservative) MLF of 600 $\mu g\ m^{-3}$ is applicable. Again, assuming the work will require two 8 hr shifts to complete the calculated exposure would be:

$(6 \times 10^{-4}\ g\ m^{-3})(171\ pCi\ g^{-1})(20\ m^3\ shift^{-1})(2\ shifts) = 4.10$ pCi (inhaled)

The total activity from the initial plume and follow-up waste removal is:

34.2 pCi + 4.10 pCi = 38.3 pCi

The resultant dose is:

U-234 = $(19.1\ pCi)(0.13\ mrem\ pCi^{-1}) = 2.48$ mrem
U-235 = $(0.843\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.10$ mrem
U-238 = $(19.1\ pCi)(0.12\ mrem\ pCi^{-1}) = 2.29$ mrem

Total dose = 4.87 mrem

Wall Demolition

1. The concentration of total uranium in wall components following remediation is calculated as follows:

Brick wall:

Thickness: 16 inches (40.64 cm)
Effective density: 1.62 g cm^{-3}
Mass under 100 cm^2 survey area: 6,584 g
Average activity in survey area:

West Wall	19,410 dpm/100 cm^2 (8,823 pCi/100 cm^2)
North Wall	3,480 dpm/100 cm^2 (1,582 pCi/100 cm^2)
South Wall	5,922 dpm/100 cm^2 (2,692 pCi/100 cm^2)

Average concentration:

West Wall	1.33 pCi g^{-1}
North Wall	0.24 pCi g^{-1}
South Wall	0.41 pCi g^{-1}

Glass windows:

Thickness: 1/8 inch (0.32 cm)
Density: 2.6 g cm^{-3}
Mass under 100 cm^2 survey area: 83.2 g
Average activity in survey area same for brick

Average concentration:

West Wall	105.9 pCi g^{-1}
North Wall	19.0 pCi g^{-1}
South Wall	32.3 pCi g^{-1}

The total interior surface area of the West Wall is approximately 1,943 ft^2, of which approximately 923 ft^2 feet is brick and 1020 ft^2 is windows. The masses of these components are:

$$(923\ ft^2 \times 1.33\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(1.62\ g\ cm^{-3})$$
$$= 5.63 \times 10^7 \text{ grams of brick}$$

$$(1020\ ft^2 \times 0.01\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(2.6\ g\ cm^{-3})$$
$$= 7.51 \times 10^5 \text{ grams of glass}$$

The West Wall is thus 98.7% brick and 1.3% glass, and the weighted average concentration is:

$$(0.987[1.3\ pCi\ g^{-1}])+(0.013[105.9\ pCi\ g^{-1}]) = 2.76\ pCi\ g^{-1}$$

The total interior surface of each of the North and South Walls Bays 1 through 8 is approximately 4,224 ft^2, of which approximately 2,304 ft^2 is brick and 1,920 ft^2 is glass. The mass of brick and glass in each of the North and South Walls is:

$$(2304\ ft^2 \times 1.33\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(1.62\ g\ cm^{-3})$$
$$= 1.40 \times 10^8 \text{ grams of brick}$$

$$(1920\ ft^2 \times 0.01\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(2.6\ g\ cm^{-3})$$
$$= 1.41 \times 10^6 \text{ grams of glass}$$

The North and South Walls are thus 99.0% brick and 1.0% glass and the weighted average concentrations of U are therefore:

North Wall

$$(0.990[0.24\ pCi\ g^{-1}])+(0.010[19.0\ pCi\ g^{-1}]) = 0.43\ pCi\ g^{-1}$$

South Wall

$$(0.990[0.41\ pCi\ g^{-1}])+(0.010[32.3\ pCi\ g^{-1}]) = 0.74\ pCi\ g^{-1}$$

The instantaneous mass loading factor is 10 mg m^{-3}; therefore the initial airborne concentration is:

West Wall (2.76 pCi g^{-1})(0.01 g m^{-3}) = 0.028 pCi m^{-3}
North Wall (0.43 pCi g^{-1})(0.01 g m^{-3}) = 0.004 pCi m^{-3}
South Wall (0.74 pCi g^{-1})(0.01 g m^{-3}) = 0.007 pCi m^{-3}

2. The dose to the hypothetical maximally exposed individual will depend on the duration of exposure. For a particle size of 1 μm AMAD, the settling velocity is 0.0075 cm s^{-1}, and assuming the contamination plume extends upward 20 ft (609.6 cm), the time required for virtually all of the particulates to settle out is:

0.0075 cm/1 s = 609.6 cm/x s
x = 81,280 s = 22.6 hrs

It is assumed that, although the settling time is long, the plume would not be present after the 16 hours between work shifts because of the dispersive effects of winds.

3. The highest exposure would occur if demolition took place at the beginning of the work shift, the airborne contamination is assumed to be uniformly distributed throughout the plume, and rubble is removed before the plume dissipates. The duration of exposure will depend on the volume of rubble to be removed and the rate it is removed. A work crew with a front loader with a 1.25 m^3 capacity can load 80 m^3 per hour (Waier 1994). Because of the irregular sizing of the building rubble, it is plausible that the work may go slower and a conservative value of 40 m^3 per hr is used. The approximate quantities of rubble are:

West Wall

(923 ft^2 x 1.33 ft thick brick) = 1228 ft^3 (34.8 m^3)
(1020 ft^2 x 0.01 ft thick windows) = 10.2 ft^3 (0.3 m^3)

South and North Walls (each)

(2304 ft^2 x 1.33 ft thick brick) = 3064 ft^3 (86.8 m^3)
(1920 ft^2 x 0.01 ft thick glass) = 19.2 ft^3 (0.5 m^3)

The duration of the cleanup of the rubble from the West Wall is:

(35.1 m^3)÷(40 m^3 hr^{-1}) ≈ 1 hr

The duration of cleanup of the rubble from each of the North and South Walls are:

(87.3 m^3)÷(40 m^3 hr^{-1}) ≈ 2.5 hrs each

4. The inhalation intakes are:

West Wall (0.028 pCi m^{-3})(2.5 m^3 hr^{-1})(1 hr) = 0.069 pCi
North Wall (0.004 pCi m^{-3})(2.5 m^3 hr^{-1})(2.5 hr) = 0.027 pCi
South Wall (0.007 pCi m^{-3})(2.5 m^3 hr^{-1})(2.5 hr) = 0.046 pCi

The total radioactivity intake is:

0.069 pCi + 0.027 pCi + 0.046 pCi = 0.142 pCi

And the dose is:

U-234 = (0.069 pCi)(0.13 mrem pCi^{-1}) = 0.009 mrem
U-235 = (0.003 pCi)(0.12 mrem pCi^{-1}) = 0.0004 mrem
U-238 = (0.069 pCi)(0.12 mrem pCi^{-1}) = 0.0083 mrem

Dose from Wall Demolition = 0.018 mrem

Structural Steel Demolition

1. The contaminated area on the steel trusses is estimated to be 1542 ft^2, and of this, 44.3 ft^2 is inaccessible.

2. Post-remedial surveys show that the removable contamination was successfully removed to below the DOE Order 5400.5 limits of 1000 dpm/100 cm^2. Therefore the only contamination considered is the fixed contamination, some of which is in inaccessible areas, which is assumed to be similar to that measured in the accessible areas during the characterization.

The average fixed contamination was measured to be:

3298 dpm/100 cm^2 = 1486 pCi/100 cm^2

The contamination on the steel is from settled dust. Assuming the dust is 0.1 cm thick and has a density similar to soil, i.e. ~1.6 g cm^{-3}. Therefore, the concentration of the contamination is:

(14.86 pCi/cm^2)/(0.1 cm)/(1.6 g cm^{-3}) = 93 pCi g^{-1}

3. The intake of radioactivity will depend on the duration of exposure and the concentration of contaminant. An engineering appraisal of the building plans for Building 23 have disclosed that there is approximately 52.5 tons of over head supporting steel in bays 1-8. For this quantity of structural metal, salvage operations will take approximately five 8 hour shifts.

A conservative mass loading factor of 600 μg m^{-3} (DOE 1983) for construction activities would result in the following intakes of uranium.

U-234 $(45.4\ pCi\ g^{-1})(6 \times 10^{-4}\ g\ m^{-3})(5 \times 20\ m^3) = 2.72\ pCi$
U-235 $(2.1\ pCi\ g^{-1})(6 \times 10^{-4}\ g\ m^{-3})(5 \times 20\ m^3) = 0.13\ pCi$
U-238 $(45.4\ pCi\ g^{-1})(6 \times 10^{-4}\ g\ m^{-3})(5 \times 20\ m^3) = 2.72\ pCi$

4. The resultant dose is:

U-234 = $(2.72\ pCi)(0.13\ mrem\ pCi^{-1}) = 0.354\ mrem$
U-235 = $(0.13\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.015\ mrem$
U-238 = $(2.72\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.327\ mrem$

Total dose = 0.70 mrem

5. The intake of uranium by ingestion can be estimated by assuming a worker will ingest 480 mg of dust and dirt per day (8 hour shift).

The work takes approximately five 8 hour work shifts (Waier 1996); therefore, the total intake of dust and dirt is: 5 shifts x 480 mg per shift = 2.4 g

The resultant radioactivity intake is thus:

U-234 = $(45.4\ pCi\ g^{-1})(2.4\ g) = 109\ pCi$
U-235 = $(2.1\ pCi\ g^{-1})(2.4\ g) = 5\ pCi$
U-238 = $(45.4\ pCi\ g^{-1})(2.4\ g) = 109\ pCi$

6. The resultant dose is:

U-234 = $(109\ pCi)(2.8 \times 10^{-4}\ mrem\ pCi^{-1}) = 0.031\ mrem$
U-235 = $(5\ pCi)(2.7 \times 10^{-4}\ mrem\ pCi^{-1}) = 0.0014\ mrem$
U-238 = $(109\ pCi)(2.7 \times 10^{-4}\ mrem\ pCi^{-1}) = 0.029\ mrem$

Total dose from ingestion = 0.061 mrem

7. The total dose to a demolition worker from overhead steel contamination is thus 0.76 mrem.

Structural Steel Recycling

Three exposure scenarios were identified for recycling of the steel trusses. These were torch cutting of the steel, sandblasting of the steel (to remove lead based paint), and exposure to slag after melting.

1. Torch cutting of the trusses

The average surface contamination of the trusses (bays 1-8) = 3298 dpm/100 cm^2.
Contaminated area involved in a torch cut = 0.5 cm wide x 10 in long = 12.7 cm^2.
Fraction of activity in cut area assumed to become airborne = 1%
Volume of air to average the activity over = 1 m^3

Assuming the torch cuts are made one at a time along the length of each truss, the welder would be exposed to the concentration of a single cut for the time it would take to make all the cuts.

The airborne concentration for a cut is

U-234 = (1486 pCi/100 cm^2)(0.489)(12.7 cm^2)(0.01)/(1 m^3) = 0.922 pCi/m^3
U-235 = (1486 pCi /100 cm^2)(0.022)(12.7 cm^2)(0.01)/(1 m^3) = 0.042 pCi/m^3
U-238 = (1486 pCi /100 cm^2)(0.489)(12.7 cm^2)(0.01)/(1 m^3) = 0.922 pCi/m^3

The estimated number of cuts per truss = 68
The time to complete each cut = 1 min
Number of contaminated trusses = 8

The time to make all cuts = (68)(1 min)(8) = 9.1 hours

The inhalation dose:

U-234 = (20 m^3/8 hr)(9.1 hr)(0.922 pCi/m^3)(0.13 mrem/pCi) = 2.72 mrem
U-235 = (20 m^3/8 hr)(9.1 hr)(0.042 pCi/m^3)(0.12 mrem/pCi) = 0.12 mrem
U-238 = (20 m^3/8 hr)(9.1 hr)(0.922 pCi/m^3)(0.12 mrem/pCi) = 2.51 mrem

The total dose is 5.34 mrem.

2. Sand blasting of trusses to remove lead paint

The average surface contamination of the trusses (bays 1-8) = 3298 dpm/100 cm^2.
Sandblasting production rate = 0.024 hr/ft^2 (p.149, Waier 1996)
Surface area of contaminated steel = 1542 ft^2

Time to sandblast trusses = (0.024 hr/ft^2)(1542 ft^2) = 37 hours or 4.6 shifts

Assume maximum dust loading of 10 mg/m^3 because this is the maximum nuisance dust loading allowed before a worker must begin wearing a respirator. It is also assumed that 100% of the nuisance dust is radioactively contaminated at the concentrations calculated in the demolition section.

The inhalation dose is calculated to be:

U-234 = (20 m^3/8 hr)(37 hr)(45.4 pCi/g)(10 mg/m^3)(0.13 mrem/pCi) = 5.46 mrem
U-235 = (20 m^3/8 hr)(37 hr)(2.1 pCi/g)(10 mg/m^3)(0.12 mrem/pCi) = 0.23 mrem
U-238 = (20 m^3/8 hr)(37 hr)(45.4 pCi/g)(10 mg/m^3)(0.12 mrem/pCi) = 5.04 mrem

Total dose = 10.7 mrem

3. Slag from melting recycled trusses

Uranium contamination in steel tends to be concentrated in the slag when the steel is melted. SEG personnel indicated from their experience in melting uranium contaminated steel, 99% of the uranium and 3% of the steel end up in the slag. The slag is often used as roadbed material which is usually under at least 6 inches of gravel and asphalt. Higher exposure would result from slag dumped on the ground in a waste area.

Total activity in the trusses (bays 1-8) = (3298 dpm/100 cm^2)(1542 ft^2) = 21.3 µCi
Concentration in the slag = (21.3 µCi)(0.99) / (52.5 tons) / (0.03) = 14.7 pCi/gm
Density of slag is assumed to be 1/2 that of steel = (7.86 gm/cm^3) / 2 =3.93 gm/cm^2
Since the slag is primarily a solid mass, the mass loading is assumed to be 20 µg/m^3
A worker is expected to be exposed to the waste slag approximately 10% of the time

The inhalation dose is

U-234 = (20 m^3/8 hr)(0.1 yr)(20 g/m^3)(14.7 pCi/g)(0.489)(0.13 mrem/pCi) = 0.0092 mrem
U-235 = (20 m^3/8 hr)(0.1 yr)(20 g/m^3)(14.7 pCi/g)(0.022)(0.12 mrem/pCi) = 3.9x10^{-4} mrem
U-238 = (20 m^3/8 hr)(0.1 yr)(20 g/m^3)(14.7 pCi/g)(0.489)(0.12 mrem/pCi) = 0.0086 mrem

The ingestion dose is, using an ingestion rate of 100 mg/day:

U-234 = (100 mg/day)(0.1 yr)(14.7 pCi/g)(0.489)(2.8x10^{-4} mrem/pCi) = 0.005 mrem
U-235 = (100 mg/day)(0.1 yr)(14.7 pCi/g)(0.022)(2.7x10^{-4} mrem/pCi) = 0.0002 mrem
U-238 = (100 mg/day)(0.1 yr)(14.7 pCi/g)(0.489)(2.7x10^{-4} mrem/pCi) = 0.0048 mrem

The external dose is

U-234 = (0.1 yr)(3.93 g/cm^2)(14.7 pCi/g)(0.489)(6.97x10^{-10} mrem/yr / pCi/m^3) = 0.0019 mrem
U-235 = (0.1 yr)(3.93 g/cm^2)(14.7 pCi/g)(0.022)(4.90x10^{-7} mrem/yr / pCi/m^3) = 0.062 mrem
U-238 = (0.1 yr)(3.93 g/cm^2)(14.7 pCi/g)(0.489)(6.97x10^{-8} mrem/yr / pCi/m^3) = 0.19 mrem

The total exposure is 0.286 mrem

Removal of Floor Blocks

1. The floor blocks will have been stockpiled in the building after their removal from the floor. The demolition workers are assumed to remove them similarly to the other building rubble at a rate of approximately 40 m^3 per hour. The volume of wood blocks has been estimated to be 150 m^3, which would be removed in:

   $(150\ m^3)/(40\ m^3\ h^{-1}) \cong 4$ hours

2. Using the conservative mass loading factor of 600 mg m-3 and the contamination concentration for the blocks as 1 pCi/g, the airborne concentration of uranium would be:

   $(1\ pCi/g)(6 \times 10^{-4}\ g\ m^{-3}) = 6 \times 10^{-4}\ pCi\ m^{-3}$

   The corresponding radioactivity intake is:

   $(6 \times 10^{-4}\ pCi\ m^{-3})(2.5\ m^3\ hr^{-1})(4\ hr) = 0.006\ pCi$

3. And the dose is

   U-234 = $(0.0029\ pCi)(0.13\ mrem\ pCi^{-1}) = 0.0004$ mrem
   U-235 = $(0.0001\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.00001$ mrem
   U-238 = $(0.0029\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.0003$ mrem

   Total dose = 0.0007 mrem

Reuse of Building Materials

RESRAD-BUILD version 1.5 was used to evaluate the dose that a person could receive if he were to live in a small building constructed with the contaminated materials from Chapman Valve. The building walls are from the west wall, ceiling from the roofing material, and floor from the wood blocks. The annual dose was calculated to be 17.3 mrem/year and the input and output are in Attachment 2.

Contamination in Building Rubble

1. To estimate the total radioactive contamination expected in the building rubble that would be disposed of following demolition, the radioactive concentrations used to estimate the dose to the demolition worker is assumed to be in the rubble.

Component	Contamination	Amount	Total Activity
Roof Material	171 pCi/g	3.35×10^7 g	5.73×10^9 pCi
West Wall	2.76 pCi/g	5.71×10^7 g	1.58×10^8 pCi
North Wall	0.43 pCi/g	1.41×10^8 g	6.06×10^7 pCi
South Wall	0.74 pCi/g	1.41×10^8 g	1.04×10^8 pCi
Steel	14.9 pCi/cm^2	4.12×10^4 cm^2	6.14×10^5 pCi
Floor Blocks	1 pCi/g	8.81×10^7 g	8.81×10^7 pCi

Total radioactivity = 6.14×10^9 pCi = 6.14 mCi

Assuming that rubble from the contaminated areas is 1/3 of the entire building, the average concentration for the building rubble is calculated as:

$$6.14 \times 10^9 \text{ pCi} / (3 \times 4.64 \times 10^8 \text{ g}) = 4.41 \text{ pCi/g}$$

This concentration is well below the uranium soil contamination guideline range 35 to 100 pCi per gram typically used with sites similar to Chapman Valve. In addition, it is comparable to the natural radioactivity found in many common building materials. Attachment 1 contains a table which lists the natural radioactivity concentrations found in common building materials.

The values of the individual uranium isotopes would therefore be:

U-234 = 3.0 mCi
U-235 = 0.14 mCi
U-238 = 3.0 mCi

These values are well below the EPA's Reportable Quantities (RQs) for U-234, U-235, and U-238 specified in 40 CFR 302.

Using the specific activity of the uranium isotopes, the mass of the uranium present is

U-234 = (3.02 mCi)/(6.19 mCi/g) = 0.5 g
U-235 = (0.14 mCi)/(2.14×10^{-5} mCi/g) = 65.4 g
U-238 = (3.02 mCi)/(3.33×10^{-4} mCi/g) = 9069 g

The total mass of uranium is:

$$U = 9135 \text{ g} = 20.1 \text{ lbs}$$

This value is well below EPA's RQ values for uranium compounds (uranyl acetate, uranyl nitrate) with respect to chemical hazards.

CONCLUSIONS

Contamination on the roof and walls of Building 23 will not result in a radiation dose above 100 mrem to any member of the general public. The maximum dose to a hypothetical demolition worker is estimated to be 5.6 mrem, and offsite exposure to the general public would be much less. Brick comprises approximately 64% of the mass of the contaminated portion of Building 23. The natural (background) radioactivity found in brick at the site is approximately 6.4 pCi per gram. The amount of radioactivity added by the surface contamination ranged from 0.18 to 1.5 pCi per gram; thus, the contamination has only increased the radioactivity naturally found in brick less than 23% Given the low radiation dose calculated to result from roof and wall contamination (and the fact that these calculations were very conservative and likely over-estimated the dose) supplemental limits for these building components should be the same as the contamination levels there at present and no remedial action need be performed.

Since no abrasive decontamination was performed on overhead steel, scenarios to model a hypothetical recycling worker were evaluated. The maximum dose to a hypothetical recycling worker cutting the steel into pieces, sandblasting it to remove the paint, and then melting the steel for recycling is estimated to be 16 mrem. These calculations are very conservative and likely over-estimate the dose.

Attachment 1

Naturally Occurring Radiation in Building Materials and Solid Waste

Material		U-238 (pCi/g)	Ra-226 (pCi/g)	Th-232 (pCi/g)	Reference
Red Brick		1.2		1.8	A
Concrete		0.9			A
Clay Brick		3.0	14	1.2	A
Silicate Brick		0.2	0.2	0.1	A
Granite		6.0	2.4	2.2	A
Natural Gypsum		0.4	0.6	0.2	A
Manufactured Gypsum	C		30 - 40		B
Concrete Block (Fly Ash)	C	1 - 12	0.2 - 4.0	1.0 - 1.2	A
Cement		1.1		0.4	A
Silica Sand		0.3		0.5	A
Photo Materials	C	0.07 - .34			A
Coal	C	0.07 - 8.4		0.21 - 19	A
Fly Ash - Anthracite		3.0			B
Fly Ash - Eastern Bituminous		12.7			B
Fly Ash - Western Bituminous		5.4			B
Fly Ash - Lignite		7.4			B
Fertilizer		D	5 - 6		A

A Eisenbud, Merril (1987), "Environmental Radioactivity From Natural, Industrial, and Military Sources," Academic Press, Inc.., New York.

B National Council on Radiation Protection and Measurements, "Exposure From the Uranium Series with Emphasis on Radon and its Daughters," NCRP Report 77, 1983.

C Values shown are ranges

D Reference does not give activities but does state that in 1974 about 26 million tons of ore (phosphate rock) mined for fertilizer production contained about 1,000 Ci of radium-226 in equilibrium with uranium-238 and 3,000 Ci of potassium-40 are added to the soil during fertilizer application in the US annually.

Attachment 2

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 0 :    1 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT
```

RESRAD-BUILD Table of Contents

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  Time Specific Parameters..... I-1
  Receptor-Source Dose Summary..... I-2
  Dose by Pathway Detail..... I-3
  Dose by Nuclide Detail..... I-4
Full Summary..... F-1

** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 1 : 2 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT

RESRAD-BUILD Input Parameters

Number of Sources : 6
Number of Receptors: 1
Total Time : 3.650000E+02 days
Fraction Inside : 5.000000E-01

Receptor Information

Receptor	Room	x [m]	y [m]	z [m]	FracTime	Inhalation [m3/day]	Ingestion(Dust) [m2/hr]
1	1	3.000	3.000	1.000	1.000	2.00E+01	1.00E-04

Receptor-Source Relationship

Receptor	Source 1 Density [g/cm3]	Source 1 Thickness [cm]	Source 2 Density [g/cm3]	Source 2 Thickness [cm]	Source 3 Density [g/cm3]	Source 3 Thickness [cm]	Source 4 Density [g/cm3]	Source 4 Thickness [cm]	Source 5 Density [g/cm3]	Source 5 Thickness [cm]
1	2.400E+00	0.000E+00	1.600E+00	0.000E+00	1.600E+00	0.000E+00	1.600E+00	0.000E+00	1.600E+00	0.000E+00

Receptor	Source 6 Density [g/cm3]	Source 6 Thickness [cm]
1	1.600E+00	0.000E+00

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 2 :    3 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT
```

Building Information

Building Air Exchange Rate: 8.00E-01 1/hr

```
Height[m]          Air Exchanges [m3/hr]
Area [m2]
                   ******************************
                   *                            *
                   *                            *
                   *                            <=Q01: 7.20E+01
H1:   2.500        *          Room  1           *      Q10 :  7.20E+01
                   *    LAMBDA: 8.00E-01        *
Area  36.000       *                            *
                   *                            *
                   ******************************
```

Deposition velocity: 1.00E-02 [m/s] Resuspension Rate: 5.00E-07 [1/s]

** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 3 : 4 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT

Source Information

Source: 1
Location:: Room : 1 x: 3.00 y: 0.00 z: 1.75[m]
Geometry:: Type: Area Area:1.50E+01 [m2] Direction: y
Pathway ::
Direct Ingestion Rate: 0.000E+00 [1/hr]
Fraction released to air: 1.000E-02
Removable fraction: 1.000E+00
Time to Remove: 9.125E+03 [day]

Radon Release Fraction: 1.000E-01

Contamination::

Nuclide	Concentration [pCi/m2]	Dose Conversion Factors: Ingestion [mrem/pCi]	Inhalation [mrem/pCi]	External (Surface) [mrem/yr/(pCi/m2)]	External (Volume) [mrem/yr/(pCi/m3)]	Submersion [mrem/yr/(pCi/m3)]
U-238	4.273E+05	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	1.966E+04	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	4.273E+05	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-230	0.000E+00	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-226	0.000E+00	1.100E-03	7.900E-03	1.940E-04	7.000E-06	1.040E-02
PB-210	0.000E+00	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 3 :    5 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT


Source:  2
        Location::  Room :  1  x:   3.00 y:   6.00  z:   1.75[m]
        Geometry::  Type: Area        Area:1.50E+01 [m2]  Direction: y
        Pathway ::
            Direct Ingestion Rate:    0.000E+00 [1/hr]
            Fraction released to air: 1.000E-02
            Removable fraction:       1.000E+00
            Time to Remove:           9.125E+03 [day]

            Radon Release Fraction:   1.000E-01
```

Contamination::

Nuclide	Concentration [pCi/m2]	Dose Conversion Factors: Ingestion [mrem/pCi]	Inhalation [mrem/pCi]	External (Surface) [mrem/yr/(pCi/m2)]	External (Volume) [mrem/yr/(pCi/m3)]	Submersion [mrem/yr/(pCi/m3)]
U-238	4.273E+05	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	1.966E+04	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	4.273E+05	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-230	0.000E+00	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-226	0.000E+00	1.100E-03	7.900E-03	1.940E-06	7.000E-06	1.040E-02
PB-210	0.000E+00	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 3 :    6 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT


Source:  3
        Location::  Room :  1  x:   0.00 y:   3.00  z:   1.75[m]
        Geometry::  Type: Area        Area:1.50E+01 [m2]  Direction: x
        Pathway ::
             Direct Ingestion Rate:     0.000E+00 [1/hr]
             Fraction released to air: 1.000E-02
             Removable fraction:        1.000E+00
             Time to Remove:            9.125E+03 [day]

             Radon Release Fraction:    1.000E-01
```

Contamination::

Nuclide	Concentration [pCi/m2]	Dose Conversion Factors: Ingestion [mrem/pCi]	Inhalation [mrem/pCi]	External (Surface) [mrem/yr/(pCi/m2)]	External (Volume) [mrem/yr/(pCi/m3)]	Submersion [mrem/yr/(pCi/m3)]
U-238	4.273E+05	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	1.966E+04	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	4.273E+05	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-230	0.000E+00	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-226	0.000E+00	1.100E-03	7.900E-03	1.940E-04	7.000E-06	1.040E-02
PB-210	0.000E+00	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 3 :    7 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT


Source:  4
        Location::  Room :  1  x:   6.00 y:   3.00  z:   1.75[m]
        Geometry::  Type: Area        Area:1.50E+01 [m2]  Direction: x
        Pathway ::
             Direct Ingestion Rate:     0.000E+00 [1/hr]
             Fraction released to air: 1.000E-02
             Removable fraction:        1.000E+00
             Time to Remove:            9.125E+03 [day]

             Radon Release Fraction:    1.000E-01
```

Contamination::

Nuclide	Concentration [pCi/m2]	Dose Conversion Factors: Ingestion [mrem/pCi]	Inhalation [mrem/pCi]	External (Surface) [mrem/yr/(pCi/m2)]	External (Volume) [mrem/yr/(pCi/m3)]	Submersion [mrem/yr/(pCi/m3)]
U-238	4.273E+05	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	1.966E+04	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	4.273E+05	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-230	0.000E+00	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-226	0.000E+00	1.100E-03	7.900E-03	1.940E-04	7.000E-06	1.040E-02
PB-210	0.000E+00	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 3 :    8 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT


Source:  5
        Location::  Room :  1  x:   3.00 y:   3.00  z:  -0.05[m]
        Geometry::  Type: Volume      Area:3.60E+01 [m2]  Direction: z
        Pathway ::
            Direct Ingestion Rate:     0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-02


      Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+01
    Density [g/cm3]        :9.00E-01
    Erosion Rate [cm/day]  :2.74E-05
    Porosity               :1.00E-01
    Eff. Diffusion [m2/s]  :2.00E-05
    Emanation Fractions(1):2.00E-01
                       (2):2.00E-01


  Contamination::
   Nuclide Concentration                    Dose Conversion Factors
```

		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
U-238	1.000E+00	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	6.000E-02	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	1.000E+00	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-230	0.000E+00	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-226	0.000E+00	1.100E-03	7.900E-03	1.940E-04	7.000E-06	1.040E-02
PB-210	0.000E+00	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:49 Page: 0- 3 : 9 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT

Source: 6
Location:: Room : 1 x: 3.00 y: 3.00 z: 2.51[m]
Geometry:: Type: Volume Area:3.60E+01 [m2] Direction: z
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-02

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:2.00E+00
Density [g/cm3]	:1.50E+00
Erosion Rate [cm/day]	:2.74E-06
Porosity	:1.00E-02
Eff. Diffusion [m2/s]	:2.00E-05
Emanation Fractions(1)	:2.00E-01
(2)	:2.00E-01

Contamination::

Nuclide	Concentration [pCi/g]	Dose Conversion Factors: Ingestion [mrem/pCi]	Inhalation [mrem/pCi]	External (Surface) [mrem/yr/(pCi/m2)]	External (Volume) [mrem/yr/(pCi/m3)]	Submersion [mrem/yr/(pCi/m3)]
U-238	8.300E+01	2.500E-04	1.200E-01	3.530E-06	9.510E-08	1.600E-04
U-235	4.400E+00	2.500E-04	1.200E-01	1.950E-05	4.740E-07	9.030E-04
U-234	8.320E+01	2.600E-04	1.300E-01	8.750E-08	2.520E-10	8.930E-07
PA-231	0.000E+00	1.100E-02	1.300E+00	4.760E-06	1.190E-07	2.010E-04
TH-230	0.000E+00	5.300E-04	3.200E-01	8.780E-08	7.570E-10	2.040E-06
AC-227	0.000E+00	1.500E-02	6.700E+00	4.530E-05	1.260E-06	2.160E-03
RA-226	0.000E+00	1.100E-03	7.900E-03	1.940E-04	7.000E-06	1.040E-02
PB-210	0.000E+00	6.700E-03	2.100E-02	4.140E-07	3.820E-09	1.430E-05

```
** RESRAD-BUILD Program Output, Version 1.50 12/06/95 10:54 Page: F- 1 :   39 **
Title : Chapman Valve - post remediation data
Input File : CVW2.DAT
```

RESRAD-BUILD Dose (Time) Tables

Receptor Doses By Time

[mrem]

	Time [yr]		
	0.00E+00	1.00E+00	1.00E+01
1	1.73E+01	1.73E+01	1.73E+01

Receptor Doses Per Year

[mrem/yr]

	Time [yr]		
	0.00E+00	1.00E+00	1.00E+01
1	1.73E+01	1.73E+01	1.73E+01

Bechtel

Calculation Sheet

Originator Carl Efird Date 2/25/95 Calc. No. 133-CV-001 Rev. No. 0
Project FUSRAP Job No. 14501 Checked RLH Date 5/30/95
Subject Hazard Assessment for Chapman Valve Site Sheet No. 1 of 15

PURPOSE

These calculations are intended to support a preliminary hazard assessment to determine whether adopting supplemental cleanup criteria at the Chapman Valve site will provide a level of protection comparable to that expected from the use of standard cleanup criteria.

In order to release a site for unrestricted use, DOE Order 5400.5 specifies that natural uranium metal surface contamination shall not exceed 5,000 dpm/100 cm^2 averaged over 1 square meter (15,000 dpm/100 cm^2 maximum) for fixed contamination and 1000 dpm/100 cm^2 for removable contamination. In addition to surface contamination limits, limits for the concentration of uranium in soil are specified on a site-by-site basis. For sites similar to Chapman Valve, this limit is typically in the range of 35 to 100 pCi per gram total uranium. These limits are intended to maintain the radiation dose to any member of the general public to below 100 mrem per year. Within this dose limit, DOE Order 5400.5 has adopted the philosophy that radiation doses shall be maintained As Low As Reasonably Achievable (ALARA).

An evaluation of the conditions at the Chapman Valve site, suggests that the use of release criteria other than those specified in DOE Order 5400.5 (i.e. supplemental limits) are capable of not only limiting the dose to any member of the general public to below 100 mrem per year, but also to maintain doses sufficiently low as to meet the goal of the ALARA philosophy. The basis for this assessment are:

- Contamination at the Chapman Valve site is found only on limited portions of Building 23. The total curie content of all affected building components is approximately 6 mCi.
- Building 23 is derelict and virtually certain to be demolished. The dose to a potential building occupant need not be considered.

In preparing this preliminary hazard assessment, supplemental limits are considered independently for each of the affected building components. The total dose resulting from the use of these limits is then compared with the limit specified in DOE Order 5400.5. In calculating the values that the supplemental limits should have for the roof and walls, the dose resulting from no remedial action is first estimated, if this dose is sufficiently below 100 mrem per year to meet the ALARA goal, then the existing contamination level is specified as the supplemental limit. If the estimated dose exceeds 100 mrem or does not satisfy ALARA, then the amount of decontamination required to meet these requirements is calculated and the corresponding contamination level is specified as the supplemental limit.

REFERENCES

DOE, 1983, Pathways Analysis and Radiation Dose Estimates for Radioactive Residues at Formerly Utilized MED/AEC Sites, ORO-832, Oak Ridge, Tenn. (March)

DOE, 1989, A Manual for Implementing Residual Radioactive Material Guidelines, DOE/CH/8901 (June)

DOE, 1990, Order 5400.5, Radiation Protection of the Public and Environment, (June)

Schleien, Bernard, ed., 1992. The Health Physics and Radiological Health Handbook, Scinta, Inc. Silver Spring, Maryland.

FUSRAP Technical Memorandum, No. 133-45-003, Chapman Valve Characterization Results

Waier, Philip R., Senior. Ed., Means Building Construction Cost Data, R.S. Means Company, 1994.

ASSUMPTIONS

1. Bldg. 23 at the Chapman Valve site is contaminated with processed uranium metal. U-238, U-234, and U-235 are present in their natural activity ratio of 1:1:0.046.

2. Among the principle exposure pathways - inhalation, ingestion, and external radiation exposure - the dose from inhalation far exceeds the contribution from the other pathways. Therefore, except for structural steel, only the dose from inhalation is calculated in determining the supplemental limit for each building component. In the case of the structural steel the highest amount of hand contact with contaminated material is expected and the dose has been calculated to illustrate that in the worse case, the dose from subsequent ingestion is well below 1 mrem.

3. Dose conversion factors for inhaled and ingested radioactivity are taken from the RESRAD computer code version 5.19. Dose calculations incorporate the dose from short-lived decay products of U-238 and U-235.

4. If left in place, the likely disposition of Bldg. 23 is demolition by a civilian demolition firm. The maximally exposed individual as a result of residual radioactivity would be a hypothetical demolition worker who participates in all phases of the demolition.

5. The breathing rate for the exposed worker is conservatively assumed to be 20 m^3 per 8 hr shift.

6. Mass loading assumptions are based on the following:

   When dust is produced during normal construction activities, such as handling building rubble, a mass loading factor of 600 $\mu g\ m^{-3}$ is employed (DOE 1983).

   When a high, instantaneous mass loading factor is required - as when the roof or walls are dropped - 10 $mg\ m^{-3}$ is employed (OSHA PEL for nuisance dust).

7. The airborne contaminant particulates are assumed to have a size of 1 µm AMAD. This is a conservative assumption designed to maximize the dose to the worker by presuming that all airborne contamination is respirable and will remain suspended a significant amount of time (the

settling velocity is taken to be 0.0075 cm^{-1}, the high specific gravity of U would probably result in a faster settling velocity).

8. The dose calculations are presented for each phase of a hypothetical demolition in order to highlight the maximum exposure associated with each building component. This approach is conservative in that the likely scenario would entail rapid, mechanical, destruction of the structure (i.e. without individual phases) and subsequent removal of the debris. The latter tactic would to some extent quench potential airborne contamination as the structure is dropped.

The hypothetical demolition phases representing each of the building components and significant conditions/assumptions are as follows.

Roof Demolition: Contamination is consolidated into built-up roofing material. Although the principal exposure pathway is inhalation of contaminated dust, exposure from this pathway is considered unlikely since the uranium is bound up in the composite roof material (asphaltic tar). The primary dust producing activities would be: destroying the roof with heavy equipment (allowing it fall to the ground); and, loading the roof rubble into trucks using heavy equipment. This scenario is plausible but conservative in that the roof contains asbestos and the likely scenario would involve appropriate asbestos controls - those controls would also mitigate release of airborne radioactivity and worker exposure in general.

The contaminated portion of the roof covers an area of 12,567 ft^2, the contaminated roofing material is 0.75 in. (0.0625 ft) thick. The density of the roofing material is taken to be 1.5 g cm^{-3} (asphaltic tar).

The average total U concentration in the built-up roofing material is estimated at 171 pCi g^{-1}.

Wall Demolition: Contamination is on the surface of the walls. Smear samples have indicated that it is primarily fixed in place. Walls are constructed of 16 inch brick and mortar with large bays of windows. The maximum dust producing activity would involve the use of a wrecking ball (or other forceful demolition technique) where the impact of the ball occurs at approximately 20 feet above ground and dislodges (and makes airborne) some surface contamination on the walls (from brick and glass surfaces) as well as producing dust as clean portions of the brick, mortar, and glass are pulverized. To highlight the maximum exposure, the calculation approach assumes demolition is performed 1 wall at a time, and the rubble from one wall is removed before the next wall is demolished. The instantaneous mass loading during wall demolition is assumed to be 10 mg m^{-3}. This is a very conservative assumption in that the dust loading is so high that the air could not suspend this mass of material for a significant amount of time. In addition, standards for fugitive dust limit the average concentration to 50 μg m^{-3}. In addition to the initial plume, contaminated dust would be resuspended as rubble is removed.

Contamination above standard release criteria has been found on walls from Bay 1 through Bay 8. The average contamination levels on the walls in that region is:

West Wall	21,945 dpm/100 cm^2
North Wall	2,610 dpm/100 cm^2
South Wall	8,029 dpm/100 cm^2

Contamination levels in side rooms between Bay 1 and Bay 8 are within release criteria (average contamination ranges from 274-1641 dpm/100 cm^2) and are not considered to contribute significantly to the dose calculated herein.

Structural Steel Demolition: Contamination is on the upper horizontal surfaces of the overhead metal structural steel (trusses) in bays 1-10 starting at the west wall.

Demolition of Building 23 would likely include salvage of the steel. Because of the likelihood of significant contact with contamination by workers as the steel is recovered, decontamination of trusses will be performed.

Decontamination will be sufficiently rigorous so that no airborne contamination would result from the exposed steel surfaces during demolition. Some areas, such as intersections of steel members, gusset plates, etc., are more difficult to access and could be the source of airborne contamination as the trusses are removed.

Following decontamination it is assumed that some of the contamination from inaccessible areas will be dislodged as the steel building framework is dismantled. The contamination on the inaccessible surfaces is assumed to be the same as the exposed surfaces surveyed during characterization, which was 6214 dpm/100 cm^2. The contamination is from deposited dust which is assumed to be 0.1 cm thick. The surface area of the contaminated steel is approximately 2060 ft^2. However, an engineering review of the structural layout of Building 23 has disclosed that the surface area of the inaccessible portions of the steel is approximately 44.3 ft^2.

The primary exposure pathways are inhalation and ingestion. The activity generating the most airborne contamination will entail toppling the supporting beams and allowing the trusses to fall to the ground where they will subsequently be salvaged for scrap. It is assumed that torch cutting will be performed to size the metal for transport. Airborne contamination will be released as: (1) the trusses fall to the ground; (2) the trusses are cut for transport; and, (3) The trusses are removed. Assuming all trusses are felled in a very short time frame, the contaminated plume from any one truss falling will not have dissipated when the next truss is dropped; thus, the plume of airborne contamination will embody all the contamination present on the trusses.

Personal contamination (and subsequent ingestion) will occur as the trusses are handled during cutting and removal.

Floor Demolition: The floor of building 23 is composed separately of 3 different building materials - concrete, fire brick, or wood block, each overlaid on a concrete base slab. Over 90% of the floor area is composed of wood blocks. Isolated contamination has been found on the surface of the floor, in adhesive between wood blocks and gaps in fire brick and possibly on the

concrete base slab. Planned remedial activities include removal of the top layer (wood, brick, concrete) to expose the sub base for survey and any necessary decontamination. Survey data obtained at locations where contamination was known to be present gave the following (average) count rates.

Concrete	20,912 dpm/100 cm^2
Wood	9416 dpm/100 cm^2

However, a composite sample made from 26 wood blocks systematically collected had an activity concentration of 1 pCi per gram. This concentration is near background levels and the wood blocks will be left at the site after they are removed.

CALCULATION

Dose From Roof Demolition

1. The volume of contaminated built-up roofing material is:

$$(12{,}576\ ft^2)(0.0625\ ft) = 7.86 \times 10^2\ ft^3 = 2.23 \times 10^7\ cm^3$$

2. The mass of the contaminated built-up roofing material is:

$$(2.23 \times 10^7\ cm^3)(1.5\ g\ cm^{-3}) = 3.35 \times 10^7\ g$$

3. The curie content of the contaminated built-up roofing material is:

$$(171\ pCi\ g^{-1})(3.35 \times 10^7\ g) = 5.73 \times 10^9\ pCi$$

4. There is no recognized value for the instantaneous mass loading that would occur as the roof is destroyed and allowed to drop to the floor of the building. An airborne dust concentration of 10 mg m^{-3} depicts severe conditions and would represent a conservative value.

The airborne radioactivity concentration is thus:

$$(0.01\ g\ m^{-3})(171\ pCi\ g^{-1}) = 1.71\ pCi\ m^{-3}$$

If it is conservatively assumed that all airborne particulates have an AMAD of 1 μm then settling of the contaminated plume will be very slow, and a worker who enters the building and begins removal of the contaminated roofing debris soon after the roof is dropped will be exposed to the concentration calculated above.

The magnitude of the exposure will depend on the duration of the work. Because the roof is constructed of 3 in. thick wood plank in addition to the contaminated built-up roofing material, the volume of rubble associated with the contaminated area of the roof is:

$(7.86 \times 10^2\ ft^3$ contaminated built-up material)
$+(12{,}576\ ft^2 \times 0.25\ ft$ wood$) = 3.93 \times 10^3\ ft^3 = 1.11 \times 10^2\ m^3$

If a front loader with a 1.25 m^3 capacity is used to remove the roof rubble, the time required to remove the rubble (during which a worker is exposed) is approximately two 8 hr shifts. 32 hrs elapses, therefore, from the start to finish of the removal of rubble (two 8 hour shifts separated by 16 hrs)

It is assumed that all dust would settle out between shifts, therefore, there would be no exposure to the initial plume after the first shift. If it is conservatively assumed that there would be no airborne reduction during the first 8 hr shift, the exposure to the worker would be:

$(1.71\ pCi\ m^{-3})(20\ m^3$ inhaled per 8 hr shift)

$= 34.2$ pCi inhaled

In addition to the initial plume of contaminated dust produced as the roof is demolished, there would also be contaminated dust generated during removal of the roof rubble. This activity is comparable to normal construction activities and a standard (although very conservative) MLF of 600 $\mu g\ m^{-3}$ is applicable. Again, assuming the work will require two 8 hr shifts to complete, the calculated exposure would be:

$(6 \times 10^{-4}\ g\ m^{-3})(171\ pCi\ g^{-1})(20\ m^3$ shift$)(2$ shifts$)$

$= 4.10$ pCi (inhaled)

The total inhaled activity from the initial plume and follow-up waste removal is:

$34.2\ pCi + 4.10\ pCi = 38.3\ pCi$

The resultant dose is:

U-234 $= (19.1\ pCi)(0.13\ mrem\ pCi^{-1}) = 2.48$ mrem
U-235 $= (0.843\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.10$ mrem
U-238 $= (19.1\ pCi)(0.12\ mrem\ pCi^{-1}) = 2.29$ mrem

Total dose = 4.87 mrem

Wall Demolition

1. The concentration of contaminant in wall components is calculated as follows:

Brick wall:

Thickness 16 inches (40.64 cm)
Effective density 1.62 g cm^{-3}
Mass under 100 cm^2 survey area 6,584 g
Average activity in survey area:

West Wall	21,945 dpm/100 cm^2 (9,885 pCi/100 cm^2)
North Wall	2,610 dpm/100 cm^2 (1,176 pCi/100 cm^2)
South Wall	8,029 dpm/100 cm^2 (3,617 pCi/100 cm^2)

Average concentration:

West Wall	1.5 pCi g^{-1}
North Wall	0.18 pCi g^{-1}
South Wall	0.55 pCi g^{-1}

Glass windows:

Thickness 1/8 inch (0.32 cm)
Density 2.6 g cm^{-3}
Mass under 100 cm survey area 83.2 g
Average activity in survey area same for brick

Average concentration:

West Wall	118.81 pCi g^{-1}
North Wall	14.13 pCi g^{-1}
South Wall	43.47 pCi g^{-1}

The total interior surface area of the West Wall is approximately 1,943 ft^2, of which approximately 923 ft^2 feet is brick and 1020 ft^2 is windows. The masses of these components are:

$$(923\ ft^2 \times 1.33\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(1.62\ g\ cm^{-3})$$
$$= 5.63 \times 10^7 \text{ grams of brick}$$

$$(1020\ ft^2 \times 0.01\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(2.6\ g\ cm^{-3})$$
$$= 7.51 \times 10^5 \text{ grams of glass}$$

The West Wall is thus 98.7% brick and 1.3% glass, and the weighted average concentration is:

$$(0.987[1.5\ pCi\ g^{-1}])+(0.013[118.81\ pCi\ g^{-1}]) = 3.03\ pCi\ g^{-1}$$

The total interior surface of each of the North and South Walls Bays 1 through 8 is approximately 4,224 ft^2, of which approximately 2,304 ft^2 is brick and 1,920 ft^2 is glass. The mass of brick and glass in each of the North and South Walls is:

$$(2304\ ft^2 \times 1.33\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(1.62\ g\ cm^{-3})$$
$$= 1.40 \times 10^8 \text{ grams of brick}$$

$$(1920\ ft^2 \times 0.01\ ft)(2.83 \times 10^4\ cm^3\ ft^{-3})(2.6\ g\ cm^{-3})$$
$$= 1.41 \times 10^6 \text{ grams of glass}$$

The North and South Walls are thus 99.0% brick and 1.0% glass and the weighted average concentrations of U are therefore:

North Wall

$$(0.990[0.18\ pCi\ g^{-1}])+(0.010[14.43\ pCi\ g^{-1}]) = 0.32\ pCi\ g^{-1}$$

South Wall

$$(0.990[0.55\ pCi\ g^{-1}])+(0.010[43.47\ pCi\ g^{-1}]) = 0.98\ pCi\ g^{-1}$$

The instantaneous mass loading factor is 10 mg m^{-3}; therefore the initial airborne concentration is:

West Wall $(3.03\ pCi\ g^{-1})(0.01\ g\ m^{-3}) = 0.030\ pCi\ m^{-3}$
North Wall $(0.32\ pCi\ g^{-1})(0.01\ g\ m^{-3}) = 0.003\ pCi\ m^{-3}$
South Wall $(0.98\ pCi\ g^{-1})(0.01\ g\ m^{-3}) = 0.010\ pCi\ m^{-3}$

2. The dose to the hypothetical maximally exposed individual will depend on the duration of exposure. For a particle size of 1 µm AMAD, the settling velocity is 0.0075 cm s^{-1}, and the contamination plume extends upward 20 ft (609.6 cm), the time required for virtually all of the particulates to settle out is:

$$0.0075\ cm/1\ s = 609.6\ cm/x\ s$$
$$x = 81{,}280\ s = 22.6\ hrs$$

It is assumed that, although he settling time is long, the plume would not be present after the 16 hours between work shifts because of the dispersive effects of winds.

3. The highest exposure would occur if demolition took place at the beginning of the work shift, the airborne contamination is assumed to be uniformly distributed throughout the plume, and rubble is removed before the plume dissipates. The duration of exposure will depend on the volume of rubble to be removed and the rate it is removed. A work crew with a front loader with a 1.25 m^3 capacity can load 80 m^3 per hour (Waier 1994). Because of the irregular sizing of the building

rubble, it is plausible that the work may go slower and a conservative value of 40 m^3 per hr is used. The approximate quantities of rubble are:

West Wall

(923 ft^2 x 1.33 ft thick brick) = 1228 ft^3 (34.8 m^3)
(1020 ft^2 x 0.01 ft thick windows) = 10.2 ft^3 (0.3 m^3)

South and North Walls (each)

(2304 ft^2 x 1.33 ft thick brick) = 3064 ft^3 (86.8 m^3)
(1920 ft^2 x 0.01 ft thick glass) = 19.2 ft^3 (0.5 m^3)

The duration of the cleanup of the rubble from the West Wall is:

$(35.1\ m^3) \div (40\ m^3\ hr^{-1}) \approx 1$ hr

The duration of cleanup of the rubble from each of the North and South Walls are:

$(87.3\ m^3) \div (40\ m^3\ hr^{-1}) \approx 2.5$ hrs each

4. The corresponding radioactivity intake is:

West Wall $(0.030\ pCi\ m^{-3})(2.5\ m^3\ hr^{-1})(1\ hr) = 0.075$ pCi
North Wall $(0.003\ pCi\ m^{-3})(2.5\ m^3\ hr^{-1})(2.5\ hr) = 0.019$ pCi
South Wall $(0.010\ pCi\ m^{-3})(2.5\ m^3\ hr^{-1})(2.5\ hr) = 0.063$ pCi

The total radioactivity intake is:

0.075 pCi + 0.019 pCi + 0.063 pCi = 0.157 pCi

And the dose from is:

U-234 = $(0.0767\ pCi)(0.13\ mrem\ pCi^{-1}) = 0.0100$ mrem
U-235 = $(0.0035\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.0004$ mrem
U-238 = $(0.0767\ pCi)(0.12\ mrem\ pCi^{-1}) = 0.0092$ mrem

Dose from Wall Demolition = 0.02 mrem

Structural Steel Demolition

1. The contaminated area on the steel trusses is estimated to be 2060 ft^2, and of this, 44.3 ft^2 is inaccessible.

2. For the purpose of this calculation, the accessible areas are assumed to be successfully decontaminated, and contaminated dust removed from the inaccessible areas by vacuuming or flushing with water. Therefore the only remaining contamination is the fixed contamination in the inaccessible areas, which is assumed to be similar to that measured in the accessible areas during the characterization.

   The average fixed contamination was measured to be:

   $$6214 \text{ dpm}/ 100 \text{ cm}^2 = 2799 \text{ pCi}/100 \text{ cm}^2$$

   The contamination on the steel is from settled dust. Assuming the dust is 0.1 cm thick and has a density similar to soil, i.e. ~1.6 g cm^{-3}. Therefore, the concentration of the contamination is:

   $$(27.99 \text{ pCi/cm}^2)/(0.1 \text{ cm})/(1.6 \text{ g cm}^{-3}) = 174.9 \text{ pCi g}^{-1}$$

3. The intake of radioactivity will depend on the duration of exposure and the concentration of contaminant. An engineering appraisal of the building plans for Building 23 have disclosed that there is approximately 74.1 tons of over head supporting steel in bays 1-11. For this quantity of structural metal, salvage operations will take approximately five 8 hour shifts.

   A conservative mass loading factor of 600 μg m^{-3} (DOE 1983) for construction activities would result in the following intakes of uranium.

   $$\text{U-234 } (85.5 \text{ pCi g}^{-1})(6 \times 10^{-4} \text{ g m}^{-3})(5 \times 20 \text{ m}^3) = 5.13 \text{ pCi}$$
   $$\text{U-235 } (3.9 \text{ pCi g}^{-1})(6 \times 10^{-4} \text{ g m}^{-3})(5 \times 20 \text{ m}^3) = 0.23 \text{ pCi}$$
   $$\text{U-238 } (85.5 \text{ pCi g}^{-1})(6 \times 10^{-4} \text{ g m}^{-3})(5 \times 20 \text{ m}^3) = 5.13 \text{ pCi}$$

4. The resultant dose is:

   $$\text{U-234} = (5.13 \text{ pCi})(0.13 \text{ mrem pCi}^{-1}) = 0.667 \text{ mrem}$$
   $$\text{U-235} = (0.23 \text{ pCi})(0.12 \text{ mrem pCi}^{-1}) = 0.028 \text{ mrem}$$
   $$\text{U-238} = (5.13 \text{ pCi})(0.12 \text{ mrem pCi}^{-1}) = 0.616 \text{ mrem}$$

   Total dose = 1.3 mrem

5. The intake of uranium by ingestion can be estimated by assuming a worker will ingest 480 mg of dust and dirt per day (8 hour shift).

   The work takes approximately five 8 hour work shifts (Waier 1944); therefore, the total intake of dust and dirt is: 5 shifts x 480 mg per shift = 2.4 g

The resultant radioactivity intake is thus:

U-234 = (85.5 pCi g^{-1})(2.4 g) = 205.2 pCi
U-235 = (3.9 pCi g^{-1})(2.4 g) = 9.36 pCi
U-238 = (85.5 pCi g^{-1})(2.4 g) = 205.2 pCi

6. The resultant dose is:

U-234 = (205.2 pCi)(2.6×10^{-4} mrem pCi^{-1}) = 0.053 mrem
U-235 = (9.36 pCi)(2.5×10^{-4} mrem pCi^{-1}) = 0.002 mrem
U-238 = (205.2 pCi)(2.5×10^{-4} mrem pCi^{-1}) = 0.051 mrem

Total dose from ingestion = 0.11 mrem

7. The total dose to a demolition worker from overhead steel contamination is thus 1.41 mrem

Removal of Floor Blocks

1. The floor blocks will have been stockpiled in the building after their removal from the floor. The demolition workers are assumed to remove them similarly to the other building rubble at a rate of approximately 40 m^3 per hour. The volume of wood blocks has been estimated to be 150 m^3, which could be removed in:

$(150\ m^3)/(40\ m^3\ h^{-1}) \approx 4$ hours

2. Using the conservative mass loading factor of 600 mg m^{-3} and the contamination concentration for the blocks as 1 pCi/g, the airborne concentration of uranium would be:

(1 pCi/g)(6×10^{-4} g m^{-3}) = 6×10^{-4} pCi m^3

The corresponding radioactivity intake is:

(6×10^{-4} pCi m^{-3})(2.5 $m^3 hr^{-1}$)(4 hr) = 0.006 pCi

3. And the dose is:

U-234 = (0.0029 pCi)(0.13 mrem pCi^{-1}) = 0.0004 mrem
U-235 = (0.0001 pCi)(0.12 mrem pCi^{-1}) = 0.00001 mrem
U-238 = (0.0029 pCi)(0.12 mrem pCi^{-1}) = 0.0003 mrem

The total dose is:

0.0004 mrem + 0.00001 mrem + 0.0003 mrem = 0.0007 mrem

Contamination in Building Rubble

1. To estimate the total radioactive contamination expected in the building rubble that would be disposed of following demolition, the radioactive concentrations used to estimate the dose to the demolition worker is assumed to be in the rubble.

Component	Contamination	Amount	Total Activity
Roof Material	171 pCi/g	3.35×10^7 g	5.73×10^9 pCi
West Wall	3.03 pCi/g	5.71×10^7 g	1.73×10^8 pCi
North Wall	0.32 pCi/g	1.41×10^8 g	4.51×10^7 pCi
South Wall	0.98 pCi/g	1.41×10^8 g	1.38×10^8 pCi
Steel	28 pCi/cm^2	4.12×10^4 cm^2	1.15×10^6 pCi
Floor Blocks	1 pCi/g	8.81×10^7 g	8.81×10^7 pCi

Total radioactivity = 6.18×10^9 pCi = 6.18 mCi

Assuming that rubble from the contaminated areas is 1/3 of the entire building, the average concentration for the building rubble is calculated as:

$$6.18 \times 10^9 \text{ pCi} / (3 \times 4.64 \times 10^8 \text{ g}) = 4.44 \text{ pCi/g}$$

This concentration is well below the uranium soil contamination guideline range 35 to 100 pCi per gram typically used with sites similar to Chapman Valve. In addition, it is comparable to the natural radioactivity found in many common building materials. Attachment 1 contains a table which lists the natural radioactivity concentrations found in common building materials.

The values of the individual uranium isotopes would therefore be:

U-234 = 3.0 mCi
U-235 = 0.14 mCi
U-238 = 3.0 mCi

These values are well below the EPA's Reportable Quantities (RQs) for U-234, U-235, and U-238 specified in 40 CFR 302.

Using the specific activity of the uranium isotopes, the mass of the uranium present is

U-234 = (3.02 mCi)/(6.19 mCi/g) = 0.5 g
U-235 = (0.14 mCi)/(2.14×10^{-3} mCi/g) = 65.4 g
U-238 = (3.02 mCi)/(3.33×10^{-4} mCi/g) = 9069 g

The total mass of uranium is:

U = 9135 g = 20.1 lbs

This value is well below EPA's RQ values for uranium compounds (uranyl acetate, uranyl nitrate) with respect to chemical hazards.

CONCLUSIONS

Contamination on the roof and walls of Building 23 will not result in a radiation dose above 100 mrem to any member of the general public. The maximum dose to a hypothetical demolition worker is estimated to be 6.3 mrem, and offsite exposure to the general public would be much less. Brick comprises approximately 64% of the mass of the contaminated portion of Building 23. The natural (background) radioactivity found in brick at the site is approximately 6.4 pCi per gram. The amount of radioactivity added by the surface contamination ranged from 0.18 to 1.5 pCi per gram; thus, the contamination has only increased the radioactivity naturally found in brick less than 23% Given the low radiation dose calculated to result from roof and wall contamination (and the fact that these calculations were very conservative and likely over-estimated the dose) supplemental limits for these building components should be the same as the contamination levels there at present and no remedial action need be performed.

If no decontamination is performed on overhead steel, it is expected there would be significant worker contact with contamination during salvage operations. For this reason, a general decontamination of overhead steel should be performed to remove all loose contamination. Removal of loose contamination would preclude the cleaned areas as a source of airborne contamination.

Attachment 1

Naturally Occurring Radiation in Building Materials and Solid Waste

Material		U-238 (pCi/g)	Ra-226 (pCi/g)	Th-232 (pCi/g)	Reference
Red Brick		1.2		1.8	A
Concrete		0.9			A
Clay Brick		3.0	14	1.2	A
Silicate Brick		0.2	0.2	0.1	A
Granite		6.0	2.4	2.2	A
Natural Gypsum		0.4	0.6	0.2	A
Manufactured Gypsum	C		30 - 40		B
Concrete Block (Fly Ash)	C	1 - 12	0.2 - 4.0	1.0 - 1.2	A
Cement		1.1		0.4	A
Silica Sand		0.3		0.5	A
Photo Materials	C	0.07 - .34			A
Coal	C	0.07 - 8.4		0.21 - 19	A
Fly Ash - Anthracite		3.0			B
Fly Ash - Eastern Bituminous		12.7			B
Fly Ash - Western Bituminous		5.4			B
Fly Ash - Lignite		7.4			B
Fertilizer		D	5 - 6		A

A - Eisenbud, Merril (1987), "Environmental Radioactivity From Natural, Industrial, and Military Sources," Academic Press, Inc., New York.
B - National Council on Radiation Protection and Measurements, "Exposure From the Uranium Series with Emphasis on Radon and its Daughters," NCRP Report 77, 1983.
C - Values shown are ranges
D - Reference does not give activities but does state that in 1974 about 26 million tons of ore (phosphate rock) mined for fertilizer production contained about 1,000 Ci of radium-226 in equilibrium with uranium-238 and 3,000 Ci of potassium-40 are added to the soil during fertilizer application in the US annually.

EXECUTIVE SUMMARY
for the
Technical Study for Remedial Action
at the Chapman Valve Site

The former Chapman Valve property, currently owned by the Crane Company, was designated for inclusion into DOE's FUSRAP Program in 1992. A copy of the designation report along with DOE information about FUSRAP were provided to you at the May 3rd meeting. The characterization study of the site conducted in December 1994 confirmed that radiological contamination above DOE guidelines is confined to portions of the roof's western end, localized spots on the south and west exterior walls, and interior areas in the western end of Building 23. Based upon discussions with the property owner, the future use scenario for Building 23 is demolition. There are no plans to occupy or utilize the structure in the interim.

As outlined in the Technical Study, DOE residual contamination guidelines for the release of contaminated structures are contained in DOE Order 5400.5 (proposed rule 10 CFR 834), "Radiation Protection of the Public and the Environment." Although the Order is used by DOE to establish requirements for the control and cleanup of radiological contamination at FUSRAP sites, it is important to note that the basis for the Order was adopted from other regulatory sources such as the Environmental Protection Agency, Nuclear Regulatory Commission, International Commission on Radiological Protection, and National Council on Radiation Protection.

The guidelines for surface contamination are specified in disintegrations per minute (dpm) per unit area (100 square centimeters). The guidelines applicable to this site are 5000 dpm/100 cm^2 average, 15000 dpm/100cm^2 maximum, and 1000 dpm/100 cm^2 removable for alpha and beta/gamma emitting radionuclides. Soil (volumetric) guidelines for uranium are site-specific, based upon site characteristics and use. A site-specific uranium soil guideline has not been developed for Chapman Valve, but values typically range between 50 and 100 pCi/g for total uranium. A site-specific uranium guideline was not developed because no soil contamination was identified.

Also outlined in DOE Order 5400.5 are radiation dose limits for members of the general public. The maximum allowable annual dose to a member of the general public is 100 millirem. As discussed in our meeting, DOE Order 5400.5 also allows for the application of supplemental limits if specific circumstances indicate that the cleanup guidelines are not appropriate for a given property or portion of that property. If the cost of a remedial action is unreasonably high relative to the benefit that is achieved with removal of the radioactive material, supplemental limits may be warranted. These limits must be shown to be protective under very conservative scenarios and the potential dose from the remaining residual contamination must be below the allowable dose limit of 100 millirem per year.

Average radiological contamination levels in Building 23 on structural surfaces ranged 1000-29000 dpm/100 cm^2 beta/gamma with isolated spots of elevated activity. Results of analysis of dust samples taken from interior trusses ranged from approximately 7 to 8000 pCi/g U-238 with average levels of 870 pCi/g. The highest concentrations were observed on truss #2 and decreased significantly from west to east. Systematic analysis of roof composite materials collected from the western portion of the roof yielded average results of 83 pCi/g for U-238.

Work plans for remediation at Chapman Valve include the decontamination of the west bridge crane, the mono hoist area and adjacent rooms, interior trusses 1-7, and the concrete base slab in bays 1-8. In addition, debris located in the building's western end will be surveyed for radiological contamination and either decontaminated for release or packaged for waste disposal. Areas in which we plan to take no action and utilize supplemental limits are the roof, interior trusses East of truss #7, interior walls, and the upper layer of flooring (wooden blocks).

As demonstrated in the Technical Study, following DOE's remediation efforts at the site, the calculated dose to the hypothetical demolition worker from any residual uranium will be less than 10 millirem. This dose was derived assuming that the worker participates in the demolition of all structural components. In addition, the assumptions used in the dose calculation regarding the exposure and the nature and extent of contamination are very conservative. This low dose and the remedial action approach will allow for the release and certification of the property for future use without radiological restrictions.

Technical Study for the Remedial Action at the Chapman Valve Site in Indian Orchard, Massachusetts

25 MAY 1995

Introduction

This technical study contains information obtained during characterization activities at the Chapman Valve Site (CHV) in Indian Orchard, a suburb of Springfield, MA, in the southern region of the state. The former owner of the site, the Chapman Valve Manufacturing Company, produced special valves and manifolds during the 1940s for the Manhattan Engineer District and the Atomic Energy Commission (AEC). During 1948 the company machined uranium metal for the Brookhaven National Laboratory, which operated under contract to the AEC. The site was transferred to the Crane Company in 1981, but it has been vacant since 1987, when the Crane Company discontinued manufacturing operations at the site.

After uranium machining operations at the site were completed in 1948, the machinery and equipment used in the operations, the floor of the machining area, and the remaining uranium cuttings, scrap metal, and sweepings were removed under the direction of the AEC. The walls of the area were washed with detergents to remove uranium residues and the building was effectively decontaminated to levels that were acceptable at that time.

A site survey conducted by the Oak Ridge National Laboratory for the Department of Energy (DOE) in 1991 (ref. 1) revealed that radioactive contamination above current cleanup guidelines, consisting primarily of uranium residues, remained inside the building on portions of the floor, walls, and roof trusses. As a result of these findings, in December 1992, the DOE designated the CHV for inclusion in its Formerly Utilized Sites Remedial Action Program.

A characterization survey was performed in December 1994. The characterization was performed to determine areas of contamination in building 23 and to obtain information necessary for remedial action planning. Sampling and survey locations for this effort were determined based on the findings of the 1991 survey (ref. 1) and information gained from site visits prior to characterization.

Standard analytical methods were used for analysis of the samples (ie. gamma spectroscopy for solid samples and alpha spectroscopy for liquid samples).

Building Description

Building 23 at CHV was constructed in 1921 and is a one story masonry building with steel frame. The roof of the building is wood plank with a tar composite set on steel trusses. The roof of the building has begun to fail and visible holes of varying size have developed. All windows have been covered with insulation and aluminum siding on the exterior surfaces. The floor of the building is covered primarily with wooden blocks sealed in tar over a concrete sub slab. In areas were it has been necessary to replace the wooden blocks either firebrick or asphalt blocks are present. Several areas of the floor also contain a three inch lift of concrete on the sub slab. All Crane property surrounding the building is either paved or concrete except for a small area just outside the west end of the building.

Methodology

The work direction for the characterization activities was contained in WI-95-026, Characterization of Chapman Valve.

To aid in identification of areas within the building the east-west support columns were numbered 1 to 30 and the north-south columns were labeled AA, A, B, and C. The northwest corner of each section was designated as the name of that section or grid block and rooms were designated with the number of the northwestern most column number (see Figure 1).

The overhead trusses near the roof were scanned to determine if they were above guidelines for alpha and/or beta-gamma contamination. The trusses in the building are complex back to back welded angles. A series of short trusses oriented north-south were spot surveyed approximately every 10 feet with no fewer than eight surveys per truss. A series of three trusses also run east-west the full length of the building and rest on top of the north-south trusses. Due to the difficulty of reaching the east-west trusses

they were assumed to contain contamination equivalent to the north-south trusses. The survey was performed in a way as to determine if the truss was contaminated and not to determine exact portions of contamination on a truss. In six elevated areas on separate trusses a dust sample was collected and analyzed for radiological parameters. A composite sample was also collected from an area determined to be radiologically contaminated and analyzed for TCLP metals including zinc and copper.

Where it was possible to reach ceiling areas through the trusses the ceiling was scanned for contamination. If roof vents were visible from the ceiling then the vent was scanned to determine if it was above criteria.

A Field Instrument for Detection of Low Energy Radiation (FIDLER) walkover was performed on the floors in the western half of the building. The walkover was abandoned because the shielding effect of the wooden blocks on the floor prevented detection of contamination between the wooden blocks and concrete. The areas around boreholes placed in the building were scanned along with the tops, bottoms and sides of concrete cores.

The wooden blocks on the floor in the northern half of grid A1 were removed and checked for contamination. Two composite samples were taken from the blocks. One sample was analyzed for radiological parameters and the other sample was analyzed for TCLP total (metals,, volatile organics, semi-volatile organics, pesticides, herbicides), flashpoint, reactivity, corrosivity, and PCBs.

The walls were scanned to determine if they were above guidelines for alpha and/or beta-gamma contamination. The measurements were single readings taken on grid intersections. Starting in the west end of the building and moving east the grid was 10 X 10 foot for the first 200 feet, 15 X 15 foot for the next 100 feet, and 20 X 20 foot for the remainder of the building.

Window sills and radiators were also scanned paying particular attention to horizontal surfaces. In elevated areas two composite dust samples were collected and analyzed for radiological parameters.

Radiological samples were collected from both biased and systematic boreholes installed outside and within the building; six (6) borehole samples were collected outside and 32 borehole samples were collected inside Building 23 (Figure 1). The location of biased boreholes was based on previous investigations from the 1991 site survey. Boreholes extended to a minimum depth of three feet, or refusal, below the bottom of the concrete. Radiological samples were taken in six inch increments. In recording the depth of the sample, the measurement began at the floor surface not the elevation of the soil. For example, if the concrete was six inches thick the first sample was recorded as six inches to one foot. In addition, the thickness of the concrete for each borehole and the presence of any reinforcement material in the concrete was recorded.

The six inch sample increment from each borehole with the highest beta/gamma reading as determined with handheld instrumentation was shipped and analyzed for radiological parameters and all others were archived. A chemical sample was analyzed for TCLP total (metals,, volatile organics, semi-volatile organics, pesticides, herbicides), flashpoint, reactivity, corrosivity, PCBs, and paint filter test.

The different roof levels of building 23 were identified as roof R1 through R5 from North to South. Direct measurements were taken and recorded on each accessible level of the roof to determine if contamination was present. In areas where the roof was not accessible from a manlift outside the building, cores were cut from inside and subsequently scanned and sampled. The samples were analyzed for radiological parameters.

Drains for the roof of the building empty into the storm sewer. The drains were surveyed utilizing a sewer probe and samples were collected from two laterals that collect runoff from the building and at a collection manhole at the junction of the two laterals. The samples were analyzed for radiological parameters.

Steam trenches bisecting the building (two with a north-south and one with an east-west orientation) were sampled at 15 foot intervals and composited. Three composites, one from each trench were analyzed for radiological parameters.

Water and sludge samples were collected from various pits and sumps located throughout the building. The samples were analyzed for radiological parameters.

An exterior FIDLER walkover was performed around the building exterior. It covered all areas within a 50 foot radius of the building exterior. In areas where 50 foot extended beyond the property boundary or past a nearby building the property boundary or the nearby building was the extent of the exterior scan. Survey readings were also taken under aluminum siding placed over the windows.

Quality control was performed in the field on sampling and survey efforts and was based on standard sampling protocols. Informational chemical sampling performed in non-radiological areas did not include quality control samples.

Residual Radioactive Contamination Guidelines

DOE residual contamination guidelines governing the release of property for future use are contained in DOE Order 5400.5. The criteria used in analyzing the data of this characterization effort for surface contamination was 5,000 dpm/100 cm^2 average direct, 15,000 dpm/100 cm^2 maximum direct, and 1,000 dpm/100 cm^2 removable for alpha and beta-gamma. Soil guidelines for radium and thorium are 5 pCi/g averaged over the first 15 cm (6 in) of soil below the surface and 15 pCi/g when averaged over any 15 cm (6 in) thick layer below the surface, excluding background concentrations. The soil guideline for uranium is a site specific guideline based on site characteristics and use. A site specific uranium guideline has not yet been developed for CHV but typically ranges between 50 and 100 pCi/g for total uranium.

Radiological Results

Data results for specific locations are discussed in the following sections and tables containing data results are located in the appendices to this technical study. Radiological results are located in Appendix A, detected chemical results (undetected results are not shown) are located in Appendix B, interior survey results are located in Appendix C, exterior survey results are located in Appendix D, and decontamination test results are located in Appendix E. A table is provided on pages 9 and 10 showing page numbers of data tables. Numbers of data packages containing the results of the characterization are listed in an interoffice memorandum (ref. 2).

Background

Background radiological samples were collected at three locations, two off site and one onsite. Soil sample results averaged 5.1, 0.98, and 3.7 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively. The average exposure rate from the three background locations was 9.3 μR/hr. Background samples were also collected from building material at the site and can be seen in Appendix A.

Building Interior

Floors

The concrete cores and areas adjacent to each borehole were scanned to determine if surface contamination was present on the concrete floor or wooden blocks. The areas around boreholes 3, 4, 11, and 14 (Figure 1) exhibited contamination near or above criteria. Spotty contamination was found on the floor in the west end of the building, however it was difficult to determine contamination that existed under the wooden blocks because of the shielding effects of the block.

All wooden blocks and concrete removed from the Northern half of grid block A-1 were scanned to determine if any were elevated above surface criteria. Approximately 3 percent of the pieces were elevated near or above criteria. The average level of beta/gamma surface contamination was 9,416 and 20,912 dpm/100 cm^2 for the elevated wooden blocks and concrete pieces respectively. A wood block, wood block composite, and tar sample was also taken for radiological analysis. The wood block samples were ground up and analyzed to obtain a volumetric activity and resulted in a ^{238}U activity of 50.5 pCi/g for the biased wood block and 1.0 pCi/g for the composite wood block. The ^{238}U results for the biased tar sample analysis was 7,891 pCi/g.

Boreholes

Boreholes were installed inside the building to determine if subsurface contamination exists at CHV below the floor of building 23. Thirty two boreholes were installed inside the building. Compared to the historical range of site specific uranium guidelines all soil samples were below soil guidelines and averaged 2.40, 0.67, and 0.44 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively.

Walls

Results of the wall survey indicate contamination above direct criteria in the main center portion of the building on a majority of the west wall, the north wall down to column 8, and the South wall down to column 8. The levels of contamination tend to increase with increasing height on the wall and are primarily found on horizontal surfaces and behind objects on the wall (i.e., conduit, radiators, pipes, etc.). Dust samples collected on the North and South wall contained uranium-238 activities of 14.8 and 97.2 pCi/g respectively.

Five wall fixtures (radiators, electrical boxes, etc.) were removed from the wall and the wall was surveyed to obtain data from behind the fixtures. The maximum and average beta/gamma result was 4,944 and 2,272 dpm/100 cm^2.

Contamination above surface criteria was found on the walls in rooms B1, B2, B4, B7, and B8 on the South side of the building. The contamination was not as widespread as the center portion of the building and levels of contamination were lower. The walls in the rooms on the North side of the building were not contaminated.

Overheads

The trusses in the center portion of the building contain contamination above direct criteria down to approximately truss 12/13 and elevated readings below criteria down to truss 17. Dust samples were collected on trusses 2, 3, 5, 7, and 19. ^{238}U analytical results for the dust samples were 8,143; 103; 14.1; 55.8; and 6.9 pCi/g on trusses 2, 3, 5, 7, and 19 respectively.

Contamination above surface criteria was found in rooms B1, B2, B4, B7, and B8 on the south side of the building in the overheads. The contamination was not as widespread as in the center portion of the building and levels of contamination were lower. A single survey result exceeded surface criteria in the rooms on the north side of the building in room AA-1.

Bridge Crane

The top horizontal surface of the bridge crane located in grid AA-25 was contaminated above criteria to levels up to 28,973 dpm/100 cm^2. The contamination was highest near the center of the crane. A second crane located farther east in the building was not contaminated above direct or removable criteria.

Sumps, Pits, Trenches, and Tunnel

All sediments and water samples collected from sumps, pits, trenches, and the tunnel were below applicable guidelines. Maximum water sample results were 6.4, 0.72, and 0.15 pCi/l and maximum sediment sample results were 10.9, 1.2, and 1.7 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively.

Building Exterior

Grounds

Results from an exterior walkover out to 50 foot from the building were all near background and did not indicate the presence of any contamination.

Boreholes

Boreholes were installed outside of the building to determine if subsurface contamination exists at CHV outside and in the vicinity of building 23. Six biased boreholes were installed and the locations were determined based on historical information and photographs. All exterior soil samples were near background, below guidelines, and averaged 1.9, 0.67, and 0.49 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively.

Surface

The exterior north and south wall survey results indicated elevated radiological conditions; however, except for a single reading on a window sill on the south side (10,743 dpm/100 cm^2) the readings were below direct and removable criteria.

Survey results from the east wall of the building did not indicate the presence of any contamination.

The exterior west wall survey results indicated elevated radiological conditions; however, except for a single reading on a window sill (8,117 dpm/100 cm^2) the readings were below direct and removable criteria. The elevated readings were primarily located near the lower southern most window that contained an exhaust to a uranium chip burner during the uranium processing in the 1940s.

Survey measurements were also obtained from behind aluminum siding that had been placed over the exterior side of the windows. All readings were below direct and removable surface criteria except for a reading taken near the location of the chip burner exhaust. The beta\gamma result at this location was 6,418 dpm/100 cm^2.

Storm Sewers

All building and roof drains exit the building on the north side and are collected in two lateral storm sewers parallel to the north wall of the building. The laterals drain to a collection basin located outside the building at approximately Grid a-24. The sewer then runs north through the rest of the Crane property where a grease and sediment trap is located. Samples were collected in each lateral, at the collection basin, and at the grease/sediment trap. Maximum sediment and water sampling results were 3.3, 0.85, and 0.44 pCi/g and 2.7, 0.21, and 0.47 pCi/l for ^{238}U, ^{232}Th, and ^{226}Ra respectively. All samples collected from the storm sewer system at CHV were below guidelines.

The drain pipes from the roof were accessed in three locations and a sewer probe was snaked down the pipe. A total of 36 feet of pipe was scanned at one foot intervals. Measurements were elevated above background (18 CPM) and averaged 21.8 CPM with a maximum of 48.1 CPM. Due to the inability to source check the sewer probe it is not possible to determine definite contamination levels. The numbers resulting from the survey of the drain lines indicate only that the drain lines contain contamination above background. Further investigation will be necessary to determine if contamination above guidelines exist in the pipes.

Roof

Survey results of the 11 cores from roofs R2, R3, and R4; survey results from roof R1 and R5; and FIDLER measurements indicate contamination present to levels as high as 146,722 dpm/100 cm^2. Roof R1 is contaminated above surface criteria down to approximately column 5, Roof R2 is contaminated above surface criteria down to approximately column 6, Roof R3 is contaminated above surface criteria down to approximately column 9, and Roofs R4 and R5 are contaminated above surface criteria down to approximately column 12.

Three biased samples of roof material were collected in areas where survey results indicated the presence of contamination. The average sample results for ^{238}U, ^{232}Th, and ^{226}Ra were 116, 0.97, and 0.64 pCi/g, respectively. Eight systematic samples from the roof yielded a maximum and average ^{238}U result of 213.8 and 83.2 pCi/g.

Core sample and survey results indicate that contamination is present in the tar composite and not in the roof planks in areas were the roof is in good condition. However, in areas were the roof is failed and water has been allowed to pass through the wood the rotten wood is contaminated as high as 942.8 pCi/g for ^{238}U.

Decontamination Tests

Truss Test

A decontamination test was performed on three trusses by hand brushing known areas of contamination with a wire brush. After light brushing additional measurements indicated that the areas tested were cleaned to levels below direct and removable surface criteria, for example on truss 2 the initial reading of 32,277 dpm/100 cm^2 was cleaned to 1,397 dpm/100 cm^2.

Paint Test

At twelve locations in the western end of the building, measurements were taken before and after paint removal on the interior walls. At all locations but two, the level of contamination fell significantly after paint removal except at locations 2 and 7 where the second measurement was slightly higher or equal to the first measurement. At locations where initial readings were high (for example, location 11 where initial readings were 50,015 dpm/100 cm^2) the second measurement after paint removal remained above direct criteria (in the case of location 11, 26,102 dpm/100 cm^2). For the 10 locations where a decrease occurred after paint removal, the decrease in beta/gamma measurements averaged 61%. A similar strategy was utilized during a data gap sampling effort at 10 locations. Of the 10 locations all decreased except location 7 where there was a slight increase after paint removal. Similarly to the initial paint test the readings after paint removal decreased by an average of 66%.

Chemical Results

A dust sample obtained from contaminated trusses in the west end of the building was analyzed for TCLP-metals, copper, and zinc. Analytical results were below RCRA guidelines for metals.

A soil sample from borehole 1 was analyzed for TCLP-total, reactivity, flashpoint, corrosivity, PCBs and paint filter. Results indicate that the soil in the sample does not contain RCRA constituents or display any RCRA characteristics.

Roof material was composited from the contaminated roof cores and analyzed for TCLP-total, reactivity, flashpoint, corrosivity, PCBs, and TOX. Results indicate that the roof material in the sample does not contain RCRA constituents or display any RCRA characteristics.

Paint chips collected during the characterization were analyzed for TCLP lead and cadmium. Analytical results were below applicable RCRA guidelines for the paint on the walls but paint from the overhead crane, electrical boxes, and structural steel failed. Based on these sampling results the wall paint is not a RCRA waste but the crane, electrical box, and structural steel paint may be a RCRA waste.

Bulk asbestos sampling indicated asbestos in the composite roof of 3% and in insulation in an electrical box of 40 to 50%.

Summary

Characterization findings indicate the presence of contamination in the western end of building 23 at CHV (see Figures 1, 2, and 3). These findings are in agreement with historical process information obtained during characterization activities from a former Chapman Valve supervisor. According to the supervisor a temporary wall was constructed across the center portion of the building from floor to ceiling between columns A-7 and B-7, this is also the location in the building where contamination on the walls begins to decrease significantly. There was also a chip burner located in the southwest corner of Grid A-1 that exhausted to the atmosphere out a nearby window. The exhaust location and the shape of the roof of the building would tend to deposit more contamination on the south roof than the north roof as the survey results indicate. The uranium storage area was located in room B-4 and this room contained the concrete cores that were elevated.

Levels of contamination decreased from west to east in the building and survey and sampling results from the east end indicated near background radiological conditions. The exterior of the building except for the roof and several locations on the exterior west and south walls indicated near background radiological conditions.

Chemical sampling results indicated the presence of lead paint on the cranes, electrical boxes, and structural steel. If removal of the paint is required, appropriate RCRA regulation will be addressed. Bulk asbestos sampling was positive for the composite roof material and electrical box insulation.

Results from decontamination tests indicated that the majority of the contamination on the walls was removed with the paint. Contamination on the trusses was significantly decreased after light brushing.

References

1. Oak Ridge National Laboratory, "Results of the Radiological Survey at the Former Chapman Valve Manufacturing company, Indian Orchard, Massachusetts", ORNL/RASA-92/1, July 1992.

2. Memo, "Chapman Valve Characterization Data Package Numbers", From Robbins to File, May 4, 1995.

CONCLUSIONS

Contamination on the roof and walls of Building 23 will not result in a radiation dose above 100 mrem to any member of the general public. The maximum dose to a hypothetical demolition worker is estimated to be 6.3 mrem, and offsite exposure to the general public would be much less. Brick comprises approximately 64% of the mass of the contaminated portion of Building 23. The natural (background) radioactivity found in brick at the site is approximately 6.4 pCi per gram. The amount of radioactivity added by the surface contamination ranged from 0.18 to 1.5 pCi per gram; thus, the contamination has only increased the radioactivity naturally found in brick less than 23% Given the low radiation dose calculated to result from roof and wall contamination (and the fact that these calculations were very conservative and likely over-estimated the dose) supplemental limits for these building components should be the same as the contamination levels there at present and no remedial action need be performed.

If no decontamination is performed on overhead steel, it is expected there would be significant worker contact with contamination during salvage operations. For this reason, a general decontamination of overhead steel should be performed to remove all loose contamination. Removal of loose contamination would preclude the cleaned areas as a source of airborne contamination.

Editor's Note:

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FUSRAP Project
Job 14501

NO. 133-95-003

~~NO.~~ REV. NO. 0
DATE: 05-08-95

FUSRAP TECHNICAL MEMORANDUM

TO: Mike Redmon

FROM: Robert Robbins

DATE: 4 May 1995

SUBJECT: Chapman Valve Characterization Results

Prepared by	Team Lead Approval	Project Engineer Approval	Project Manager Approval
5/4/95	5/5/95		5/5/95

Introduction

This technical memorandum contains information obtained during characterization activities at the Chapman Valve Site (CHV) in Indian Orchard, a suburb of Springfield, MA, in the southern region of the state. The former owner of the site, the Chapman Valve Manufacturing Company, produced special valves and manifolds during the 1940s for the Manhattan Engineer District and the Atomic Energy Commission (AEC). During 1948 the company machined uranium metal for the Brookhaven National Laboratory, which operated under contract to the AEC. Records indicate that the Chapman Valve Manufacturing Company also conducted rolling operations on uranium metal during 1949 under contract to the AEC. The site was transferred to the Crane Company in 1981, but it has been vacant since 1987, when the Crane Company discontinued manufacturing operations at the site.

After uranium machining operations at the site were completed in 1948, the machinery and equipment used in the operations, the floor of the machining area, and the remaining uranium cuttings, scrap metal, and sweepings were removed under the direction of the AEC. The walls of the area were washed with detergents to remove uranium residues and the building was effectively decontaminated to levels that were acceptable at that time.

A site survey conducted by the Oak Ridge National Laboratory for the Department of Energy (DOE) in 1991 (ref. 1) revealed that radioactive contamination above current cleanup guidelines, consisting primarily of uranium residues, remained inside the building on portions of the floor, walls, and roof trusses. As a result of these findings, in December 1992, the DOE designated the CHV for inclusion in its Formerly Utilized Sites Remedial Action Program.

A characterization survey was performed in December 1994. The characterization was performed to determine areas of contamination in building 23 and to obtain information necessary for remedial action planning. Sampling and survey locations for this effort were determined based on the findings of the 1991 survey (ref. 1) and information gained from site visits prior to characterization.

Standard analytical methods were used for analysis of the samples (ie. gamma spectroscopy for solid samples and alpha spectroscopy for liquid samples).

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Building Description

Building 23 at CHV was constructed in 1921 and is a one story masonry building with steel frame. The roof of the building is wood plank with a tar composite set on steel trusses. The roof of the building has begun to fail and visible holes of varying size have developed. All windows have been covered with insulation and aluminum siding on the exterior surfaces. The floor of the building is covered primarily with wooden blocks sealed in tar over a concrete sub slab. In areas were it has been necessary to replace the wooden blocks either firebrick or asphalt blocks are present. Several areas of the floor also contain a three inch lift of concrete on the sub slab. All Crane property surrounding the building is either paved or concrete except for a small area just outside the west end of the building.

Methodology

The work direction for the characterization activities was contained in WI-95-026, Characterization of Chapman Valve.

To aid in identification of areas within the building the east-west support columns were numbered 1 to 30 and the north-south columns were labeled AA, A, B, and C. The northwest corner of each section was designated as the name of that section or grid block and rooms were designated with the number of the northwestern most column number (see Figure 1 in Attachment 1).

The overhead trusses near the roof were scanned to determine if they were above guidelines for alpha and/or beta-gamma contamination. The trusses in the building are complex back to back welded angles. A series of short trusses oriented north-south were spot surveyed approximately every 10 feet with no fewer than eight surveys per truss. A series of three trusses also run east-west the full length of the building and rest on top of the north-south trusses. Due to the difficulty of reaching the east-west trusses they were assumed to contain contamination equivalent to the north-south trusses. The survey was performed in a way as to determine if the truss was contaminated and not to determine exact portions of contamination on a truss. In six elevated areas on separate trusses a dust sample was collected and analyzed for radiological parameters. A composite sample was also collected from an area determined to be radiologically contaminated and analyzed for TCLP metals including zinc and copper.

Where it was possible to reach ceiling areas through the trusses the ceiling was scanned for contamination. If roof vents were visible from the ceiling then the vent was scanned to determine if it was above criteria.

A Field Instrument for Detection of Low Energy Radiation (FIDLER) walkover was performed on the floors in approximately the western half of the building. The walkover was abandoned because the shielding effect of the wooden blocks on the floor prevented detection of contamination between the wooden blocks and concrete. The areas around boreholes placed in the building were scanned along with the tops, bottoms and sides of concrete cores.

The wooden blocks on the floor in the northern half of grid A1 were removed and checked for contamination. Two composite samples were taken from the blocks. One sample was analyzed for radiological parameters and the other sample was analyzed for TCLP total (metals,, volatile organics, semi-volatile organics, pesticides, herbicides), flashpoint, reactivity, corrosivity, and PCBs.

The walls were scanned to determine if they were above guidelines for alpha and/or beta-gamma contamination. The measurements were single readings taken on grid intersections. Starting in the west end of the building and moving east the grid was 10 X 10 foot for the first 200 feet, 15 X 15 foot for the next 100 feet, and 20 X 20 foot for the remainder of the building.

Window sills and radiators were also scanned paying particular attention to horizontal surfaces in areas where they were accessible. In elevated areas two composite dust samples were collected and analyzed for radiological parameters.

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Radiological samples were collected from both biased and systematic boreholes within the building. See Figure 1 in Attachment 1 for locations. The location of biased boreholes was based on previous investigations from the 1991 site survey. Boreholes extended to a maximum depth of three feet, or refusal, below the bottom of the concrete. Radiological samples were taken in six inch increments. In recording the depth of the sample, the measurement began at the floor surface not the elevation of the soil. For example, if the concrete was six inches thick the first sample was recorded as six inches to one foot. In addition, the thickness of the concrete for each borehole and the presence of any reinforcement material in the concrete was recorded.

The six inch sample increment from each borehole with the highest beta gamma reading as determined with handheld instrumentation was shipped and analyzed for radiological parameters and all others were archived. A chemical sample was analyzed for TCLP total (metals, volatile organics, semi-volatile organics, pesticides, herbicides), flashpoint, reactivity, corrosivity, PCBs, and paint filter test.

The different roof levels of building 23 were identified as roof R1 through R5 from North to South. Direct measurements were taken and recorded on each accessible level of the roof to determine if contamination was present. In areas where the roof was not accessible from a manlift outside the building, cores were cut from inside and subsequently scanned and sampled. The samples were analyzed for radiological parameters.

Drains for the roof of the building empty in the storm sewer. The drains were surveyed utilizing a sewer probe and samples were collected from two laterals that collect runoff from the building and at a collection manhole at the junction of the two laterals. The samples were analyzed for radiological parameters.

Steam trenches bisecting the building (two with a north-south and one with an east-west orientation) were sampled at 15 foot intervals and composited. Three composites, one from each trench were analyzed for radiological parameters.

Water and sludge samples were collected from various pits and sumps located throughout the building. The samples were analyzed for radiological parameters.

An exterior FIDLER walkover was performed around the building exterior. It covered all areas within a 50 foot radius of the building exterior. In areas where 50 foot extended beyond the property boundary or past a nearby building the property boundary or the nearby building was the extent of the exterior scan. Survey readings were also taken under aluminum siding placed over the windows.

Quality control was performed in the field on sampling and survey efforts and was based on standard sampling protocols. Informational chemical sampling performed in non-radiological areas did not include quality control samples.

Residual Radioactive Contamination Guidelines

DOE residual contamination guidelines governing the release of property for future use are contained in DOE Order 5400.5. The criteria used in analyzing the data of this characterization effort for surface contamination was 5,000 dpm/100 cm^2 average direct, 15,000 dpm/100 cm^2 maximum direct, and 1,000 dpm/100 cm^2 removable for alpha and beta-gamma. Soil guidelines for radium and thorium are 5 pCi/g averaged over the first 15 cm (6 in) of soil below the surface and 15 pCi/g when averaged over any 15 cm (6 in) thick layer below the surface, excluding background concentrations. The soil guideline for uranium is a site specific guideline based on site characteristics and use. A site specific uranium guideline has not yet been developed for CHV but typically ranges between 50 and 100 pCi/g for total uranium.

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Radiological Results

Data results for specific locations are discussed in the following sections and tables containing data results are located in the appendices to this technical memorandum. Radiological results are located in Appendix A, detected chemical results (undetected results are not shown) are located in Appendix B, interior survey results are located in Appendix C, exterior survey results are located in Appendix D, and decontamination test results are located in Appendix E. A table is provided on pages 9 and 10 showing page numbers of data tables. Numbers of data packages containing the results of characterization are listed in an interoffice memorandum (ref. 2).

Background

Background radiological samples were collected at three locations, two off site and one onsite. Soil sample results averaged 5.1, 0.98, and 1.7 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively. The average exposure rate from the three background locations was 9.1 μR/hr. Background samples were also collected from building material at the site and can be seen in Appendix A.

Building Interior

Floors

The concrete cores and areas adjacent to each borehole were scanned to determine if surface contamination was present on the concrete floor or wooden blocks. The areas around boreholes 3, 4, 11, and 14 (Figure 1) exhibited contamination near or above criteria. Spotty contamination was found on the floor in the west end of the building, however it was difficult to determine contamination that existed under the wooden blocks because of the shielding effects of the block.

All wooden blocks and concrete removed from the Northern half of grid block A-1 were scanned to determine if any were elevated above surface criteria. Approximately 3 percent of the pieces were elevated near or above criteria. The average level of beta/gamma surface contamination was 9,416 and 20,912 dpm/100 cm^2 for the elevated wooden blocks and concrete pieces respectively. A wood block, wood block composite, and tar sample was also taken for radiological analysis. The wood block samples were ground up and analyzed to obtain a volumetric activity and resulted in a ^{238}U activity of 50.5 pCi/g for the biased wood block and 1.0 pCi/g for the composite wood block. The ^{238}U results for the biased tar sample analysis was 7,891 pCi/g.

Boreholes

Boreholes were installed inside the building to determine if subsurface contamination exists at CHV below the floor of building 23. Thirty two boreholes were installed inside the building. Compared to the historical range of site specific uranium guidelines all soil samples were below soil guidelines and averaged 2.40, 0.67, and 0.44 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively.

Walls

Results of the wall survey indicate contamination above direct criteria in the main center portion of the building on a majority of the west wall, the north wall down to column 8, and the South wall down to column 8. The levels of contamination tend to increase with increasing height on the wall and are primarily found on horizontal surfaces and behind objects on the wall (i.e., conduit, radiators, pipes, etc.). Dust samples collected on the North and South wall contained uranium-238 activities of 14.8 and 97.2 pCi/g respectively.

Five wall fixtures (radiators, electrical boxes, etc.) were removed from the wall and the wall was surveyed to obtain data from behind the fixtures. The maximum and average beta/gamma result was 4,944 and 2,272 dpm/100 cm^2.

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Contamination above surface criteria was found on the walls in rooms B1, B2, B4, B7, and B8 on the South side of the building. The contamination was not as widespread as the center portion of the building and levels of contamination were lower. The walls in the rooms on the North side of the building were not contaminated.

Overheads

The trusses in the center portion of the building contain contamination above direct criteria down to approximately truss 12/13 and elevated readings below criteria down to truss 17. Dust samples were collected on trusses 2, 3, 5, 7, and 19. ^{238}U analytical results for the dust samples were 8.143, 103, 14.1, 55.8, and 6.9 pCi/g on trusses 2, 3, 5, 7, and 19 respectively.

Contamination above surface criteria was found in rooms B1, B2, B4, B7, and B8 on the south side of the building in the overheads. The contamination was not as widespread as in the center portion of the building and levels of contamination were lower. A single survey result exceeded surface criteria in the rooms on the north side of the building in room AA-1.

Bridge Crane

The top horizontal surface of the bridge crane located in grid AA-25 was contaminated above criteria to levels up to 28,973 dpm/100 cm^2. The contamination was highest near the center of the crane. A second crane located farther east in the building was not contaminated above direct or removable criteria.

Sumps, Pits, Trenches, and Tunnel

All sediments and water samples collected from sumps, pits, trenches, and the tunnel were below applicable guidelines. Maximum water sample results were 6.4, 0.72, and 0.15 pCi/l and maximum sediment sample results were 10.9, 1.2, and 1.7 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively.

Building Exterior

Grounds

Results from an exterior walkover out to 50 foot from the building were all near background and did not indicate the presence of any contamination.

Boreholes

Boreholes were installed outside of the building to determine if subsurface contamination exists at CHV outside and in the vicinity of building 23. Six biased boreholes were installed and the locations were determined based on historical information and photographs. All exterior soil samples were near background, below guidelines, and averaged 1.9, 0.67, and 0.49 pCi/g for ^{238}U, ^{232}Th, and ^{226}Ra respectively.

Surface

The exterior north and south wall survey results indicated elevated radiological conditions; however, except for a single reading on a window sill on the south side (10,743 dpm/100 cm^2) the readings were below direct and removable criteria.

Survey results from the east wall of the building did not indicate the presence of any contamination.

The exterior west wall survey results indicated elevated radiological conditions; however, except for a single reading on a window sill (8,117 dpm/100 cm^2) the readings were below direct and removable criteria. The elevated readings were primarily located near the lower southern most window that contained an exhaust to a uranium chip burner during the uranium processing in the 1940s.

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Survey measurements were also obtained from behind aluminum siding that had been placed over the exterior side of the windows. All readings were below direct and removable surface criteria except for a reading taken near the location of the chip burner exhaust. The beta gamma result at this location was 6,418 dpm/100 cm^2.

Storm Sewers

All building and roof drains exit the building on the north side and are collected in two lateral storm sewers parallel to the north wall of the building. The laterals drain to a collection basin located outside the building at approximately Grid a-24. The sewer then runs north through the rest of the Crane property where a grease and sediment trap is located. Samples were collected in each lateral, at the collection basin, and at the grease/sediment trap. Maximum sediment and water sampling results were 3.3, 0.85, and 0.44 pCi/g and 2.7, 0.21, and 0.47 pCi/l for ^{238}U, ^{232}Th, and ^{226}Ra respectively. All samples collected from the storm sewer system at CHV were below guidelines.

The drain pipes from the roof were accessed in three locations and a sewer probe was snaked down the pipe. A total of 36 feet of pipe was scanned at one foot intervals. Measurements were elevated above background (18 CPM) and averaged 21.8 CPM with a maximum of 48.1 CPM. Due to the inability to source check the sewer probe it is not possible to determine definite contamination levels. The numbers resulting from the survey of the drain lines indicate only that the drain lines contain contamination above background. Further investigation will be necessary to determine if contamination above guidelines exist in the pipes.

Roof

Survey results of the 11 cores from roofs R2, R3, and R4, survey results from roof R1 and R5, and FIDLER measurements indicate contamination present to levels as high as 146,722 dpm/100 cm^2. Roof R1 is contaminated above surface criteria down to approximately column 5. Roof R2 is contaminated above surface criteria down to approximately column 6. Roof R3 is contaminated above surface criteria down to approximately column 9, and Roofs R4 and R5 are contaminated above surface criteria down to approximately column 12.

Three biased samples of roof material were collected in areas where survey results indicated the presence of contamination. The average sample results for ^{238}U, ^{232}Th, and ^{226}Ra were 116, 0.97, and 0.64 pCi/g, respectively. Eight systematic samples from the roof yielded a maximum and average ^{238}U result of 213.8 and 83.2 pCi/g.

Core sample and survey results indicate that contamination is present in the tar composite and not in the roof planks in areas were the roof is in good condition. However, in areas were the roof is failed and water has been allowed to pass through the wood the rotten wood is contaminated as high as 942.8 pCi/g for ^{238}U.

Decontamination Tests

Truss Test

A decontamination test was performed on three trusses by hand brushing known areas of contamination with a wire brush. After light brushing additional measurements indicated that the areas tested were cleaned to levels below direct and removable surface criteria, for example on truss 2 the initial reading of 32,277 dpm/100 cm^2 was cleaned to 1,397 dpm/100 cm^2.

Paint Test

At twelve locations in the western end of the building, measurements were taken before and after paint removal on the interior walls. At all locations but two, the level of contamination fell significantly after

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paint removal except at locations 2 and 7 where the second measurement was slightly higher or equal to the first measurement. At locations where initial readings were high (for example, location 11 where initial readings were 50,015 dpm/100 cm^2) the second measurement after paint removal remained above direct criteria (in the case of location 11, 26,102 dpm/100 cm^2). For the 10 locations where a decrease occurred after paint removal, the decrease in beta/gamma measurements averaged 61%. A similar strategy was utilized during a data gap sampling effort at 10 locations. Of the 10 locations all decreased except location 7 where there was a slight increase after paint removal. Similarly to the initial paint test the readings after paint removal decreased by an average of 66%.

Chemical Results

A dust sample obtained from contaminated trusses in the west end of the building was analyzed for TCLP-metals, copper, and zinc. Analytical results were below RCRA guidelines for metals.

A soil sample from borehole 1 was analyzed for TCLP-total, reactivity, flashpoint, corrosivity, PCBs and paint filter. Results indicate that the soil in the sample does not contain RCRA constituents or display any RCRA characteristics.

Roof material was composited from the contaminated roof cores and analyzed for TCLP-total, reactivity, flashpoint, corrosivity, PCBs, and TOX. Results indicate that the roof material in the sample does not contain RCRA constituents or display any RCRA characteristics.

Paint chips collected during the characterization were analyzed for TCLP lead and cadmium. Analytical results were below applicable RCRA guidelines for the paint on the walls but paint from the overhead crane, electrical boxes, and structural steel failed. Based on these sampling results the wall paint is not a RCRA waste but the crane, electrical box, and structural steel paint is a RCRA waste.

Bulk asbestos sampling indicated asbestos in the composite roof of 3% and in insulation in an electrical box of 40 to 50%.

Summary

Characterization findings indicate the presence of contamination in the western end of building 23 at CHV (see Figures 1, 2, and 3). These findings are in agreement with historical process information obtained during characterization activities from a former Chapman Valve supervisor. According to the supervisor a temporary wall was constructed across the center portion of the building from floor to ceiling between columns A-7 and B-7, this is also the location in the building where contamination on the walls begins to decrease significantly. There was also a chip burner located in the southwest corner of Grid A-1 that exhausted to the atmosphere out a nearby window. The exhaust location and the shape of the roof of the building would tend to deposit more contamination on the south roof than the north roof as the survey results indicate. The uranium storage area was located in room B-4 and this room contained the concrete cores that were elevated.

Levels of contamination decreased from west to east in the building and survey and sampling results from the east end indicated near background radiological conditions. The exterior of the building except for the roof and several locations on the exterior west and south walls indicated near background radiological conditions.

Chemical sampling results indicated the presence of lead paint on the cranes, electrical boxes, and structural steel. If removal of the paint is required, appropriate RCRA regulation will be addressed. Bulk asbestos sampling was positive for the composite roof material and electrical box insulation.

Results from decontamination tests indicated that the majority of the contamination on the walls was removed with the paint. Contamination on the trusses was significantly decreased after light brushing.

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References

1. Oak Ridge National Laboratory, "Results of the Radiological Survey at the Former Chapman Valve Manufacturing company, Indian Orchard, Massachusetts", ORNL/RASA-92/1, July 1992.

2. Memo, "Chapman Valve Characterization Data Package Numbers", From Robbins to File, May 4, 1995.

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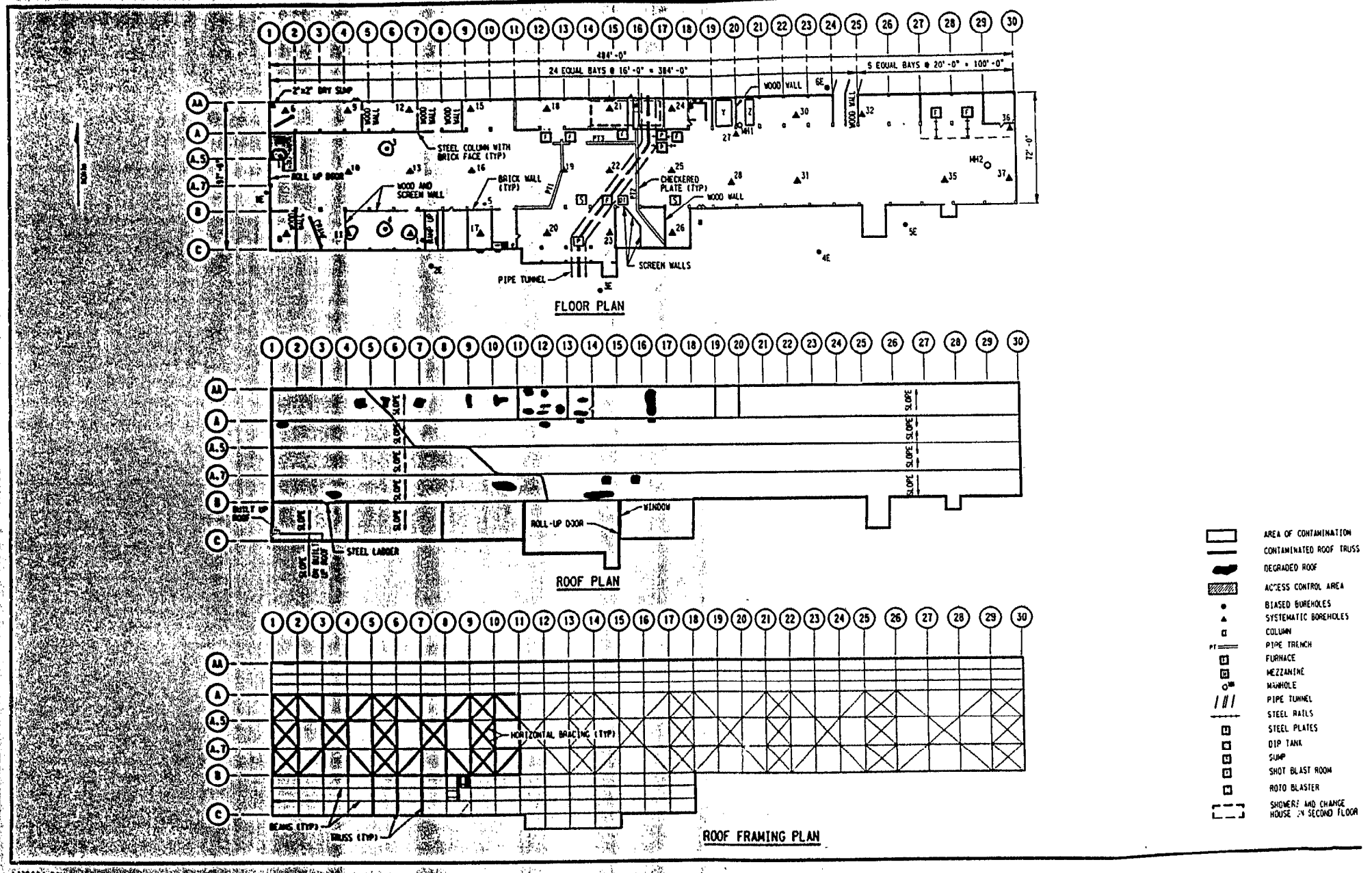
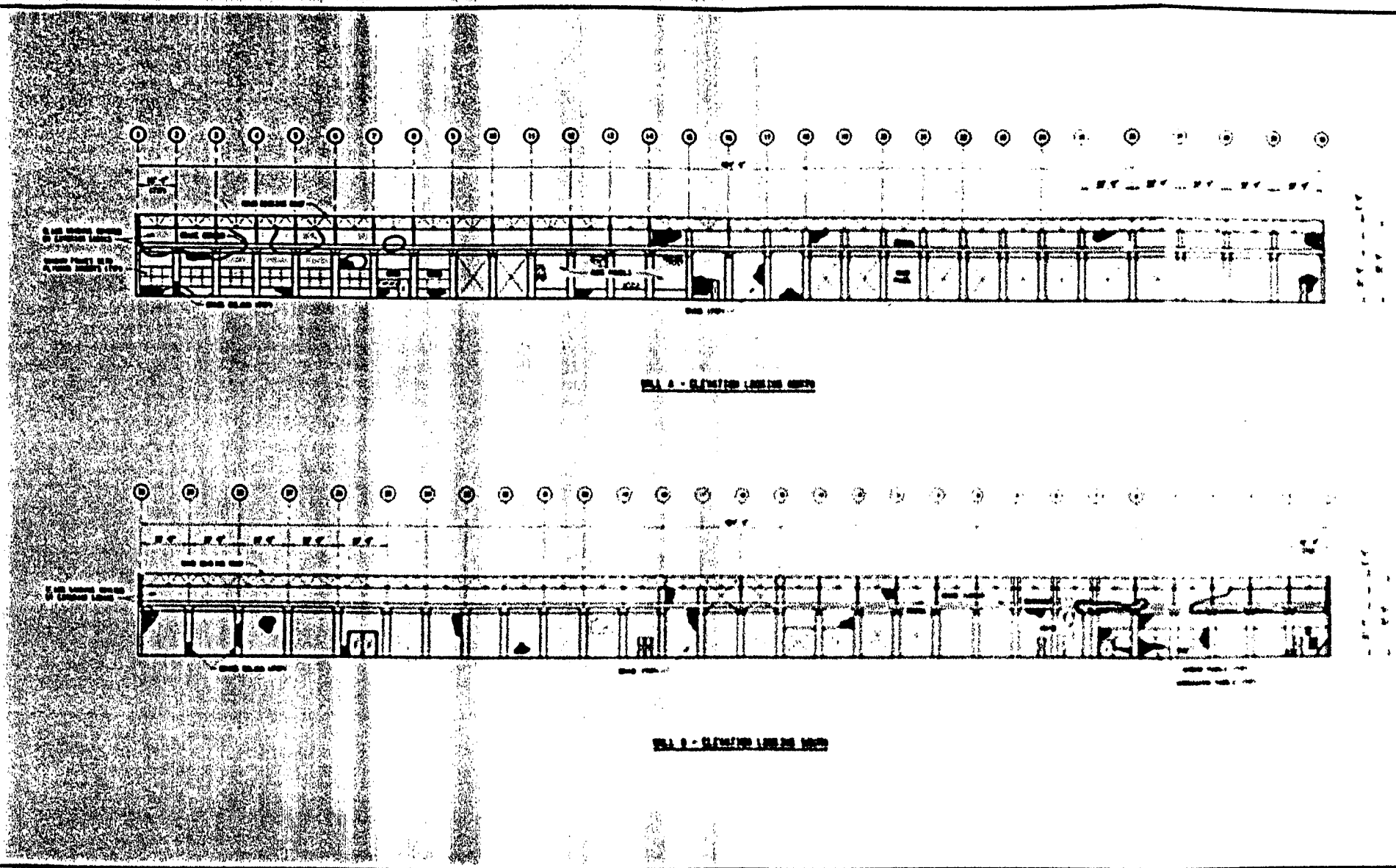
Attachments

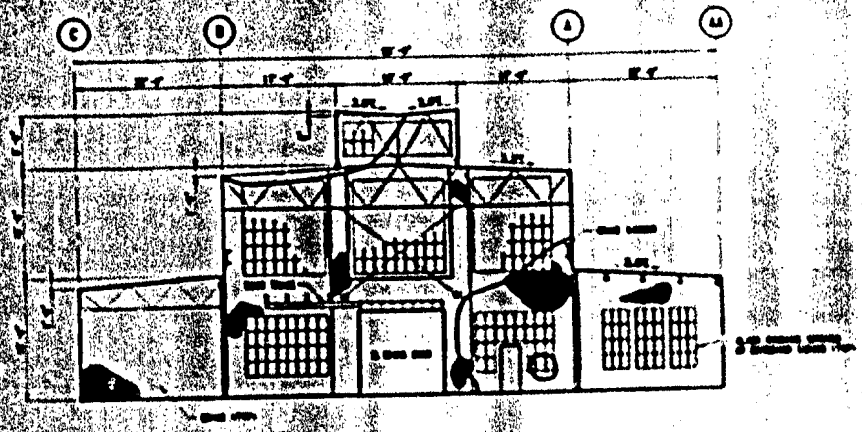


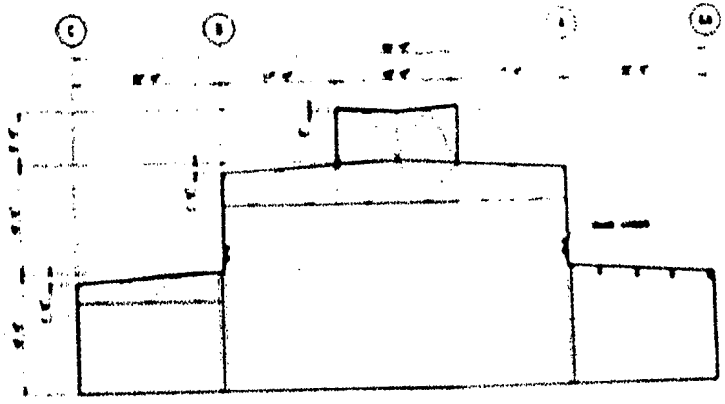
Chapman Valve - Floor, Roof, and Roof Framing Plans

Figure 1



Chapman Valve - Elevations
Figure 2





Chapman Valve - Elevations

Figure 3

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Appendix A

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SOLID MATRIX RADIOLOGICAL SAMPLING RESULTS

SAMPLE ID	LOCATION	GAMMA SPEC (pCi/g)			ALPHA SPEC (pCi/g)		
		U-238	TH-232	RA-226	U-234	U-235	U-238
133RS001	MAIN & MYTRLE ST (MYRTLE PARK)	2.8	1.1	6			
133RS002	ROUTE 21 & WRONA ST (HUBBARD PARK)	7.3	1.4	1.8			
133RS003	BUILDING 1 ON SITE (BUILT 1874)	5.2	0.45	3.3			
133RS012	BOREHOLE 1E	1.7	0.3	0.41			
133RS013	BOREHOLE 2E	1.6	0.77	0.46			
133RS014	BOREHOLE 3E	3.3	0.85	0.42			
133RS015	BOREHOLE 4E	1.6	0.53	0.42			
133RS009	BOREHOLE 5E	1.7	0.93	0.59			
133RS011	BOREHOLE 6E	1.4	0.66	0.62			
	EXTERIOR BOREHOLE AVERAGE	1.9	0.67	0.49			
133RS077	BOREHOLE 1	1.8	0.6	0.15			
133RS032	BOREHOLE 3	0.21	0.4	0.36			
133RS034	BOREHOLE 4	2.9	0.64	0.34			
133RS004	BOREHOLE 5	2.2	0.53	0.56			
133RS007	BOREHOLE 6	1.5	0.58	0.35			
133RS016	BOREHOLE 8	3.9	1.4	0.78			
133RS008	BOREHOLE 9	3.5	0.77	0.68			
133RS033	BOREHOLE 10	3.1	0.35	0.38			
133RS017	BOREHOLE 11	3.7	0.63	0.52			
133RS006	BOREHOLE 12	3.3	0.72	0.38			
133RS026	BOREHOLE 13	3	0.61	0.3			
133RS020	BOREHOLE 14	3.4	0.79	0.72			
133RS022	BOREHOLE 15	3.3	0.78	0.43			
133RS005	BOREHOLE 16	3.2	0.71	0.31			
133RS010	BOREHOLE 17	3.1	0.64	0.24			
133RS027	BOREHOLE 18	0.13	0.65	0.35			
133RS039	BOREHOLE 19	1.7	0.47	0.29			
133RS040	BOREHOLE 20	0.28	0.67	0.57			
133RS024	BOREHOLE 21	3.7	0.68	0.43			
133RS042	BOREHOLE 22	3.1	0.52	0.49			
133RS038	BOREHOLE 23	1.4	0.52	0.49			
133RS029	BOREHOLE 24	3.2	0.68	0.42			
133RS023	BOREHOLE 25	3	0.75	0.38			
133RS031	BOREHOLE 26	2.8	0.43	0.34			
133RS030	BOREHOLE 27	0.53	0.75	0.57			
133RS028	BOREHOLE 28	1.6	0.53	0.4			
133RS036	BOREHOLE 30	2.1	1	0.48			
133RS037	BOREHOLE 31	1.8	0.6	0.43			
133RS021	BOREHOLE 32	3.1	0.51	0.53			
133RS035	BOREHOLE 35	3	0.73	0.56			
133RS041	BOREHOLE 36	3.1	0.8	0.55			
133RS025	BOREHOLE 37	0.04	0.86	0.33			
	INTERIOR BOREHOLE AVERAGE	2.4	0.67	0.44			
133RS043	TRUSS 2 DUST	4167	1.5	0.54	8207	417.8	8143
133RS044	TRUSS 3 DUST	103	0.59	0.35			
133RS045	TRUSS 5 DUST	14.1	0.63	0.49			
133RS046	TRUSS 7 DUST	55.8	0.24	0.71			
133RS047	TRUSS 19 DUST	6.9	0.54	0.58			

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SOLID MATRIX RADIOLOGICAL SAMPLING RESULTS

SAMPLE ID	LOCATION	GAMMA SPEC (pCi/g)			ALPHA SPEC (pCi/g)		
		U-238	TH-232	RA-226	U-234	U-235	U-238
133RS078	SOUTH WALL-WEST END DUST COMPOSITE	97.2	0.99	0.49			
133RS085	NORTH WALL-WEST END DUST COMPOSITE	14.8	0.3	0.23			
133RS018	ROOM B-2 CEILING WOOD	942.8	1.1	0.72			
133RS019	ROOM B-2 ROOFING MATERIAL	126.2	1.1	0.59			
133RS079	ROOFING MATERIAL	NO DATA	0.8	0.97			
133RS080	ROOFING MATERIAL COMPOSITE	103	1	0.35			
	BIASED ROOFING MATERIAL AVERAGE	116	0.97	0.64			
133RS048	ROOM AA-1 TRENCH/SUMP SEDIMENT	6.9	0.73	0.42			
133RS049	PIPE TRENCH 1 SEDIMENT	2.5	0.56	0.16			
133RS050	PIPE TRENCH 2 SEDIMENT	0.82	0.8	0.46			
133RS051	PIPE TRENCH 3 SEDIMENT	5.8	0.66	0.39			
133RS052	A-12/13 FURNACE TRENCH SEDIMENT	4.7	1.7	0.59			
133RS053	A-17/18 PIT SEDIMENT	10.9	1.4	0.91			
133RS054	MANHOLE SEDIMENT (E131*N13)	2.8	1.3	1.2			
133RS055	BATHROOM FLOOR DRAIN SEDIMENT	0.96	0.93	0.52			
133RS056	A-20/21 HOPPER SEDIMENT	3.9	0.38	0.21			
133RS057	MANHOLE SEDIMENT A-20	2.7	0.93	0.63			
133RS058	A-27/29 FURNACE PIT SEDIMENT	1.3	0.48	0.6			
133RS059	AA-1 DRYWELL SEDIMENT	8.9	0.46	0.48			
133RS070	WEST STORM DRAIN SEDIMENT	2.8	0.76	0.57			
133RS071	COLLECTION STORM DRAIN SEDIMENT	2.7	0.85	1			
133RS072	EAST STORM DRAIN SEDIMENT	0.67	0.54	0.43			
133RS060	AA-19/20 SAND BLAST ROOM SAND	0.17	0.4	0.21			
133RS081	WOOD BLOCK	50.5	0.23	0.15			
133RS086	TAR SAMPLE FROM FLOOR		0.22	0.59	7653	228.4	7891
133DG001	WOOD BLOCK COMPOSITE				1.0	0.06	1.0
133DG004	ROOF CORE 1 (R58)				45.4	2.6	50.5
133DG005	ROOF CORE 2 (R56)				12.7	0.52	11.1
133DG006	ROOF CORE 3 (R45)				12.7	0.93	13.4
133DG007	ROOF CORE 4 (R47)				56	2.7	57.7
133DG008	ROOF CORE 5 (R54)				129.1	7.2	126.7
133DG009	ROOF CORE 6 (R53)				211.3	10.9	213.8
133DG010	ROOF CORE 7 (R43)				82.1	4.1	76.7
133DG011	ROOF CORE 8 (R42)				116.5	6.5	114
	SYSTEMATIC ROOFING MATERIAL AVERAGE				83.2	4.4	83.0
133DG012	SEDIMENT FROM STORM SEWER TRAP	3.3					
133RS087	BACKGROUND WOOD BLOCK	1.2	0.44	0.33			
133RS088	BACKGROUND WOOD BLOCK	2.2	0.66	0.46			
133RS089	BACKGROUND FIREBRICK	10.8	2.6	2.4			
133RS090	BACKGROUND FIREBRICK	1.9	2.6	2.3			
133RS091	BACKGROUND ASPHALT BRICK	0.76	0.45	0.38			
133RS092	BACKGROUND ASPHALT BRICK	1.5	0.46	0.3			
133RS093	BACKGROUND CEMENT	1.3	0.72	0.45			
133RS094	BACKGROUND CEMENT	2.0	0.46	0.33			
133RS095	BACKGROUND ROOF WOOD	1.2	0.34	0.23			
133RS096	BACKGROUND ROOF WOOD	1.0	0.36	0.19			
133RS097	BACKGROUND BRICK	3.8	1.2	1.1			
133RS098	BACKGROUND BRICK	3.8	1.3	1.3			

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LIQUID MATRIX RADIOLOGICAL SAMPLING RESULTS

SAMPLE ID	LOCATION	ALPHA SPEC (pCi/l)						
		U-234	U-235	U-238	RA-226	TH-228	TH-230	TH-232
133RS061	A-17/18 PIT WATER	0.59	0.04	0.95	0.34	0.03	0.35	0.04
133RS062	A-27/29 FURNACE PIT WATER	0.68	-0.01	0.6	0.09	-0.05	0.11	-0.01
133RS063	A-20/21 HOPPER WATER	0.09	-0.02	0.1	0.12	-0.02	0.14	0.04
133RS064	A-12/13 FURNACE PIT WATER	1.4	-0.01	1.5	0.22	-0.03	0.41	0.02
133RS065	MANHOLE WATER	0.08	0.11	0.12	0.31	0.1	0.38	0.03
133RS066	MANHOLE WATER A-20	0.46	-0.01	0.57	0.13	-0.02	0.08	0.11
133RS067	A-12 SUMP WATER	2.2	0.15	2.7	0.09	-0.01	0.17	0.02
133RS068	A-14 TUNNEL ACCESS WATER	1.3	0.09	1.1	-0.01	0.1	0.37	0.04
133RS069	AA-19/20 SAND BLAST PIT WATER	0.29	0.04	0.24	-0.02	0.05	0.16	0.03
133RS073	WEST STORM DRAIN WATER	0.25	-0.01	0.49	0.47	0.24	0.32	0.03
133RS074	EAST STORM DRAIN WATER	0.26	0.08	0.04	0.08	0.02	0.48	0.04
133RS075	COLLECTION STORM DRAIN WATER	0.65	0.1	0.67	0.44	0.15	0.19	0.21
133RS076	B-14/15 DIP TANK YELLOW LIQUID	4.9	0.61	6.4	0.72	0.26	1.3	0.15
133OG012	STORM SEWER SEDIMENT TRAP WATER	3.1	0.07	2.7				

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Appendix B

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CHAPMAN VALVE CHEMICAL RESULTS

SAMPLE LOCATION	TCLP METALS (ug/l)							
	SILVER	BARIUM	CADMIUM	CHROMIUM	COPPER	MERCURY	LEAD	ZINC
TRUSS DUST COMPOSIT	7.9	363	486	437	8200	0.66	1180	121000
WOOD BLOCK		346	5.1	12.3			41.7	
ROOFING MATERIAL		583	6.8			0.28	160	
SOIL (BOREHOLE 1)		607		5.3				
WALL PAINT			379				315	
ELECTRICAL BOX PAINT			442				6870	
TRUSS PAINT			750				71100	
CRANE PAINT			1630				405000	

SAMPLE LOCATION	VOLATILES AND SEMIVOLOTILES (mg/l)		
	PYRIDINE	2-METHYLPHENOL	2,3-METHYLPHENOL
WOOD BLOCK	0.019	0.5	1.1
ROOFING MATERIAL	0.1	0.042	0.043
SOIL (BOREHOLE 1)	0.1		

SAMPLE LOCATION	PESTICIDES AND HERBICIDES (ug/l)		
	HEPTACHLOR	LINDANE	ENDRIN
WOOD BLOCK		1.0	0.44
ROOFING MATERIAL	1.0	1.0	

SAMPLE LOCATION	INORGANICS				
	% SOLIDS	CYANIDE REACTIVE (mg/Kg)	SULFIDE REACTIVE (mg/Kg)	SOIL pH	FLASHPOINT (DEG F)
WOOD BLOCK	86	0.33	0.67	6.6	>140
ROOFING MATERIAL	91.1	0.25	0.5	6.0	96.9
SOIL (BOREHOLE 1)	94.7	0.25	0.5	7.8	>140

SAMPLE LOCATION	PCBs (ug/Kg)						
	AROCLOR 1016	AROCLOR 1221	AROCLOR 1232	AROCLOR 1242	AROCLOR 1248	AROCLOR1254	AROCLOR 1260
WOOD BLOCK	46	46	46	46	46	210	93
ROOFING MATERIAL	43	43	43	43	43	87	87

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Appendix C

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DRAINS.XLS

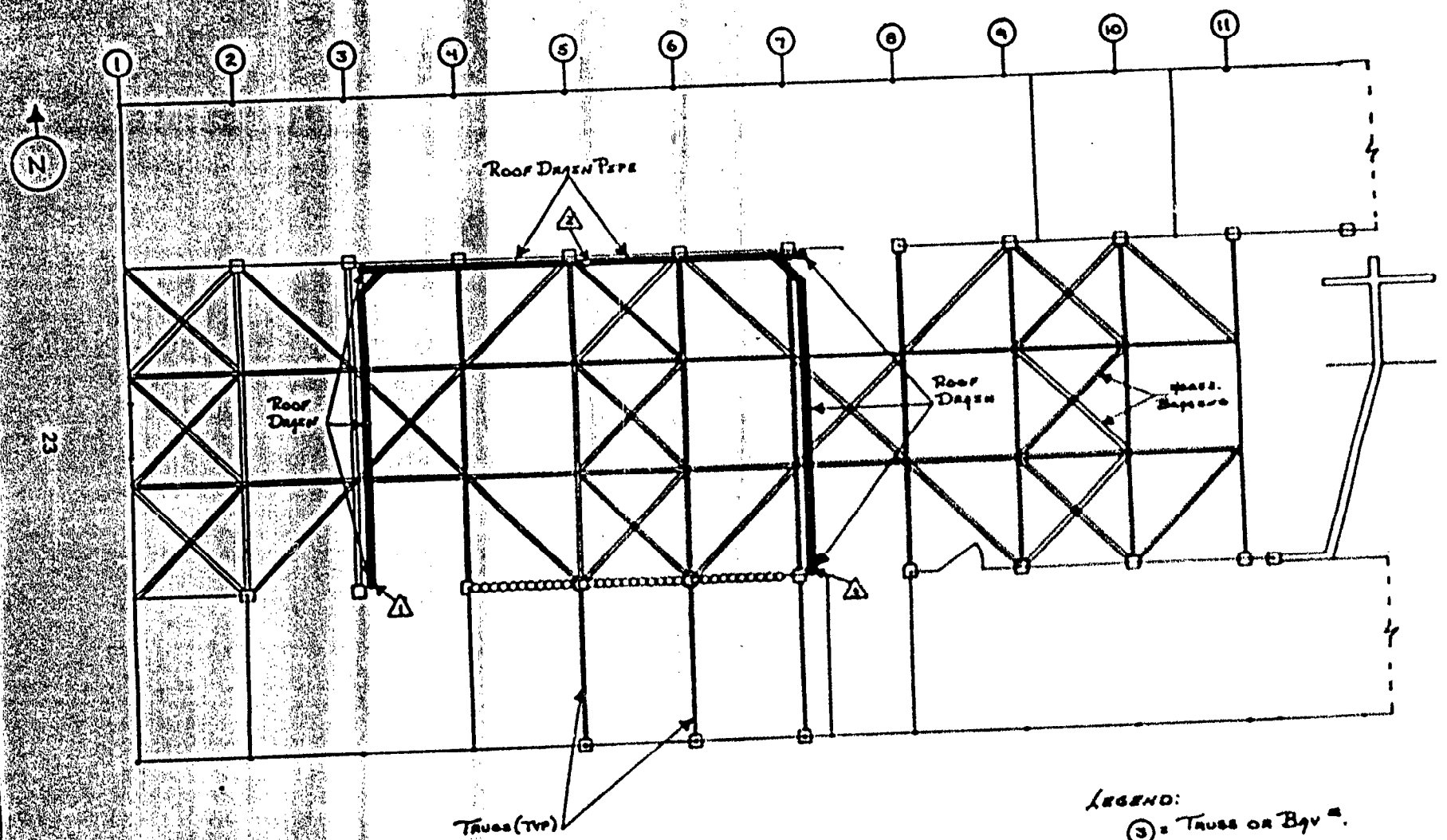
Roof Drainpipe Scan Data

Entry Point 1		Entry Point 2		Entry Point 3	
Distance in Pipe (ft)	CPM Beta	Distance in Pipe (ft)	CPM Beta	Distance in Pipe (ft)	CPM Beta
1	10.7	19	11.7	1	23.5
2	18.4	18	44.4	2	32.1
3	9.8	17	48.1		
4	8.6	16	27.7		
5	10.1	15	14.3		
6	6.5	14	29.2		
7	6.0	13	34.6		
8	5.4	12	25.1		
9	13.5	11	22.5		
10	9.9	10	18.8		
11	7.3	9	15.1		
12	13.7	8	27.0		
		7	27.2		
		6	31.1		
		5	34.5		
		4	28.5		
		3	42.0		
		2	23.7		
		1	18.1		
		0	37.1		
		-1	28.2		
		-2	34.6		
		-3	22.6		

Background equals 18 CPM.

SURVEY : [illegible]

SURVEYOR: Darin Parsons



LEGEND:

③ = TRUSS OR BAY #.

△ = SURVEY PENETRATION POINT

13857

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Interior Gamma Walkover Survey (CPM X 1000 with FIDLER)

South — North

West \ East	-3	0	3	6	9	12	15	18	21	24	27	30
0		10-18	11-16	10-22	9-30	10-12				11-15	11-15	7-12
3		11-16	10-16	9-14	9-12	9-11	9-11	9-11	10-26	10-15	10-13	11-13
6				7-13	9-12	9-11	9-11	9-11	9-17	11-15	10-14	10-14
9			9-15	9-15	9-12	9-11	9-10	9-12	9-16	10-15	9-14	9-14
12		11-15	11-15	9-15	9-11	9-11	9-10	7-12	9-14	10-15	9-13	9-13
15				9-17	9-11	9-12	9-12	9-11	10-15	9-14	9-14	9-13
18				9-11	7-11	9-11	9-11	7-13	9-26	9-14	9-13	
21				9-14	9-11	7-9	9-11	9-11	10-15	9-15	9-13	9-12
24				9-13	9-11	7-10	9-18	9-19	9-14		11-13	10-14
27				9-14	7-11	7-11	9-12	9-15	9-15		10-13	
30				9-14	9-12	7-10	7-15	10-13	9-17			
33				9-15	9-12	7-17	9-28	10-21	9-17			
36				9-15	9-12	7-14	11-29	10-18	9-12			
39				9-16	9-12	7-11	12-25	9-17	9-13	9-13	9-12	
42				9-15	10-12	7-12	9-17	9-18	9-12	10-12	10-11	
45				9-11	10-13	9-13	9-10	9-11	9-13	10-13	10-13	
48	10-13	13-16	6-13	9-11	9-12	9-11	9-10	9-10	9-11	11-12	11-13	
51		9-15	11-14	7-12	9-11				9-13			
54	9-12	7-13	9-13	7-13		4-11			9-12			
57						8-10	9-12	9-12	9-14			
60						7-9	7-10	7-11	7-12			
63						7-10	8-12	8-12	7-14			
66						7-10	9-12	8-10	9-11			
69						7-11	9-13	9-11	9-12	9-12	13-15	13-16
72						7-10	9-13	7-10	9-13	9-13	11-16	13-14
75						8-10	8-14	9-10	9-11			
78						8-10	8-11		11-13			
81	-3	0	3	6	9	12	15	18	21	24	27	30

South — North

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Scan Data Behind Wall Fixtures in the West End

FIXTURES.XLS

Location Number	Direct (DPM/100 cm^2)		Transferrable (DPM/100 cm^2)	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	165	4895	5	46
2	527	4074	2	25
3	1105	4944	8	-50
4	47	-106		
5	8	-10		
6	8	353		
7	47	1754	5	60

1, 2, and 3 located 2 feet above floor behind radiator; 3, 6, and 9 feet east of column 5, respectively.
4 located 5.5 feet above floor on column 6 behind junction box.
5 located 5 feet above floor on column 7 behind terminal box.
6 located 4 feet above floor on column 5.
7 located 6 feet above floor on column 5 behind junction box.

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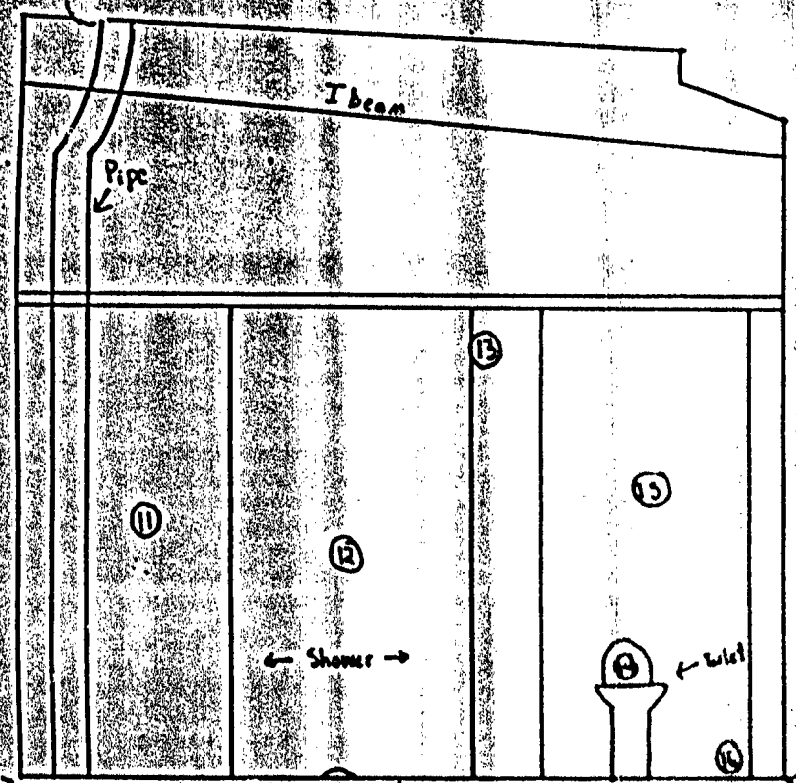
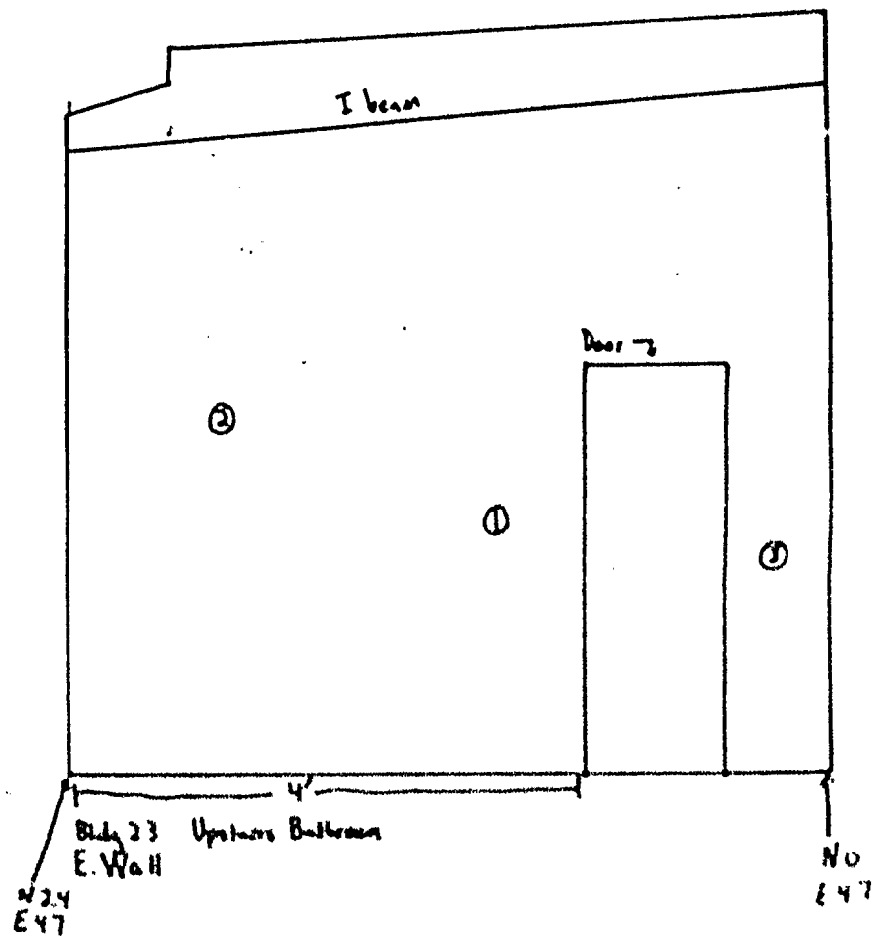
0022

Upstairs Bath Room Scan Data

BATHRM.XLS

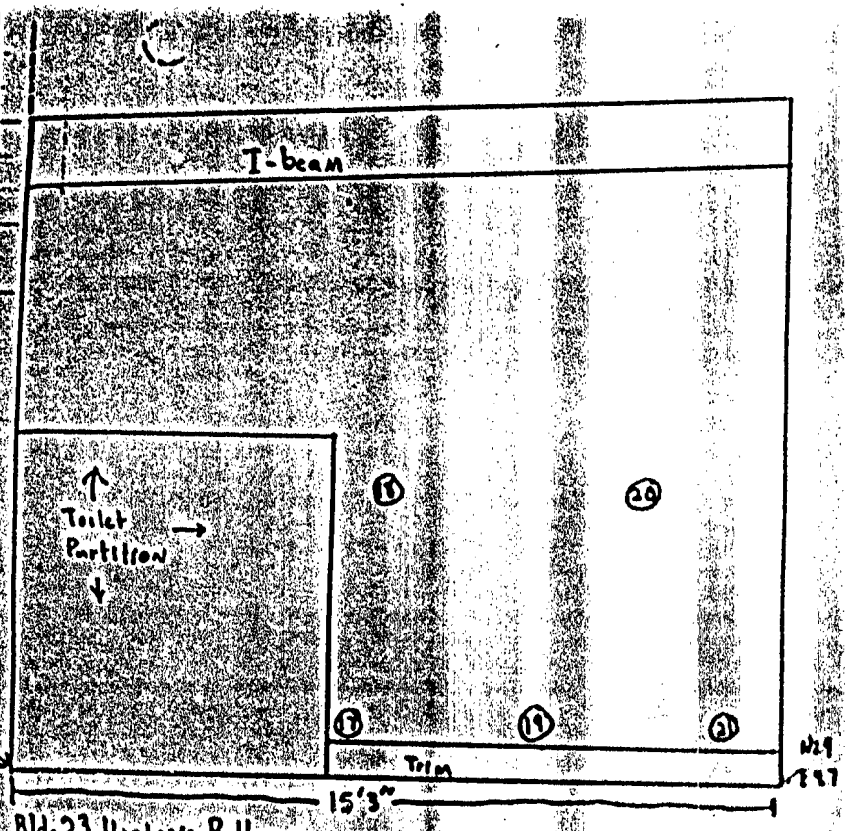
Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	49	268				
e	2	78	327				
r	3	-8	149				
l	4	21	387				
z	5	68	297				
o	6	21	1100	30	1338	2	8
n	7	11	832	78	-59	2	-75
t	8	68	387				
a	9	49	-208				
l	10	59	-119				
	11	68	416				
S	12	40	30				
u	13	21	-30				
r	14	30	2200	2	1368	2	29
f	15	49	535				
a	16	11	565				
c	17	68	-357				
e	18	21	268				
s	19	2	59				
	20	30	-268				
	21	21	119				
	22	21	-297				
	23	2	1665	2	1516	-1	-33
	24	2	-297				
	25	2	-238				

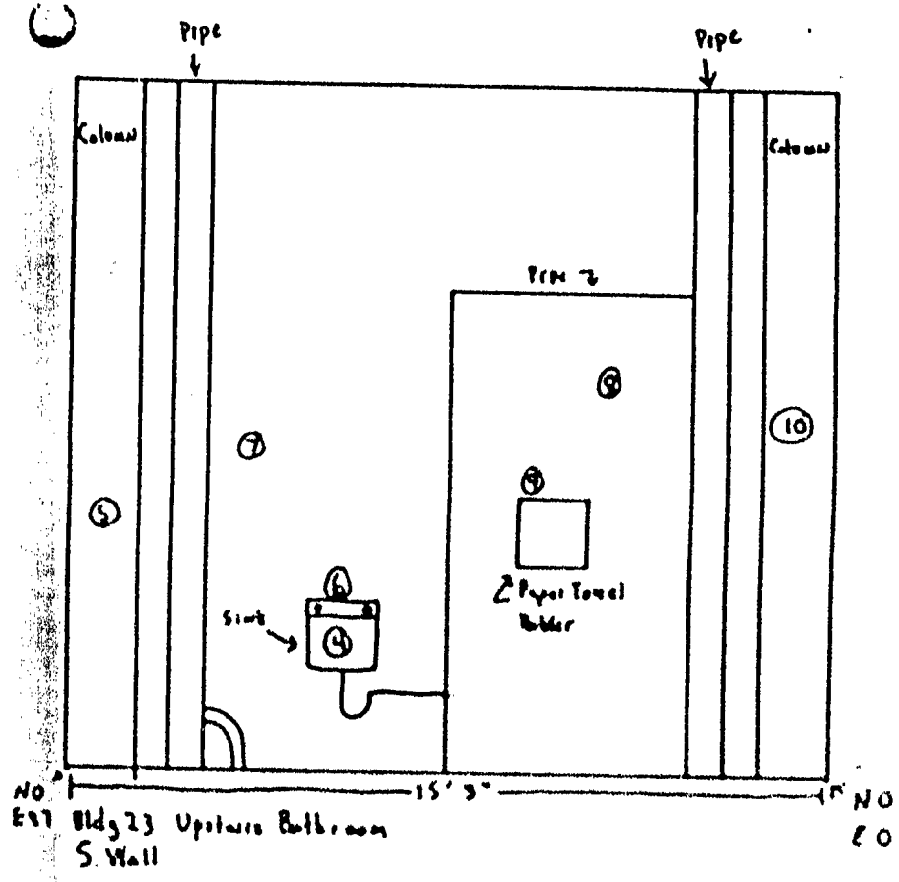


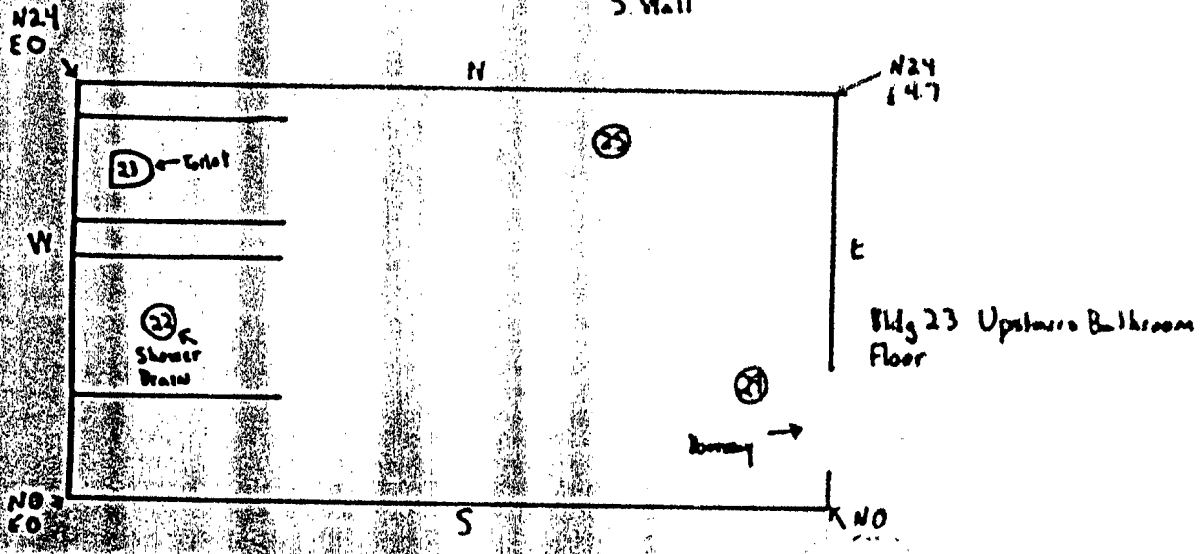


Note: Ice covers many surfaces in this room - BKM.
Grid coordinates for this room separate from downstairs coordinate system.

0-19646
L25687


I-beam
Toilet Partition
Trim
15'3"
N29
E97
Bldg 23 Upstairs Bathroom
N. Wall


Pipe
Pipe
Column
Column
Trim
Sink
Paper Towel Holder
15' 3"
N0
E97
N0
E0
Bldg 23 Upstairs Bathroom
S. Wall


N24
E0
N
N24
E47
Toilet
W
E
Shower Drain
Bldg 23 Upstairs Bathroom
Floor
Doorway
N0
E0
S
N0

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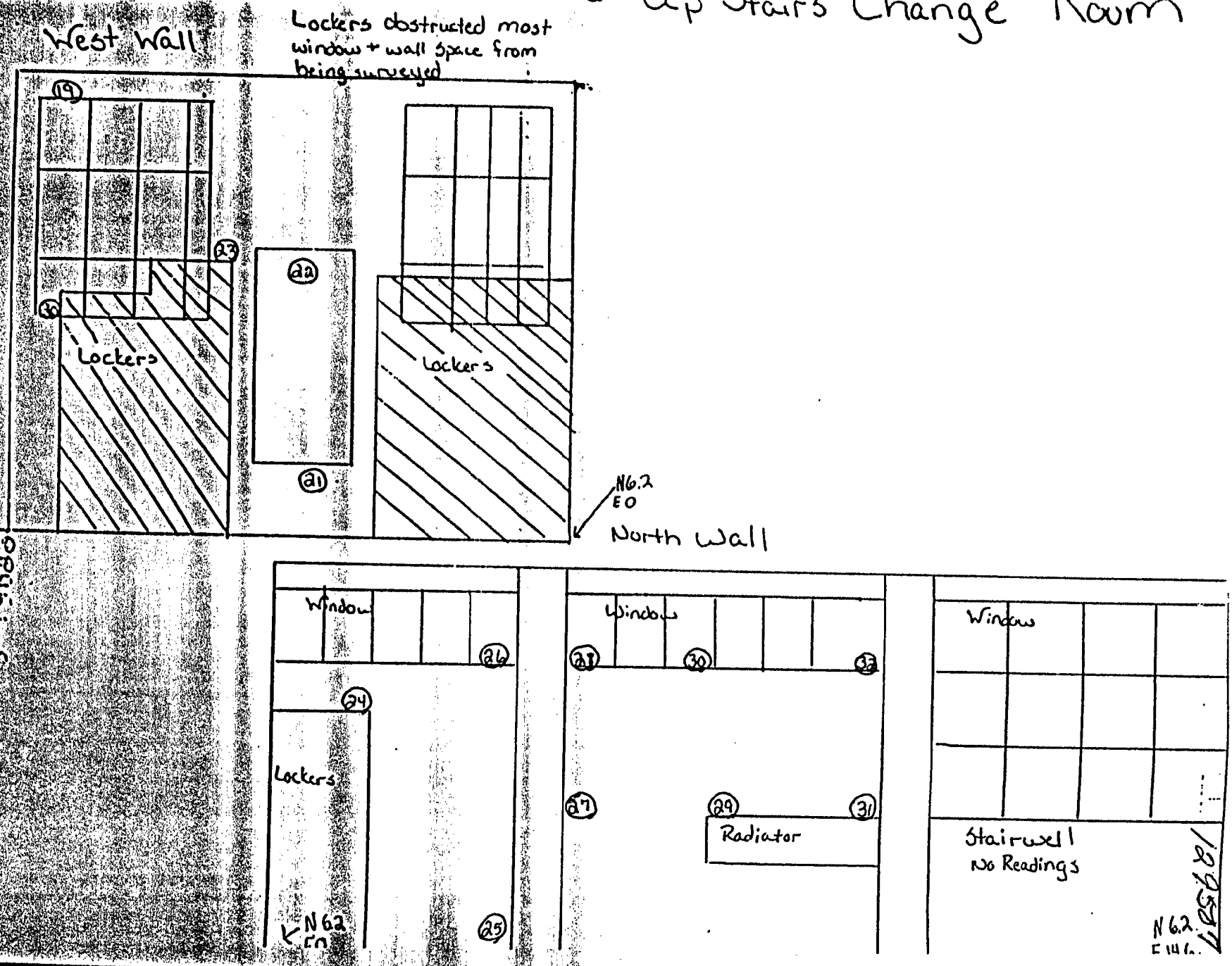
Upstairs Change Room Scan Data

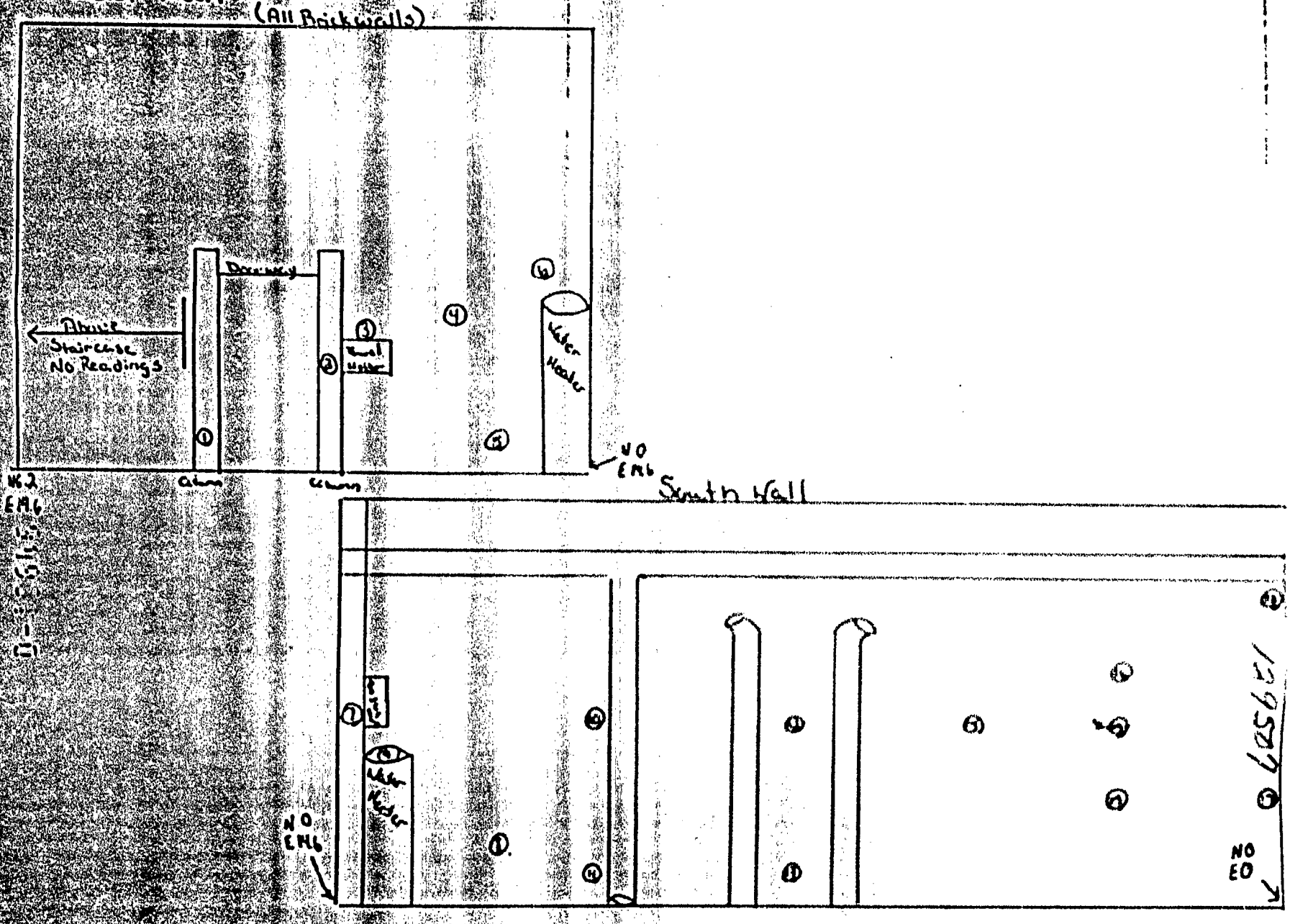
CHGROOM.XLS		Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
	Location	Before Dust Removal		After Dust Removal			
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
H	1	-44	1914	88	1308	-1	-25
o	2	58	1734	29	1308	13	-25
r	3	15	535				
i	4	88	535				
z	5	-15	535				
o	6	44	357				
n	7	44	595				
t	8	15	505				
a	9	-15	0				
l	10	88	327				
	11	-15	505				
S	12	29	1011	0	982	2	-4
u	13	-15	387				
r	14	44	387				
f	15	132	951				
a	16	15	357				
c	17	-58	832				
e	18	15	714				
s	19	58	-30				
	20	29	327				
	21	15	1189	102	238	-1	54
	22	190	-238				
	23	-15	327				
	24	-15	-59				
	25	58	654				
	26	-44	238				
	27	15	416				
	28	29	476				
	29	-73	-30				
	30	-15	208				
	31	73	89				
	32	88	357				

Up Stairs Change Room



0 0 2 2

Up Stairs Change Room



0022

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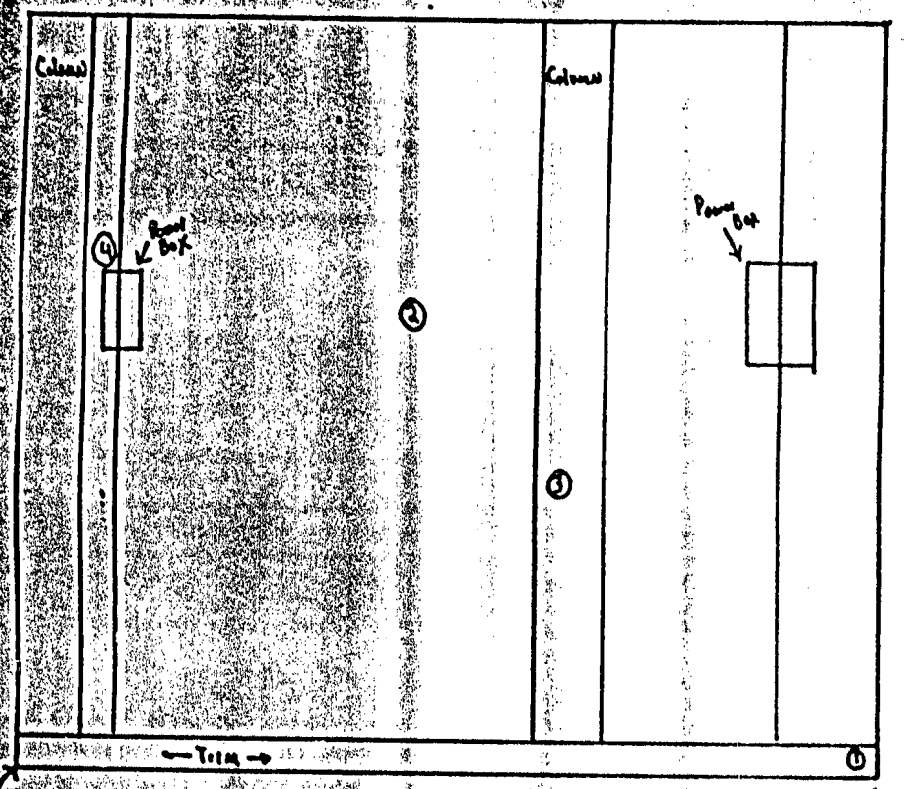
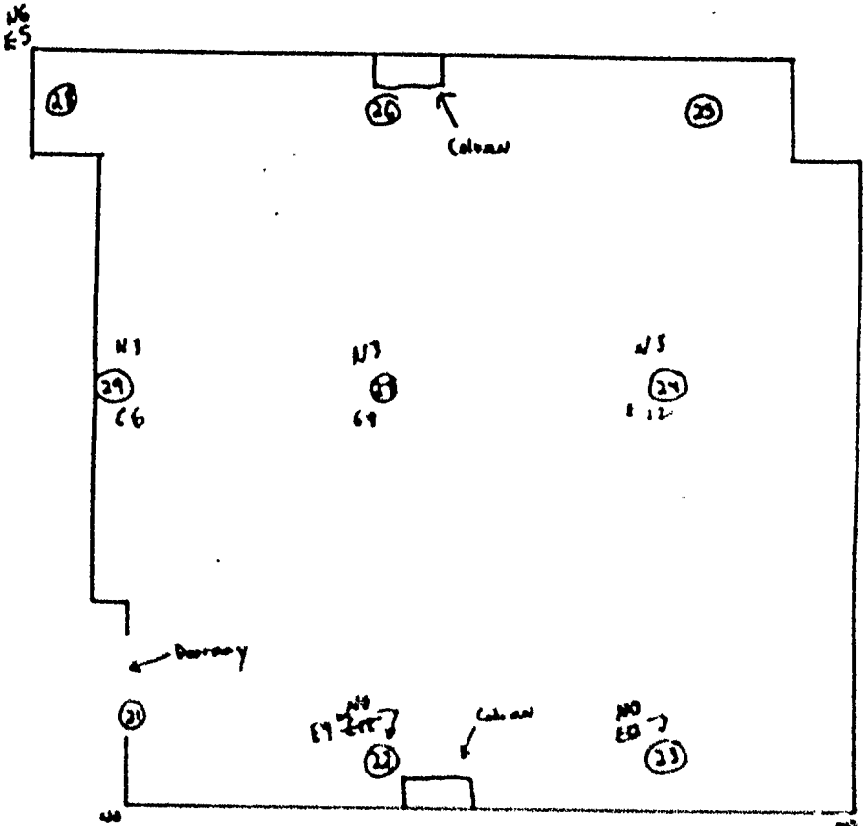
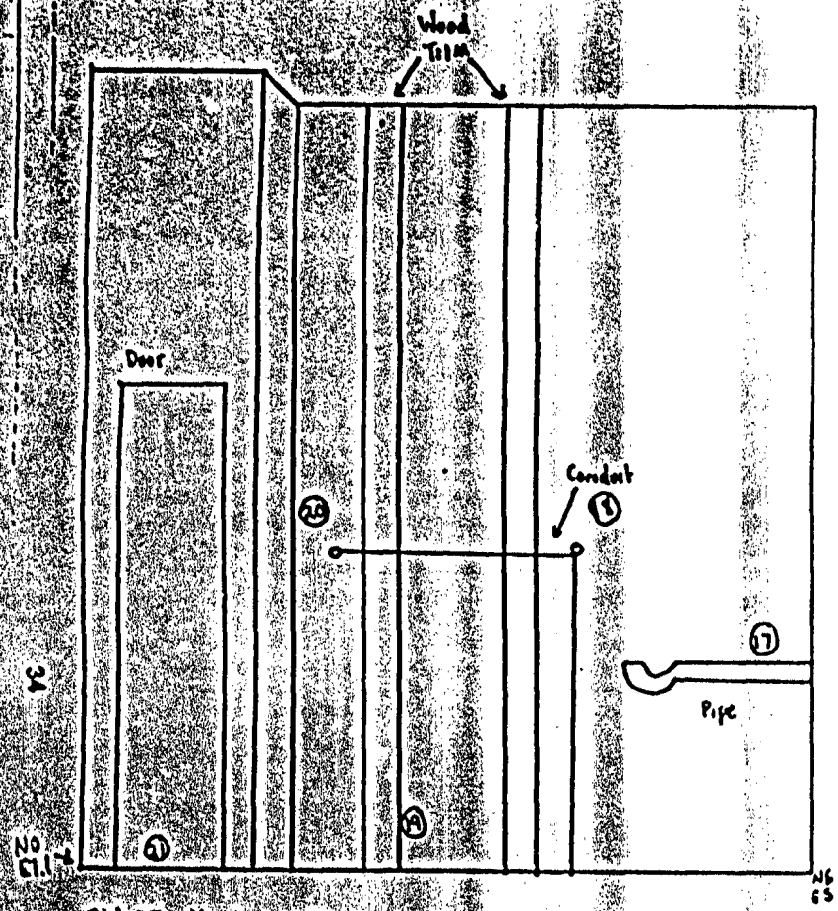
Upstairs Office Room Scan Data

OFICERM.XLS

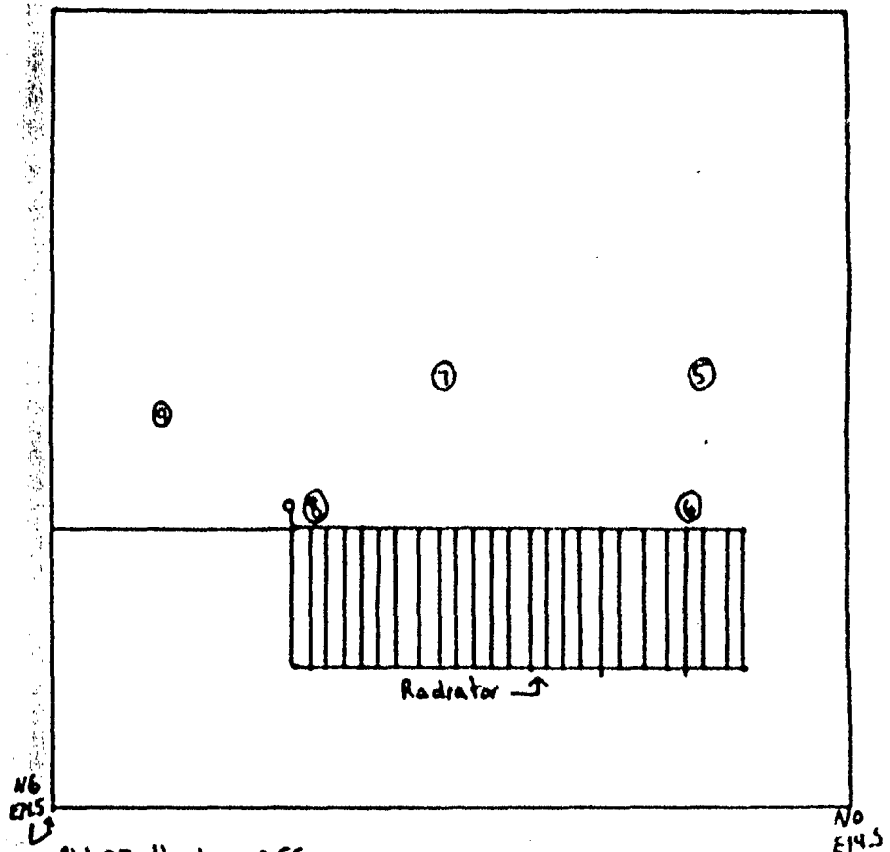
Location	Location Number	Direct (CPM/100 cm^2) Before Dust Removal Alpha	Direct (CPM/100 cm^2) Before Dust Removal Beta/Gamma	Direct (CPM/100 cm^2) After Dust Removal Alpha	Direct (CPM/100 cm^2) After Dust Removal Beta/Gamma	Transferable (CPM/100 cm^2) Alpha	Transferable (CPM/100 cm^2) Beta/Gamma
H	1	59	773				
o	2	87	1100	21	1160	-1	-8
r	3	21	1368	40	773	-1	-13
i	4	2	149				
z	5	40	803				
o	6	30	-89				
n	7	59	-298				
t	8	11	-357				
a	9	59	-476				
l	10	30	-298				
	11	30	-179				
S	12	2	-119				
u	13	59	119				
r	14	21	-238				
f	15	11	-416				
a	16	182	357				
c	17	30	-238				
e	18	21	30				
s	19	-8	-268				
	20	21	-238				
	21	11	-149				
	22	11	-357				
	23	-8	-535				
	24	2	-59				
	25	11	0				
	26	-8	119				
	27	-8	-149				
	28	49	-297				
	29	2	-595				


Column
Column
Power Box
Power Box
Trim
NO E21
Bldg 23 Upstairs Office S. Wall
NO EH5


NO E5
Column
Doorway
Column
NO E21
Bldg 23 Upstairs Office Floor



Bldg23 Upstairs Office
W. Wall

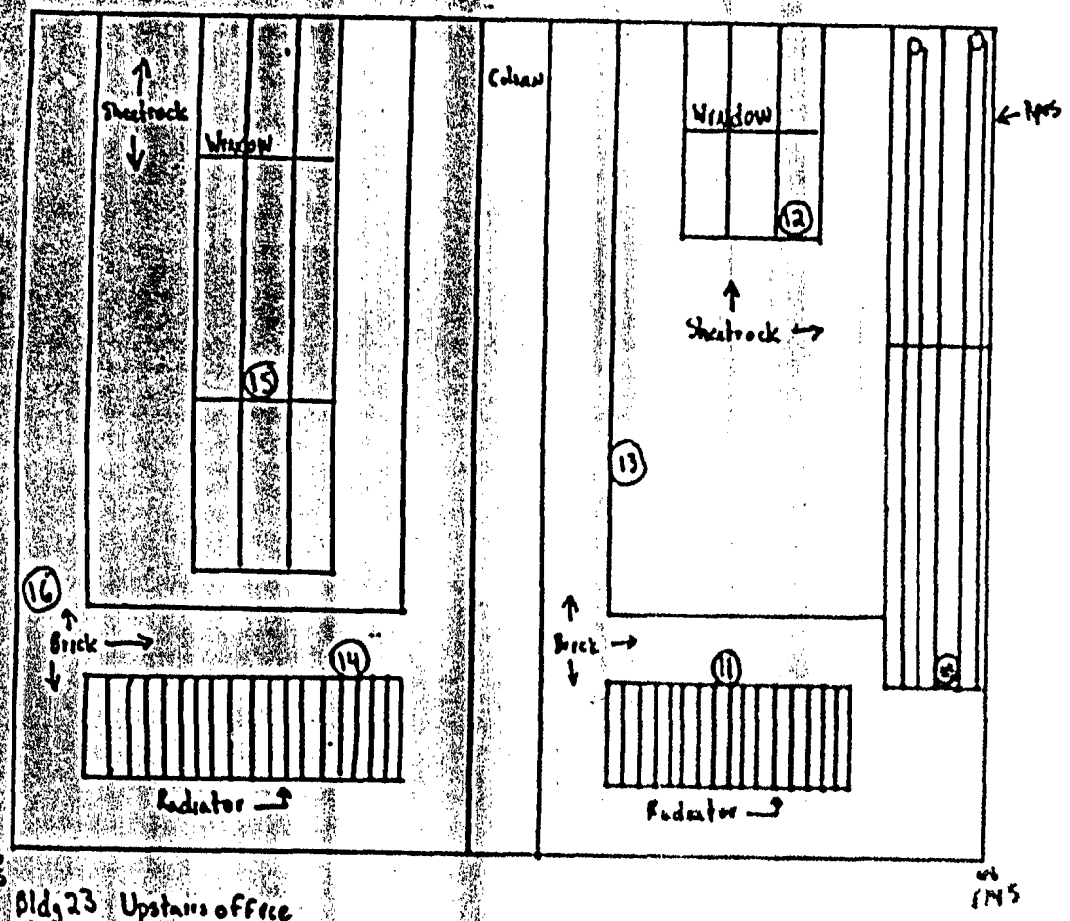


Bldg23 Upstairs Office
E. Wall

0-15546

18957

0022

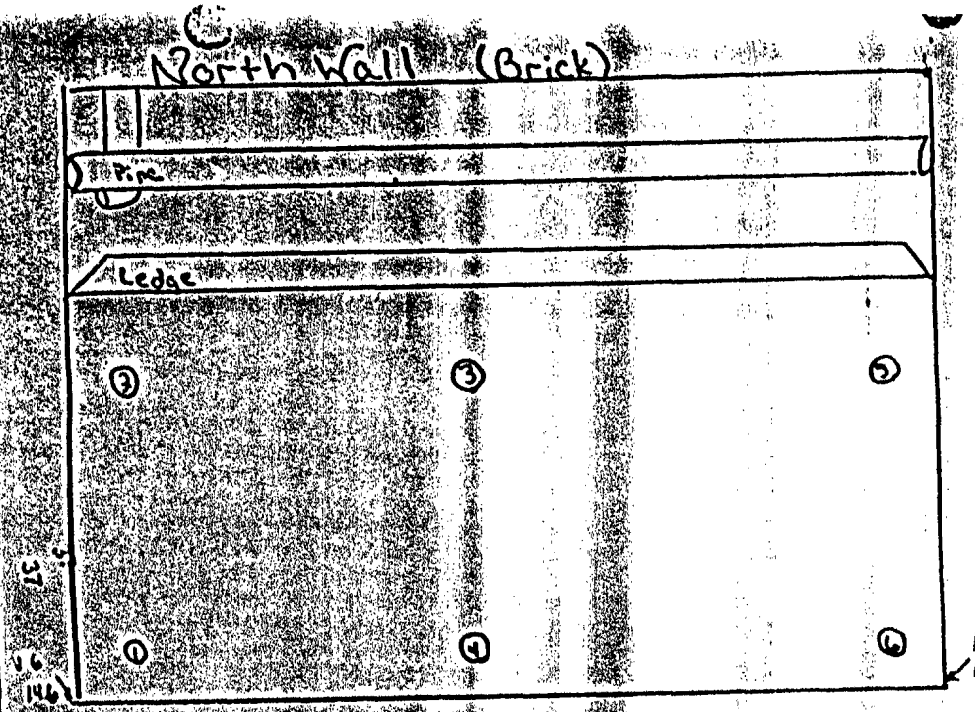


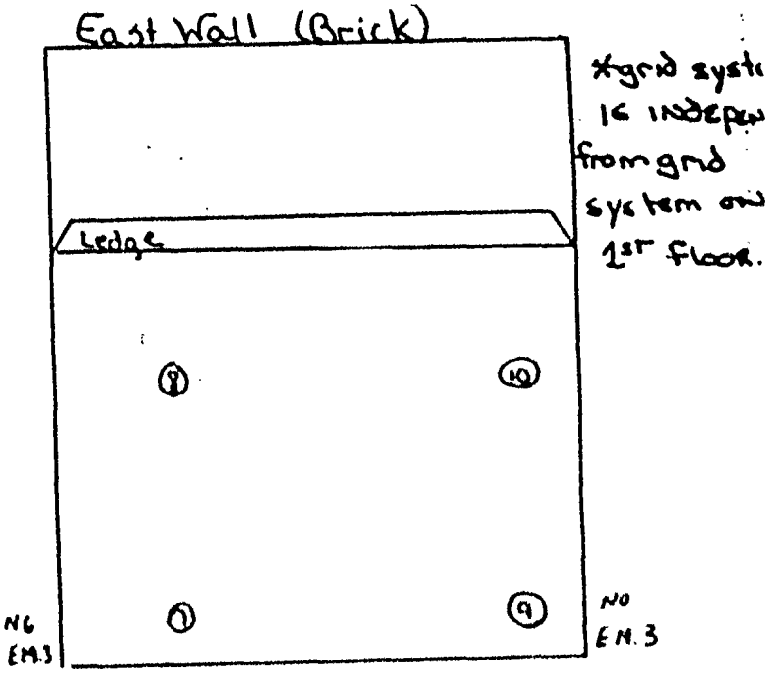
Bldg 23 Upstairs office
N. Wall

139527

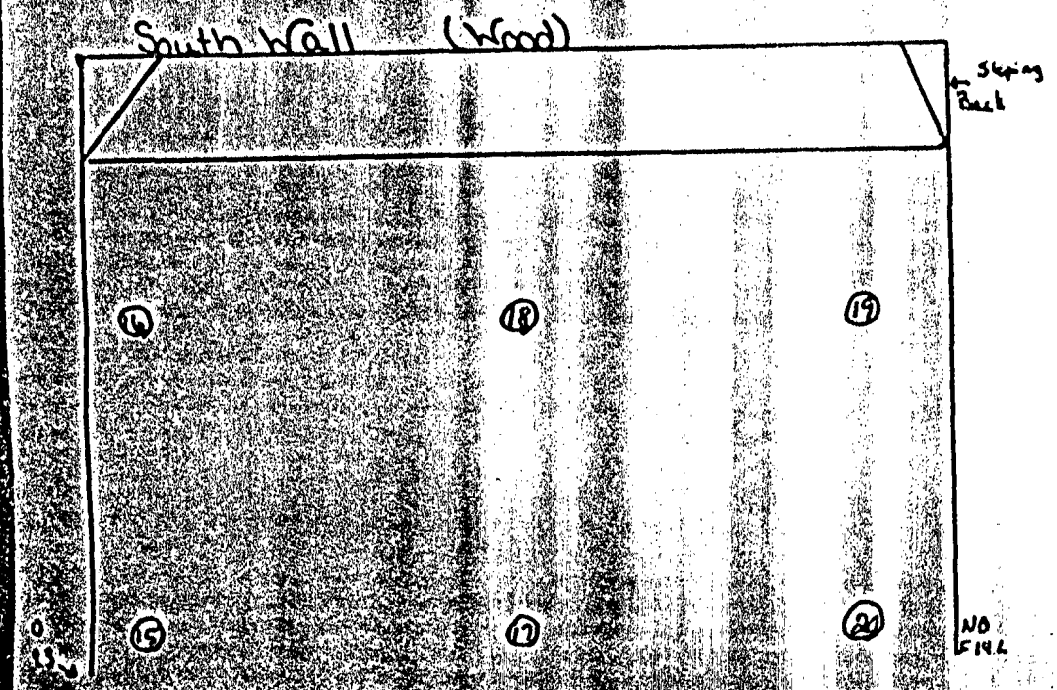
Upstairs Shower Room Scan Data

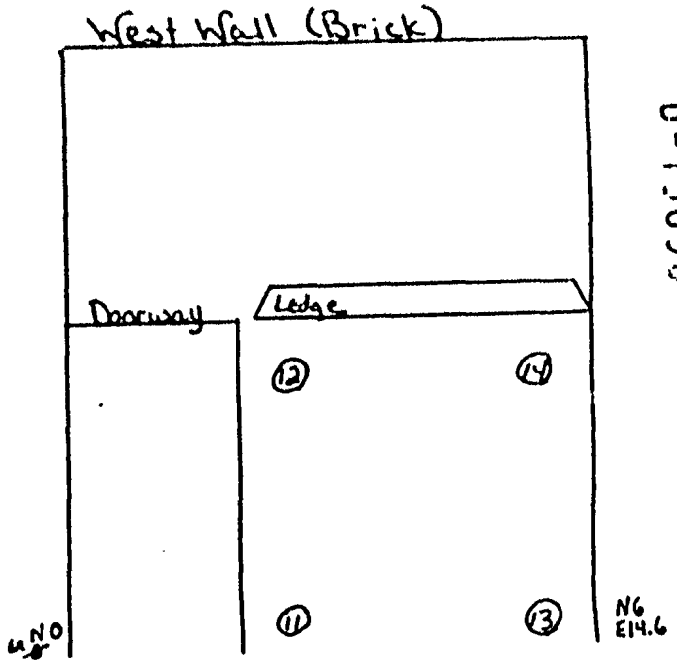
SHWRRM.XLS		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
H	1	15	1665	29	1933	2	67
o	2	190	2557	58	2230	-1	21
r	3	73	1249	88	1576	2	-4
i	4	146	1457	73	1665	-1	-17
z	5	44	1724	117	2051	2	38
o	6	117	1962	44	1457	-1	8
n	7	117	2022	117	2557	2	4
t	8	102	2141	58	1814	2	-8
a	9	88	1546	29	1992	5	17
l	10	44	2051	88	1962	2	-8
	11	73	1992	15	2170	-1	13
S	12	73	2111	88	1546	-1	71
u	13	44	2022	102	1962	2	-13
r	14	58	2111	29	1814	2	-13
f	15	29	149			2	-13
a	16	146	-149			-1	17
c	17	102	59				
e	18	0	208				
s	19	15	89				
	20	117	208				
	21	-29	2200	58	2854	-1	17
	22	-44	-238				
	23	29	1903	-29	1873	5	25
	24	0	1546	-15	1962	2	21





*grid system is independent from grid system on 1st floor.

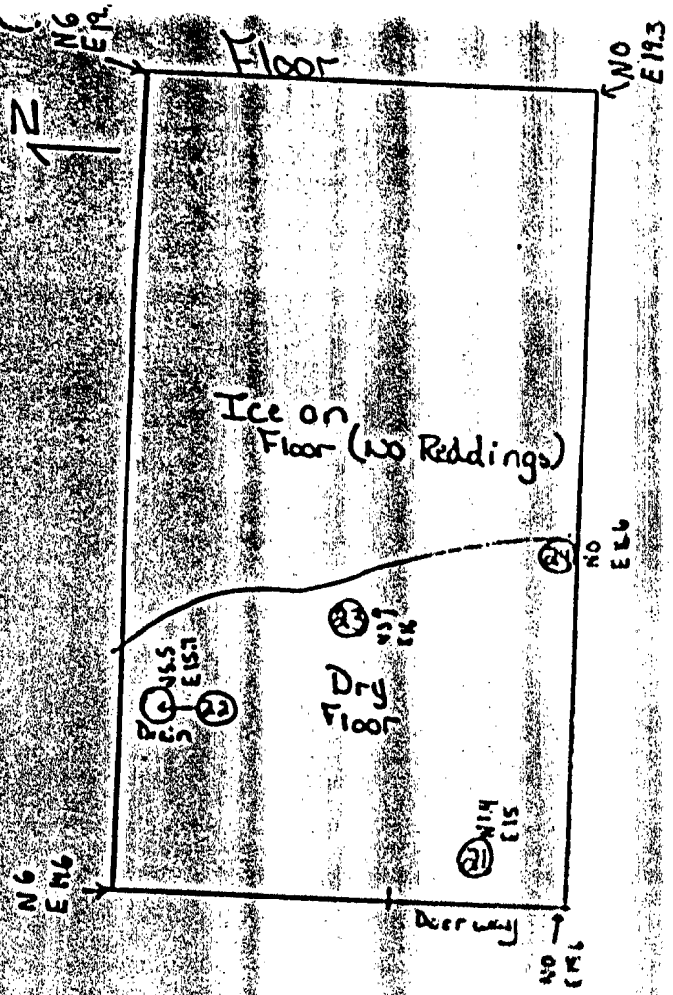




39691-0

19952

Up Stairs Shower Room



D-19693

36

* grid system is independant from grid system on 1st floor

L85621

0 0 2 2

129507

0022

Truss Scan Data

TRUSS.XLS

Truss Number	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
1	1	89	421				
1	2	316	10508		3167	27	-4
1	3	184	5919		2841	2	17
1	4	127	7389		10447	24	21
1	5	449	4558		1391	18	4
1	6	222	11808		6659	10	-25
1	7	146	3048		444	5	29
1	8	241	3344		1805	2	-17
2	1	-40	10832		7887	59	130
2	2	821	32791		18881	343	778
2	3	40	4794		6333	13	25
2	4	256	18763		21012	59	163
2	5	336	18573		11801	27	0
2	6	496	20583		25806	283	958
2	7	5966	26872		32791	69	238
2	8	363	11542		5366	7	38
3	1	188	2338		1006	8	-13
3	2	386	57532		15861	32	-13
3	3	632	45043		2190	-1	17
3	4	605	14797		2841	46	59
3	5	135	7835		2723	514	941
3	6	7708	750459	13896	19710	46	126
3	7	496	19177		2190	10	13
3	8	457	8523		3374	10	21
4	1	8	2189	135	1513	5	-37
4	2	67	2697	165	1656	28	55
4	3	67	4799	468	3542	5	38
4	4	165	5620	155	3784	11	38
4	5	76	3615	410	764		
4	6	233	4219	1135	210	19	68
4	7	145	9801	302	474		
4	8	86	2890	370	981	14	29
5	1	202	5445		414	2	-13
5	2	309	1361			-1	8
5	3	94	2545		4617	2	-8
5	4	94	39124		30482	2	4
5	5	148	4498		1598	2	-17
5	6	121	4735	13896	2959	10	38
5	7	552	30956			-1	84
5	8	363	19298		1746	10	33
7	1	229	2131	1439	6274	5	-13
7	2	175	21012	794	2308	-1	0
7	3	94	1302	256	3551	6	21
7	4	498	829				
7	5	229	1835	175	2545	-1	-4
7	6	175	3492	417	4439	10	0
7	7	94	5327	202	1778	2	-13
7	8	148	4558	94	947	5	54

129507

Truss Scan Data

TRUSS.XLS		Direct (DPM/100 cm^2)					
Truss	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Number	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
11	1	11	2368		1391	10	-33
11	2	118	1391				
11	3	145	5505		1894	27	79
11	4	-70	8168		8316	13	17
11	5	-70	6807		1864	16	67
11	6	-43	3818		5061	18	84
11	7	38	740				
14	1	56	1864	191	1273	-1	13
14	2	56	2220	83	740	5	25
14	3	56	3581	30	1036	-1	46
14	4	110	2042	-24	562	18	4
14	5	110	2752	56	977	-1	46
16	1		1332	102	1509	-1	17
16	2		2220	-59	1332	5	29
16	3		2893	129	1864	-1	8
16	4		1864	156	3107	-1	29
16	5	129	562			-1	-21
16	6	35	710			-1	4
16	7	35	385			-1	17
16	8	62	858			-1	-8
17	1	-51	-178				
17	2	15	118				
17	3	-32	740		829	2	-33
17	4	-42	858		207	-1	4
17	5	-4	1716		59	-1	17
17	6	25	385				
17	7	25	1509		237	-1	29
17	8	6	1125		355	-1	-38
19	1	3	799	-24	-266		
19	2	83	266	-105	30		
19	3	164	444	83	30		
19	4	3	562	30	266		
19	5	-24	-30				
19	6	30	-148	-24	30		
19	7	56	44				
19	8	-24	621				
21	1	-51	-89				
21	2	-42	89				
21	3	6	740		-237	-1	-38
21	4	-32	355				
21	5	-13	89				
21	6	-61	414		30	13	-38
21	7	-51	503		1095	10	-4
21	8	6	474	-4	148	-1	-13

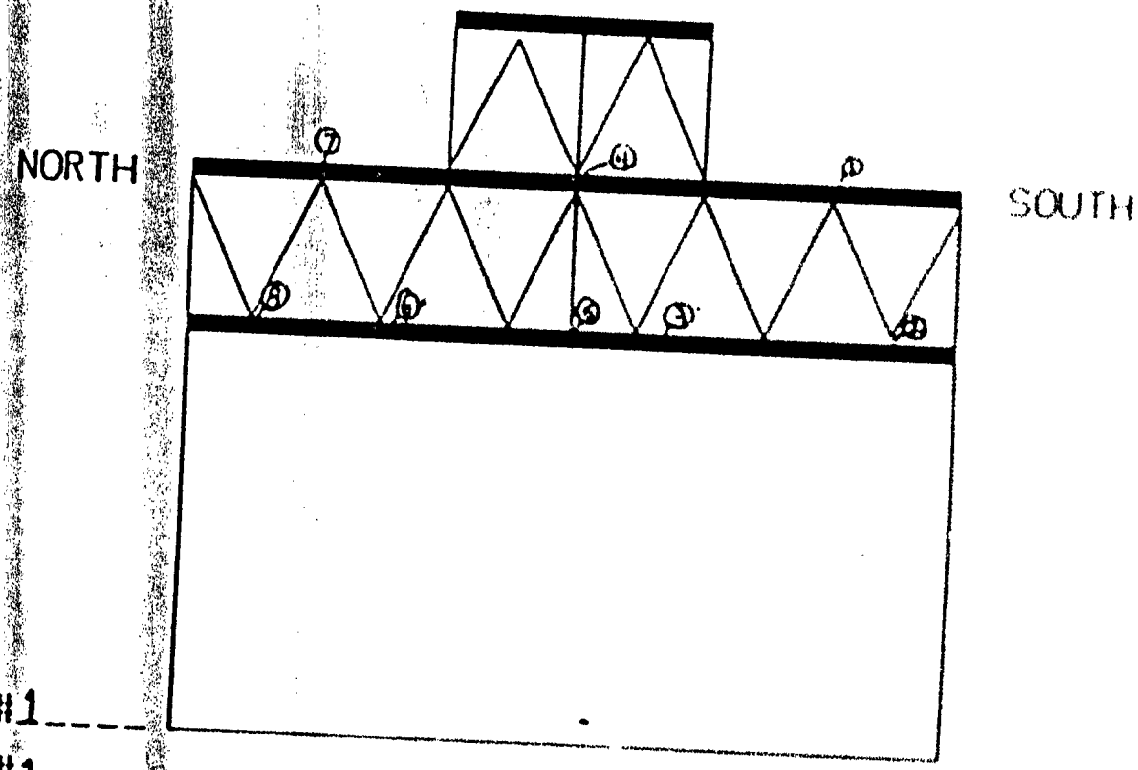
0022

129507

Truss Scan Data

TRUSS.XLS		Direct (DPM/100 cm^2)					
Truss	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Number	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
23	1	8	-266				
23	2	8	355		237	-1	59
23	3	27	-118				
23	4	27	326		30	-1	-13
23	5	-21	-148				
23	6	55	355		414	-1	25
23	7	17	148				
23	8	27	-266				
25	1	27	148				
25	2	8	355		444	2	25
25	3	-21	207				
25	4	8	-237				
25	5	-11	296		-148	2	-29
25	6	17	-326				
25	7	27	-237				
25	8	64	89				
27	1	8	-237				
27	2	55	0				
27	3	-11	-296				
27	4	-21	-178				
27	5	17	-296				
27	6	17	-326				
27	7	-11	-414				
27	8	27	-207				
27	9	-21	-266				
29	1	9	-237				
29	2	-19	-118				
29	3	9	-266				
29	4	38	237				
29	5	38	-503				
29	6	9	-118				

BUILDING #23
NORTH AND SOUTH TRUSS

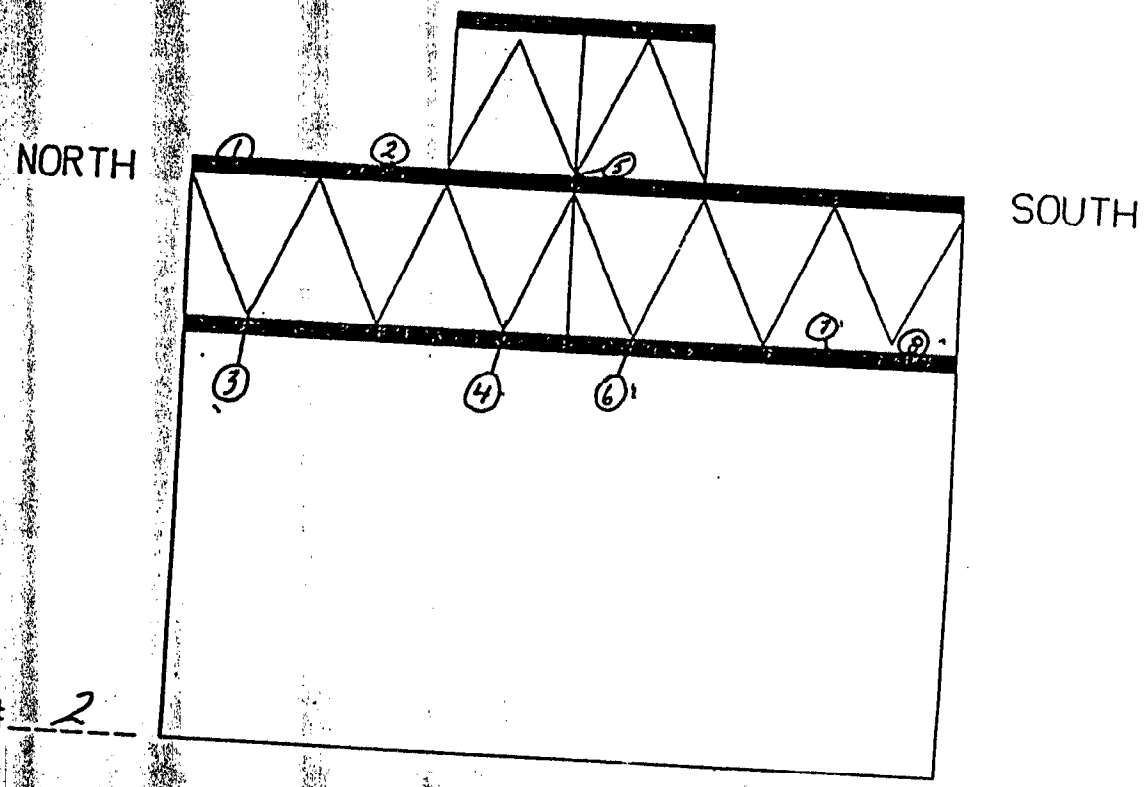


TRUSS #1____

BAY #1____

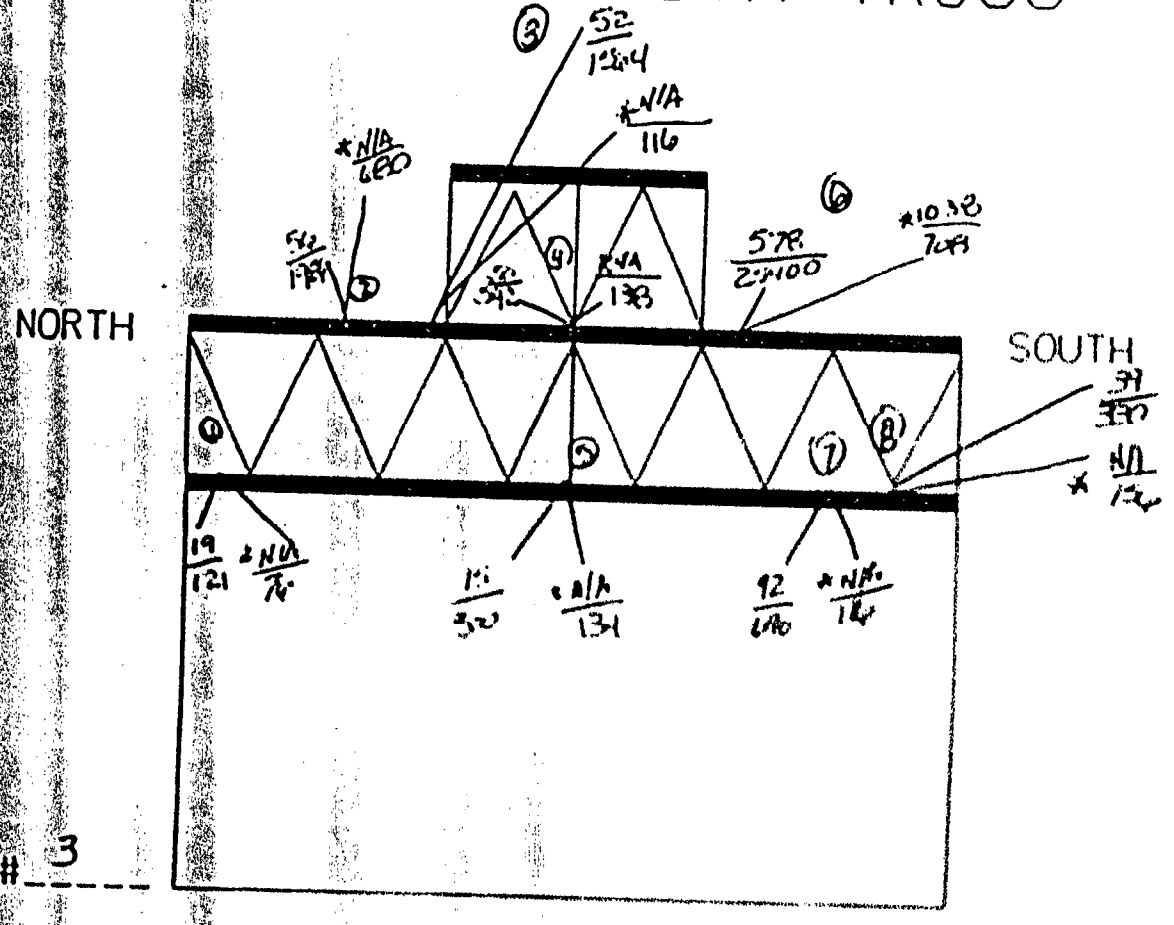
329:1-B
189527
0 0 2 2 7

BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS # 2

BUILDING #23
NORTH AND SOUTH TRUSS



* AFTER DUST REMOVAL. NPA

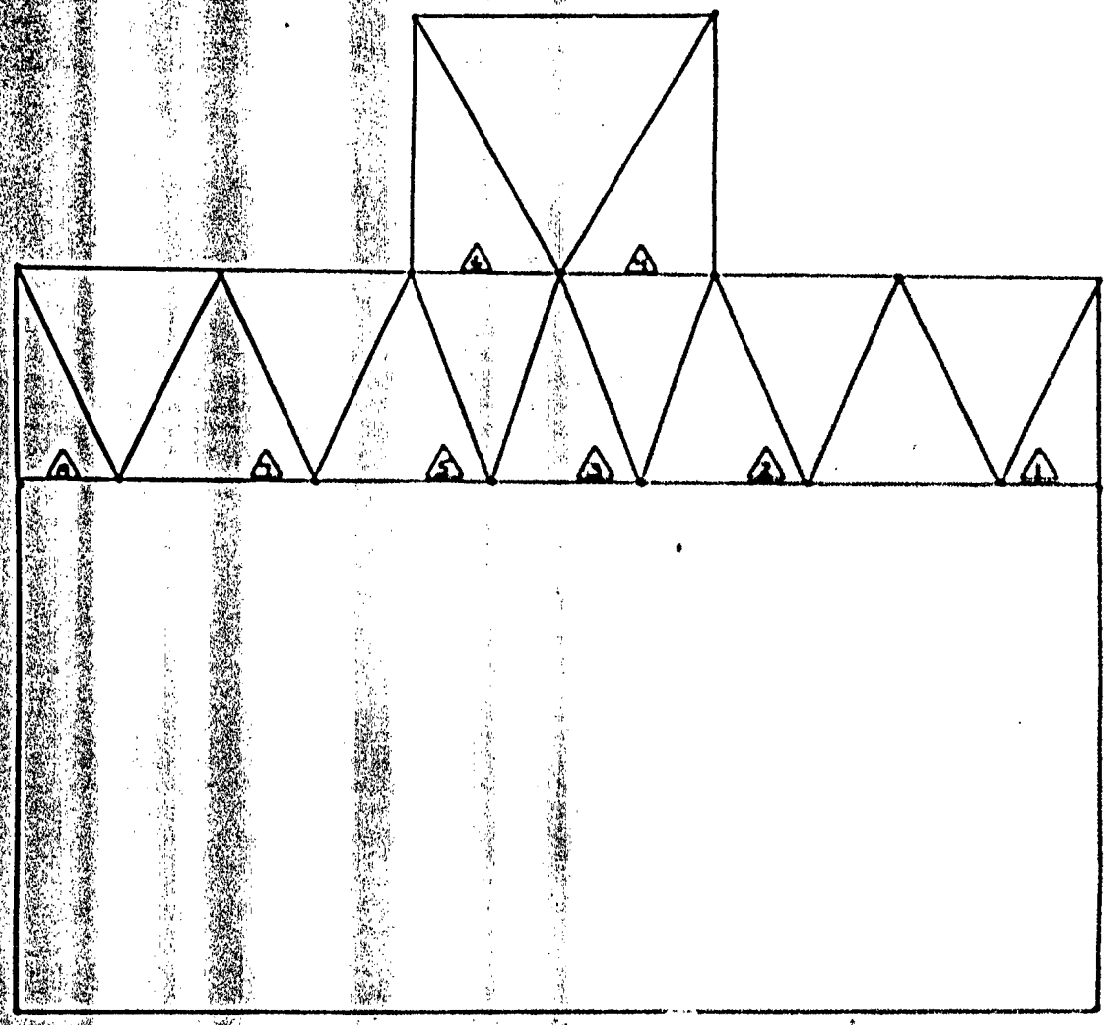
Survey #: 135-DT150

Surveyor: B. Miller

Location: Bay 4 Truss

Site: Cypress Valley

Date: 4-5-98



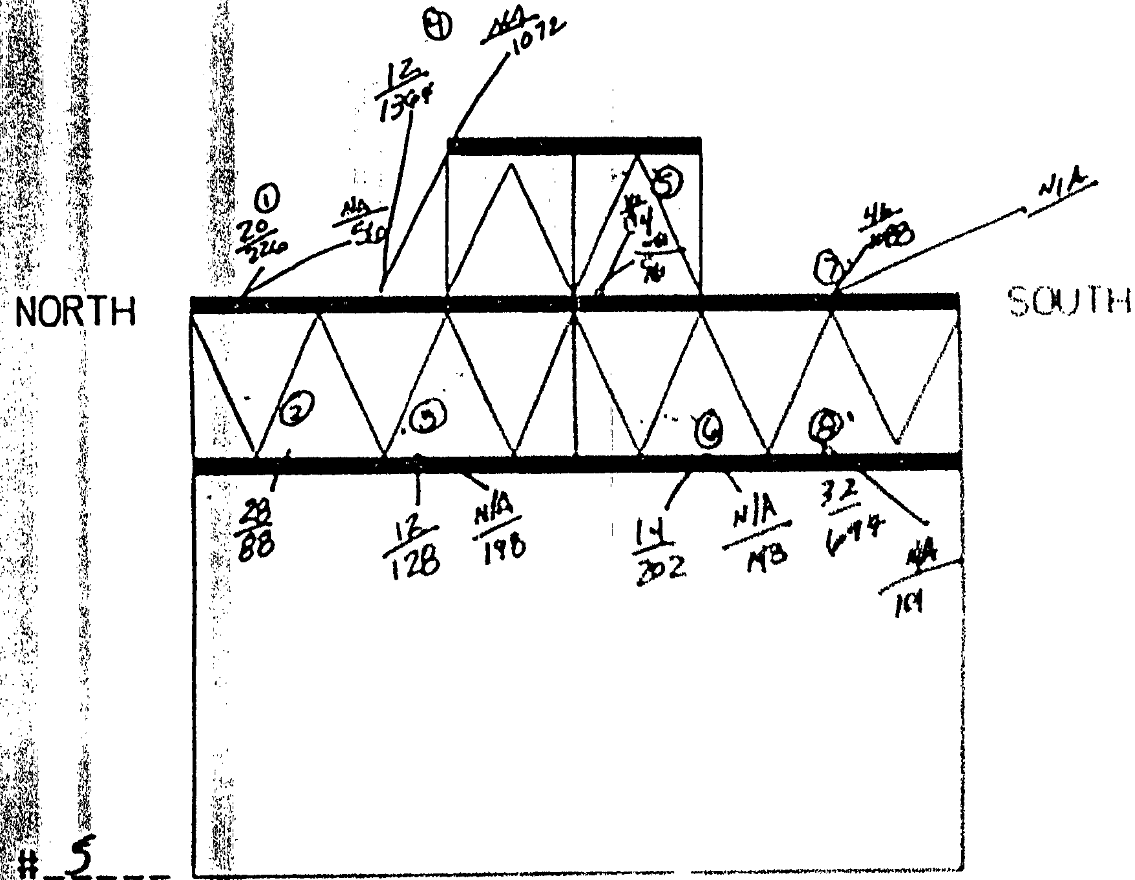
Legend:

△ = Survey Location
: Nymex

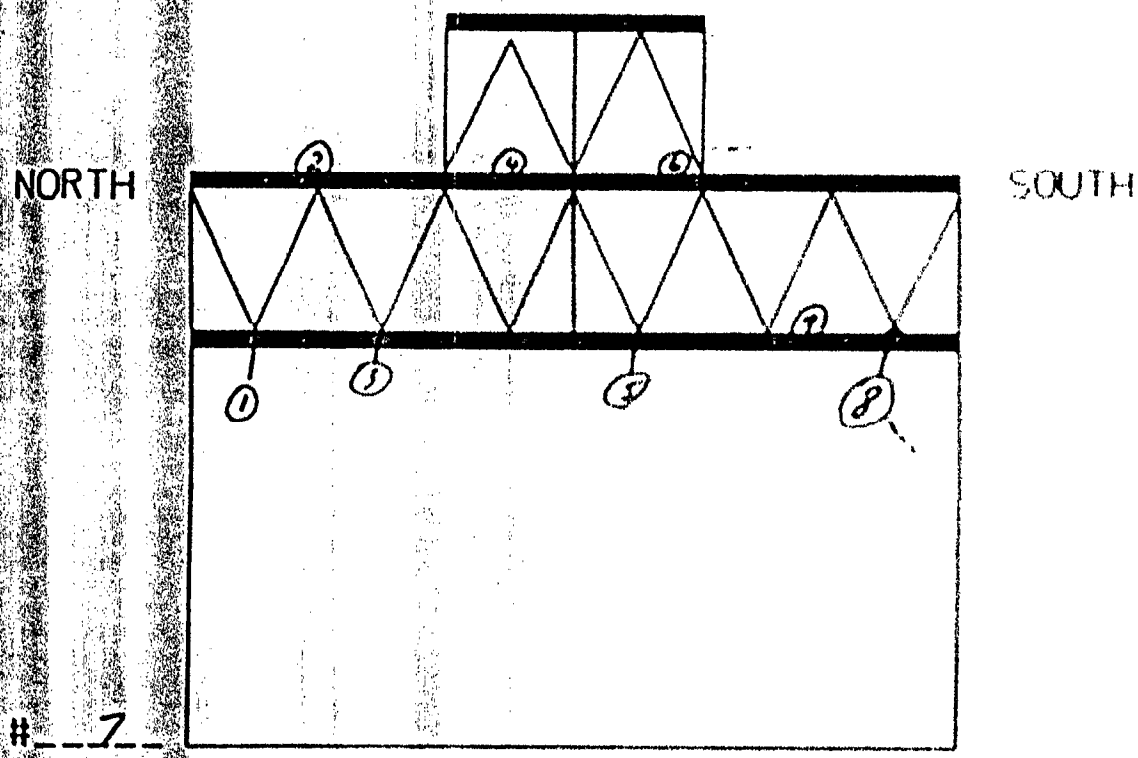
LASER

0022

BUILDING #23
NORTH AND SOUTH TRUSS



BUILDING #23
NORTH AND SOUTH TRUSS

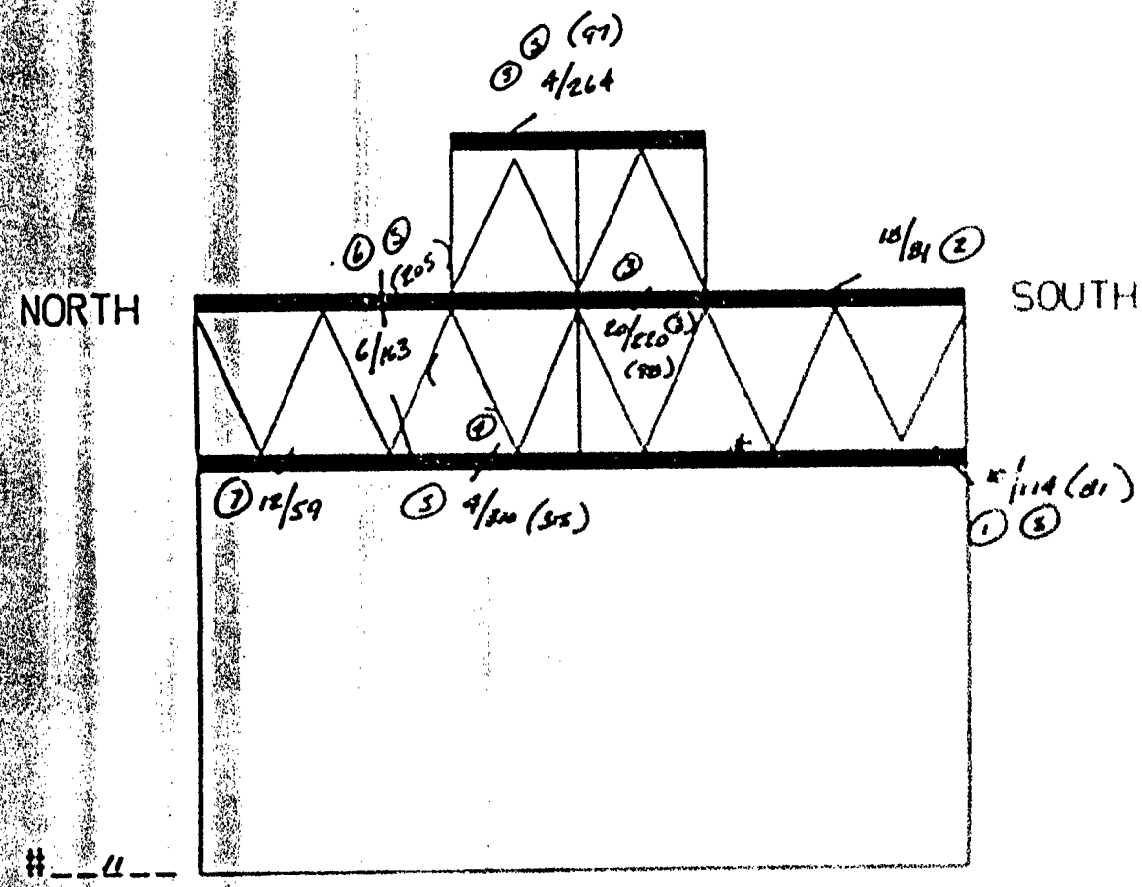


TRUSS #___7___

J-9623

139537

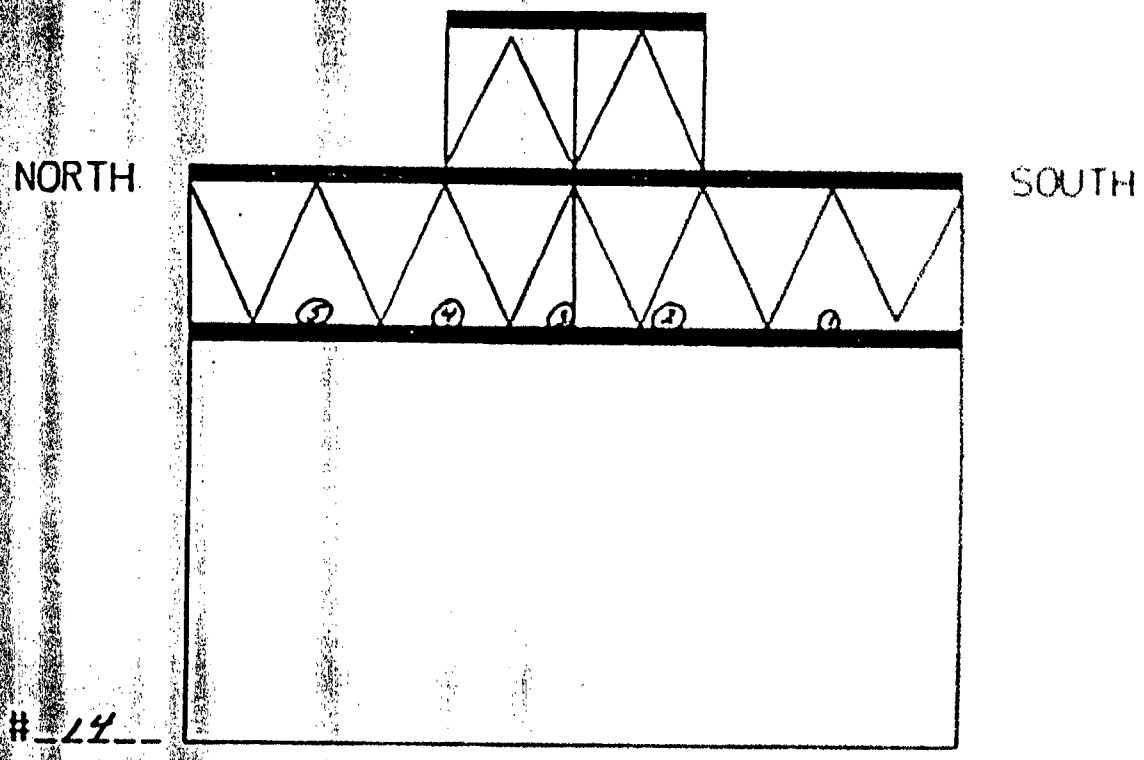
BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS #__4__

139261

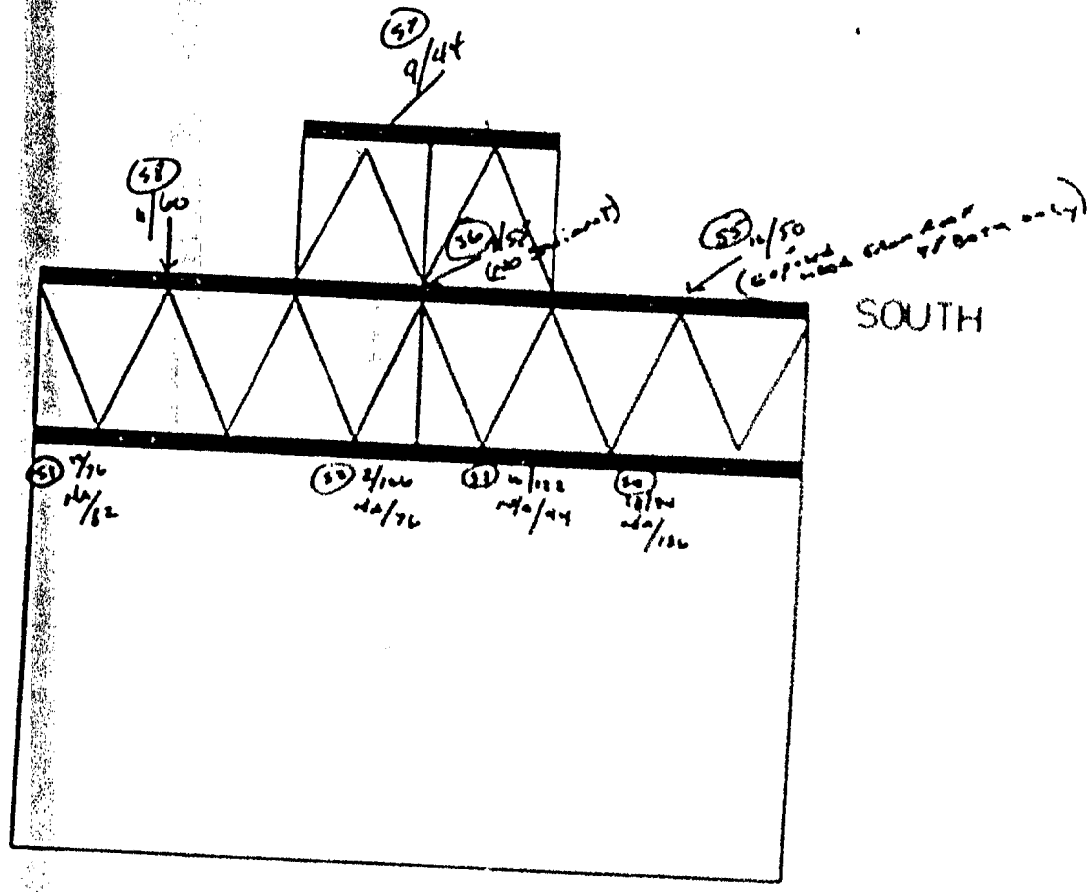
BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS #_44__

J-1-3623
49
189527
0022

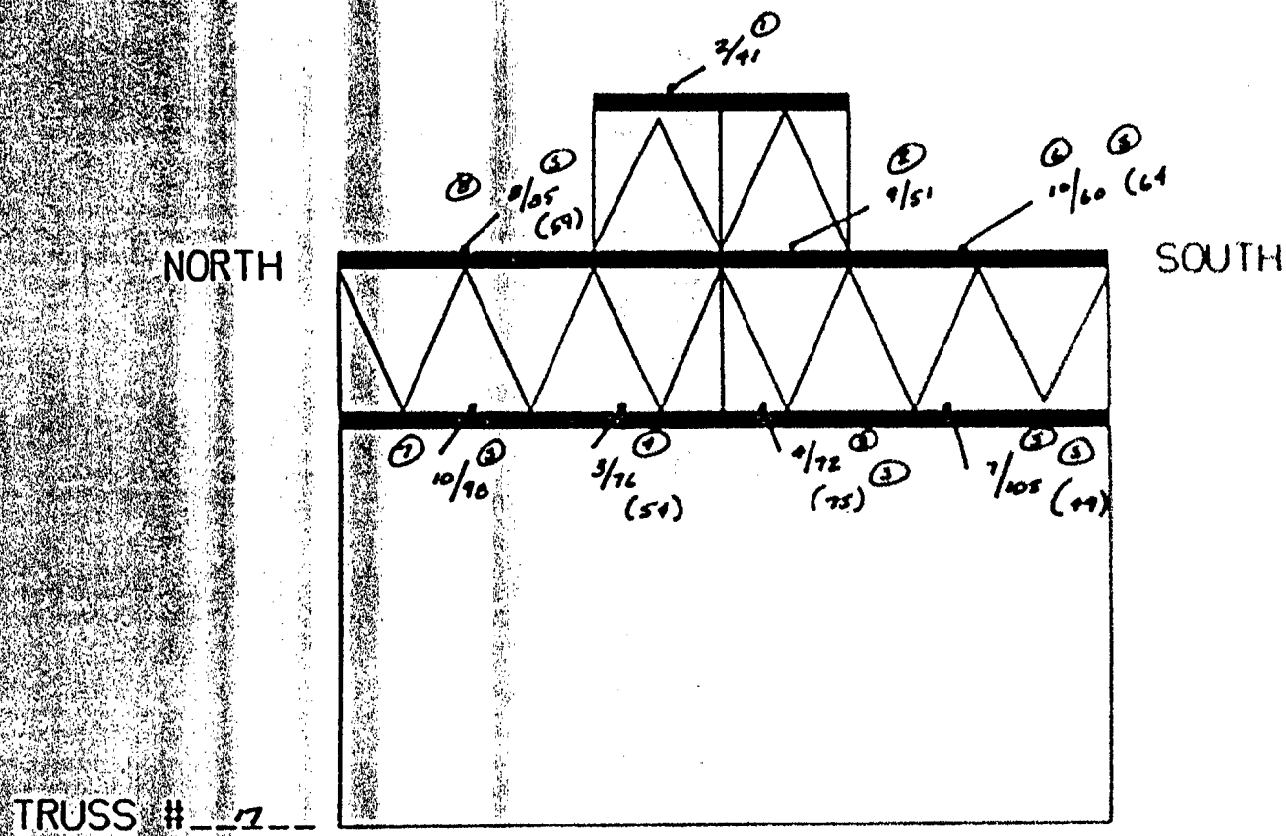
BUILDING #23
NORTH AND SOUTH TRUSS #16



TRUSS #_16_

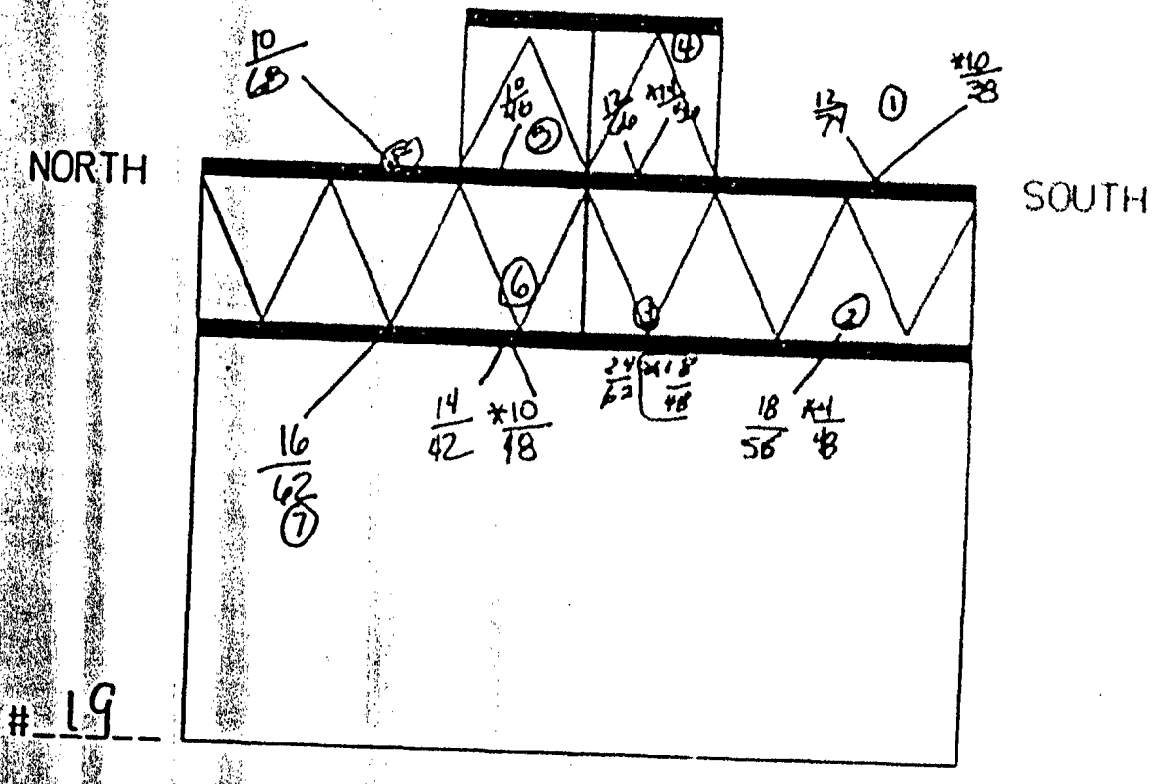
3-15523
50
129527
0022

BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS #__7__

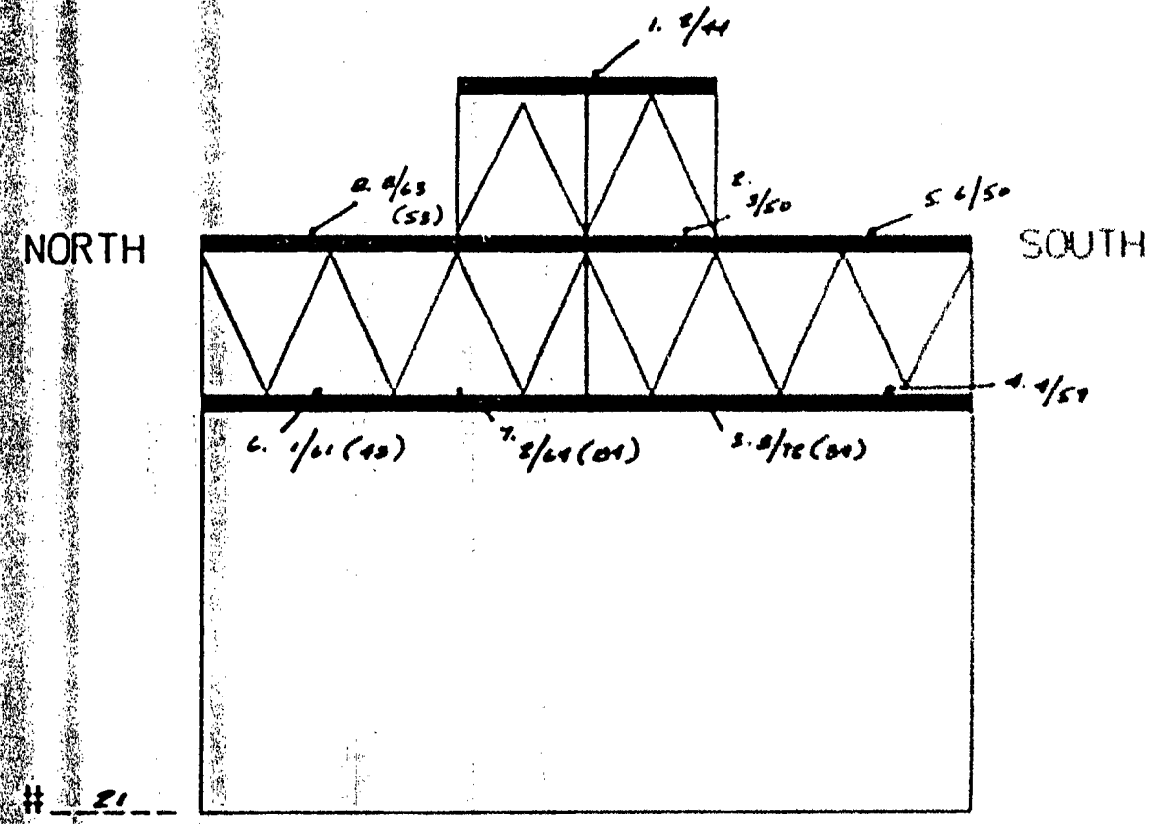
BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS #_19__

* AFTER DUST REMOVAL. MPA

BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS # 21

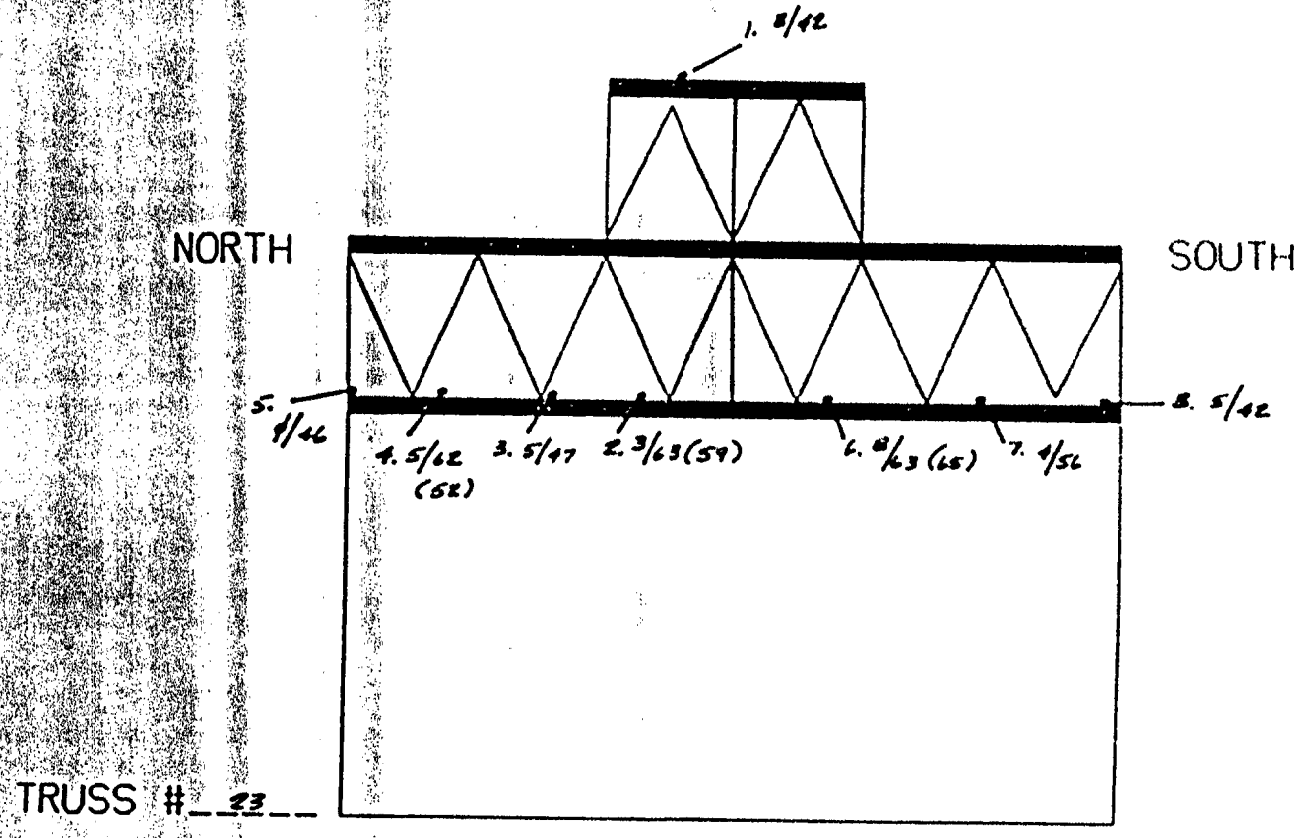
J-13623

53

12957

0022

BUILDING #23
NORTH AND SOUTH TRUSS

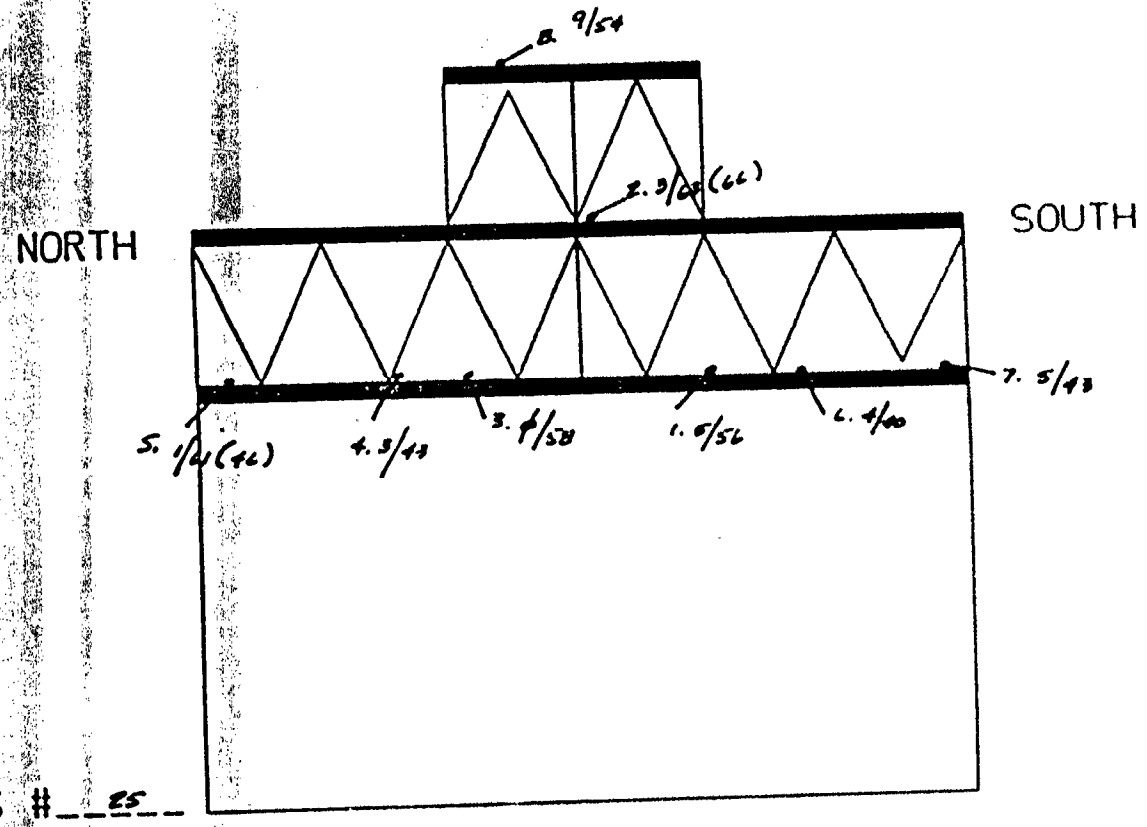


TRUSS #__23__

D-13623

129527

BUILDING #23
NORTH AND SOUTH TRUSS

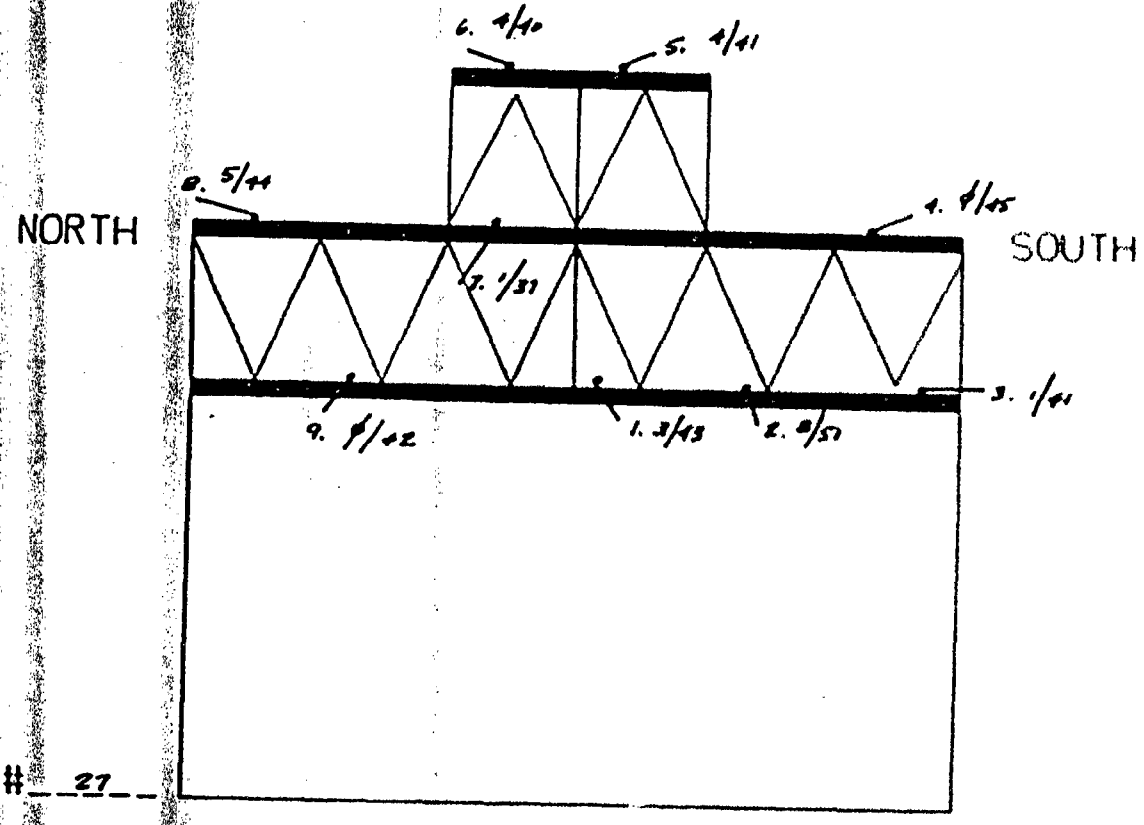


TRUSS #__25__

329511-0

189587

BUILDING #23
NORTH AND SOUTH TRUSS

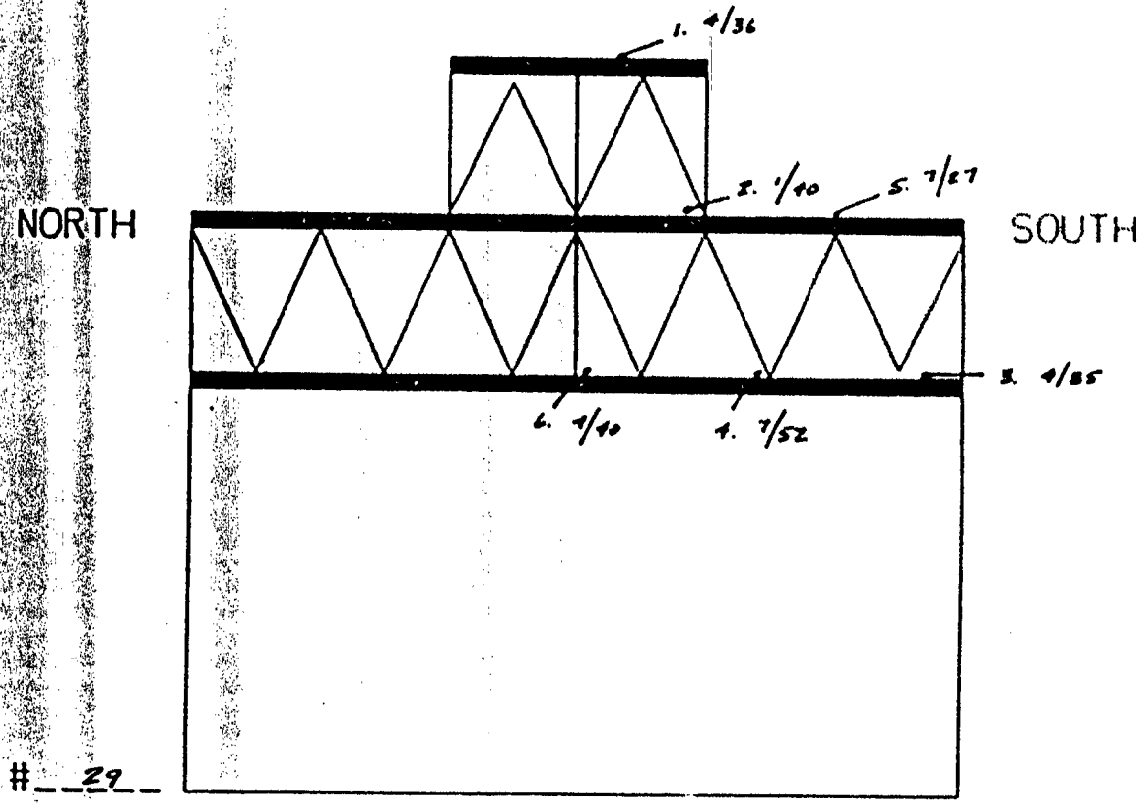


TRUSS # 27

139527

0 0 2 2

BUILDING #23
NORTH AND SOUTH TRUSS



TRUSS #__29__

U-13623

57

12952

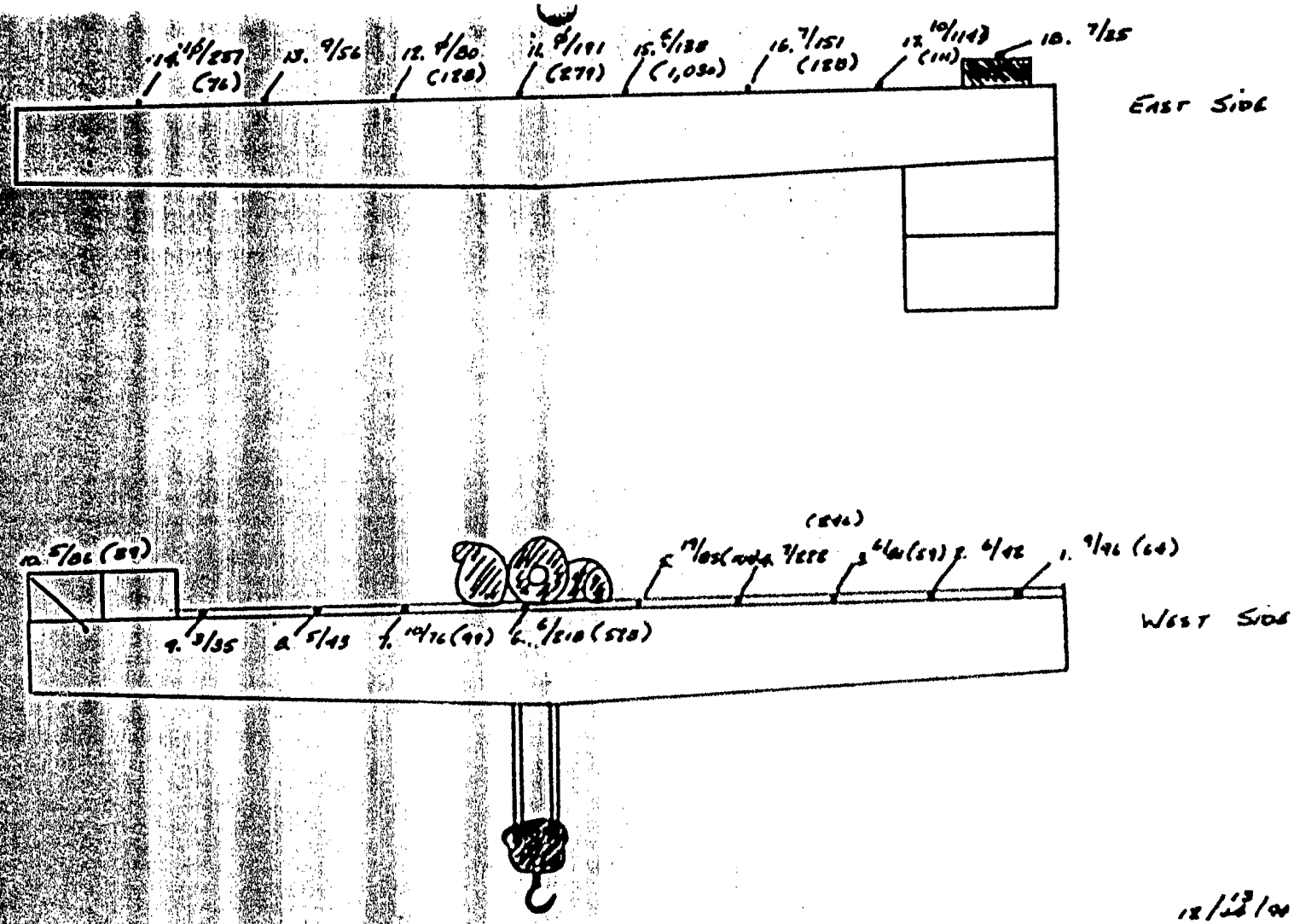
0 0 2 2

129587

Overhead Crane Data

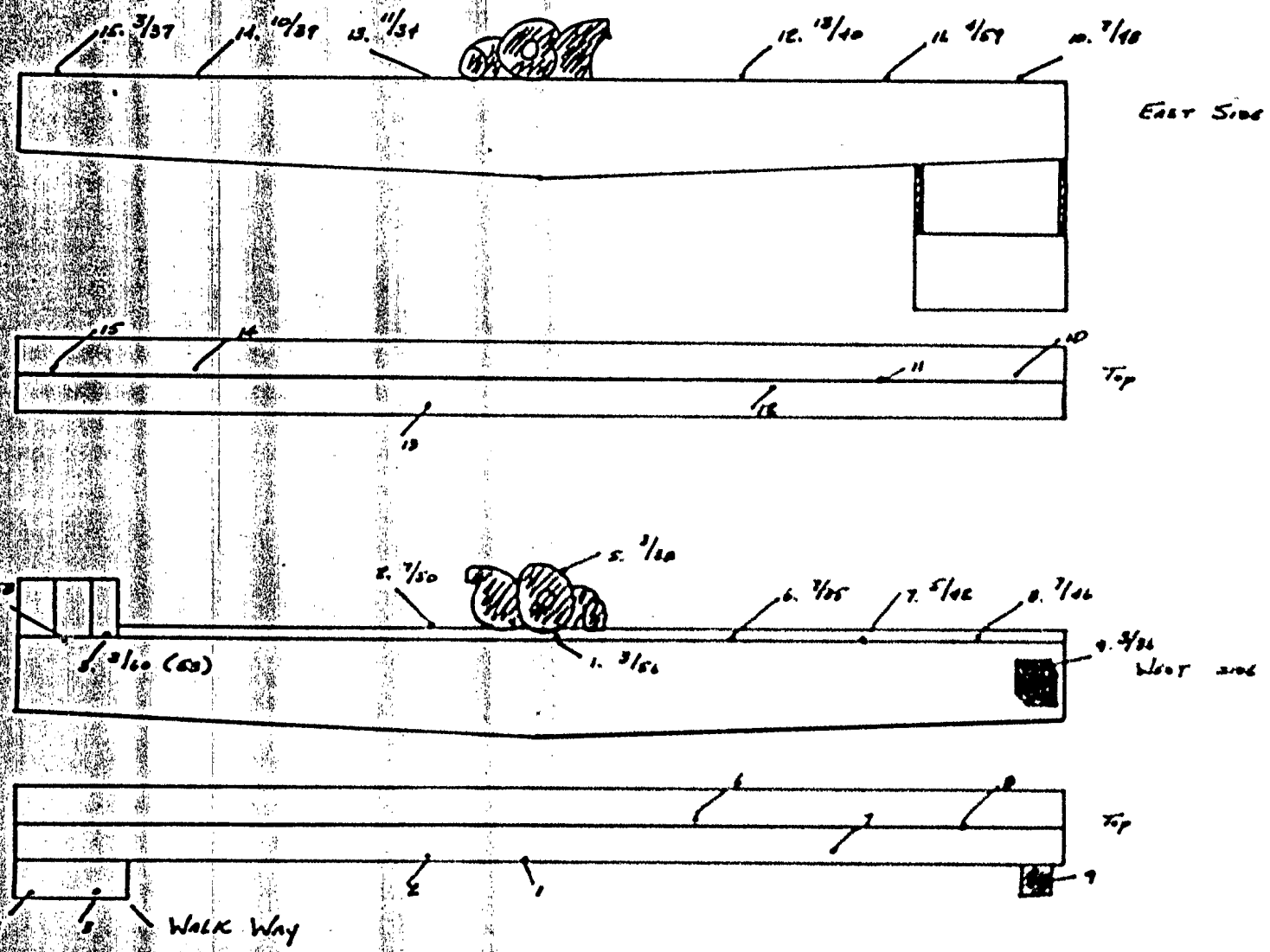
CRANE.XLS

Crane Number	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
C-14	1	0	355				
C-14	2	36	178				
C-14	3	0	474		414	-1	-46
C-14	4	28	414				
C-14	5	0	-178				
C-14	6	36	-296				
C-14	7	19	-89				
C-14	8	36	59				
C-14	9	0	-237				
C-14	10	36	118			-1	6
C-14	11	9	444				
C-14	12	95	-118				
C-14	13	78	-296				
C-14	14	66	-148				
C-14	15	0	-237				
C-15	1	64	1332		365	13	-8
C-15	2	36	-296				
C-15	3	36	888		237	2	8
C-15	4	45	5081		5771	10	-21
C-15	5	159	1006		1716	5	21
C-15	6	36	4942		14117	7	-2
C-15	7	74	740		1421	2	-17
C-15	8	27	-237				
C-15	9	8	-474				
C-15	10	27	1036		-355		
C-15	11	-21	4143		6746	13	-47
C-15	12	-21	858		2279	2	92
C-15	13	64	148				
C-15	14	74	5505		740	7	0
C-15	15	36	2575		28973	-1	-8
C-15	16	45	2959		2279	2	-25
C-15	17	74	1864		1776	10	17
C-15	18	45	-474				


EAST SIDE
WEST SIDE
OVER HEAD CRANE
C-15
SURVEYORS: J. DAMS
J. DEGAR
CHAMPION VALVE

U-19623

60



OVER HEAD CRANE
C-14
SURVEYORS: J. Davis
J. Robinson

12-14-46
CHAPMAN Valve

1.25661

0 0 2 2

129507

West Wall Scan Data

W-WALL.XLS	Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
Location	Before Dust Removal		After Dust Removal			
Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	176	89401	2432	143511	73	259
2	-24	56697	577	51940	5	25
3	483	178513	3219	134582	141	510
4	710	66805	1284	111402	250	439
5	1564	43913	1351	20217	21	50
6	1658	12100	1311	9157	10	25
7	1044	2230	270	565	2	-29
8	670	7819	697	4192	5	13
9	1818	184540	1858	65319	84	305
10	29	13678	230	3678	-1	-13
11	857	112886	1391	60265	40	54
12	109	20009	3960	7641	2	-17
13	1257	10673	1631	18493	-1	38
14	243	5500	203	3300	-1	29
15	203	36777	777	32318	46	138
16	643	2884	296	862	10	8
17	1284	182539	1684	151241	114	218
18	136	9306	109	14360	43	121
19	83	3479	163	1160	5	17
20	29	35469	29	25301	43	176
21	136	8533	56	14182	13	21
22	3	24052	403	191675	136	552
23	109	2111	136	1754	7	59
24	-24	44269	3	5619	37	134
25	3	6571	-51	2587	43	46
26	283	3954	56	684	5	-4
27	3	3033	43	1695	2	-4
28	69	2624	-51	1992	7	-29
29	-64	12338	149	2706	13	29
30	163	2289	96	1457	5	4
31	190	2051	163	803	7	-4
32	323	8354	3553	20425	198	372
33	83	2051	83	2111	5	29
34	136	17452	1564	13171	43	50
35	243	9484	403	8414	16	29
36	216	10436	109	11625	7	4
37	363	1992	176	1873	10	0
38	350	26074	1044	40702	65	134
39	109	14538	109	11030	7	59
40	617	28155	590	16203	29	96
41	270	26550	243	16857	5	17
42	-24	14301	3	16857	2	42
43	56	8890	176	3835	24	33
44	93	12576	67	9641	0	-29
45	107	30415	1229	27382	6	13
46	120	19831	147	19295	0	-29

129507

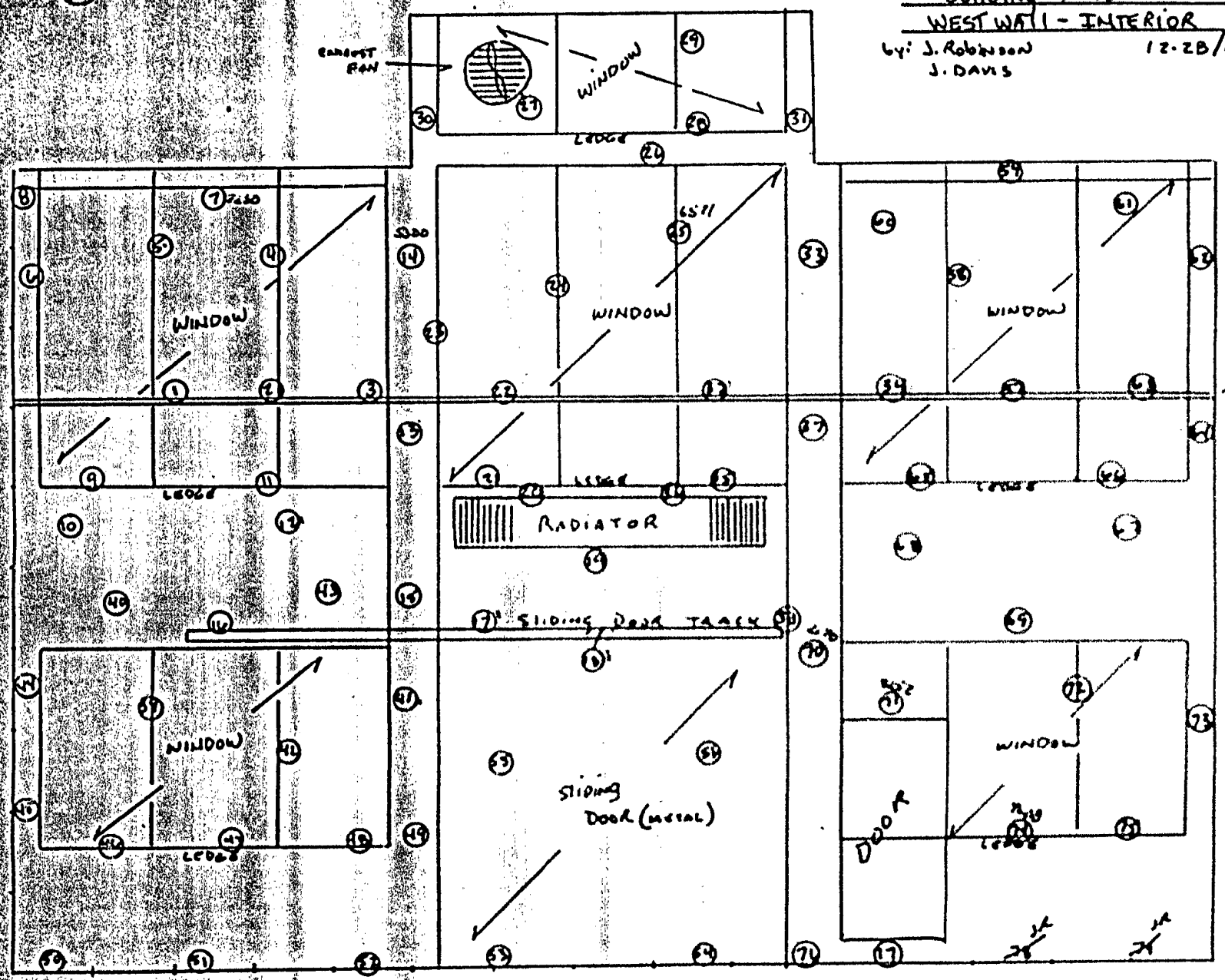
West Wall Scan Data

W-WALL.XLS	Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
Location	Before Dust Removal		After Dust Removal			
Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
47	360	24647	467	23041	13	8
48	334	33568	227	24706	6	17
49	13	10436	27	6630	3	13
50	254	10079	120	7165	6	-21
51	87	20068	294	20366	0	-21
52	67	11327	67	7284	3	-17
53	-13	-446				
54	-40	-387				
55	360	-535				
56	187	-268				
57	147	62643			20	79
58	40	6154			10	-42
59	120	268			3	-29
60	240	14033			60	138
61	200	6392			6	0
62	93	2765			10	0
63	360	40939			10	-21
64	67	2319			0	-71
65	147	14776			6	-21
66	173	1338			6	-63
67	-40	1160			3	-8
68	-13	1278			0	-8
69	27	2884			6	-17
70	13	2170			3	-21
71	67	4252			0	-4
72	-93	773			6	13
73	-13	89				
74	67	4608			10	-13
75	133	7849			3	-46
76	53	1189			6	-17
77	93	1487			6	-13

CHAPMAN VALVE
Building # 23
WEST WALL - INTERIOR

by: J. Robinson
J. Davis

12-28/29-54



D-19693

19950

129507

South Wall Scan Data

S-WALL.XLS

Bay	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
1	1	87	1011			5	-43
1	2	260	0				
1	3	418	22923	1029	6600	13	33
1	4	313	16798	799	10814	5	-54
1	5	193	22714	3517	8879	2	4
1	8	260	79400			174	372
1	9	101	6333	21	6719	0	-13
1	10	-32	1457				
2	11	-32	1100				
2	12	-138	2795	111	2438	0	-29
2	13	21	773				
2	14	458	6541	418	2364	5	17
2	15	763	15044	975	4816	0	8
2	16	2219	127129	6614	57914	133	205
2	17	445	13201	3034	5054	16	0
3	18	641	11387	1529	6400	36	-33
3	19	853	5738	1250	10584	35	38
3	20	191	135			5	-25
3	21	111	2943			10	-37
3	48	19	1041			2	-13
4	22	244	387				
4	23	111	5600	95	5233	2	-42
4	24	244	3151			2	-29
4	25	58	1933			5	-25
4	26	164	446				
4	49	125	1249			-1	-25
4	50	-8	743				
5	27	297	178				
5	28	217	1457				
5	29	297	6184	826	6771	27	33
5	30	164	1011			2	4
5	31	323	714				
5	51	19	297				
5	52	-8	387				
6	32	297	1665			5	-54
6	33	506	-178				
6	34	826	2406			0	8
6	35	164	8085	350	4035	8	-4
6	36	164	2846			8	-50
6	37	588	-59				
6	38	297	1754				
6	53	-8	416				
7	39	1263	1070	1503	832	8	-38
7	40	1343	6511	1423	1992	8	13
7	41	61	2111			-1	-25
7	42	1129	2557	1316	832	10	-25
7	54	152	1546	72	1754	2	-4
7	55	-35	446				

129507

South Wall Scan Data

S-WALL.XLS

		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Bay	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
8	43	1449	1278			2	0
8	44	1263	268	1316	119	-1	8
8	45	1129	4579	35	-238	13	59
8	46	1182	491	302	595	-1	-21
8	47	889	1160	1209	1695	8	-17
8	56	72	327				
8	57	19	387				
9	49	-29	603				
9	50	-83	505				
9	51	-56	832				
9	52	451	297				
9	53	344	-30				
9	54	-56	1338	291	1487	-1	54
9	55	211	89				
9	56	264	387				
9	57	558	327				
18	1	53	503				
18	2	82	740				
18	3	272	1036	316	947	2	-59
18	4	38	1450	9	1243	2	-17
18	5	67	651				
18	6	286	474				
18	66	-6	0				
18	67	9	-148				
19	7	23	1480	23	2160	-1	0
19	8	82	533				
19	9	301	533				
19	68	82	-207				
19	69	53	266				
20	10	111	1184	38	1095	2	-25
20	11	316	474				
20	12	345	207				
20	13	38	1539	82	1509	2	13
20	14	169	710				
20	15	111	-30				
20	70	53	207				
20	71	53	-30				
21	16	53	89				
21	17	96	296				
21	18	228	266				
21	19	-6	59				
21	20	111	326				
21	21	272	237				
21	72	23	-266				
21	73	67	355				

129587

South Wall Scan Data

S-WALL.XLS

Bay	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
22	22	96	207				
22	23	126	-30				
22	24	213	296				
22	74	38	178				
22	75	23	-533				
23	25	82	-30				
23	26	184	444				
23	27	345	621				
23	28	9	-148				
23	29	9	681				
23	30	140	59				
23	76	38	237				
24	31	9	148				
24	32	155	592				
24	33	67	266				
24	34	-6	414				
24	35	67	414				
24	36	140	296				
25	37	38	-148				
25	38	11	-326				
25	39	228	59				
25	40	67	562				
25	41	-6	355				
25	42	9	89				
25	77	82	-118				
26	43	-6	-296				
26	44	23	710				
26	45	23	-118				
26	46	-6	474				
26	47	243	207				
26	48	140	-474				
26	78	53	355				
26	79	111	59				
27	49	-6	0				
27	50	126	207				
27	51	169	-326				
27	52	-6	-148				
27	53	96	0				
27	54	213	266				
27	80	67	-118				
28	55	9	-266				
28	56	82	-30				
28	57	-20	-30				
28	58	-6	59				
28	59	96	266				
28	60	243	0				

129507

South Wall Scan Data

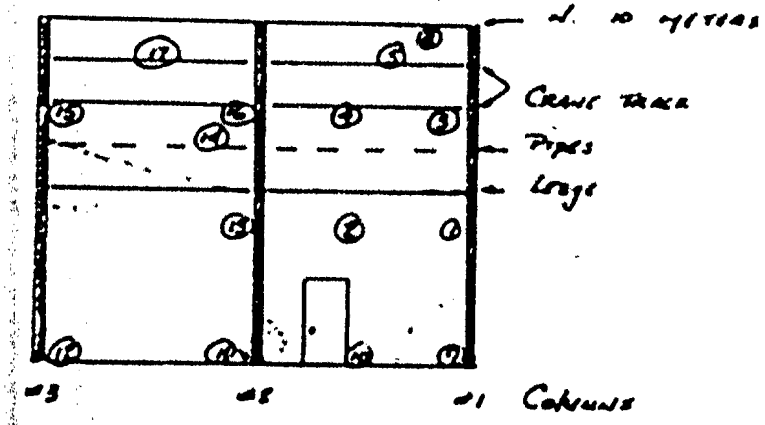
S-WALL.XLS

Bay	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Direct (DPM/100 cm^2) Before Dust Removal Beta/Gamma	Direct (DPM/100 cm^2) After Dust Removal Alpha	Direct (DPM/100 cm^2) After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable (DPM/100 cm^2) Beta/Gamma
28	81	67	89				
28	82	53	-296				
29	61	-6	-118				
29	62	36	237				
29	63	9	1065	36	829	-1	29
29	64	36	444				
29	65	257	89				
29	83	67	355				

0023

SOUTH WALL Bldg #23
INSIDE

30/94



D-19968

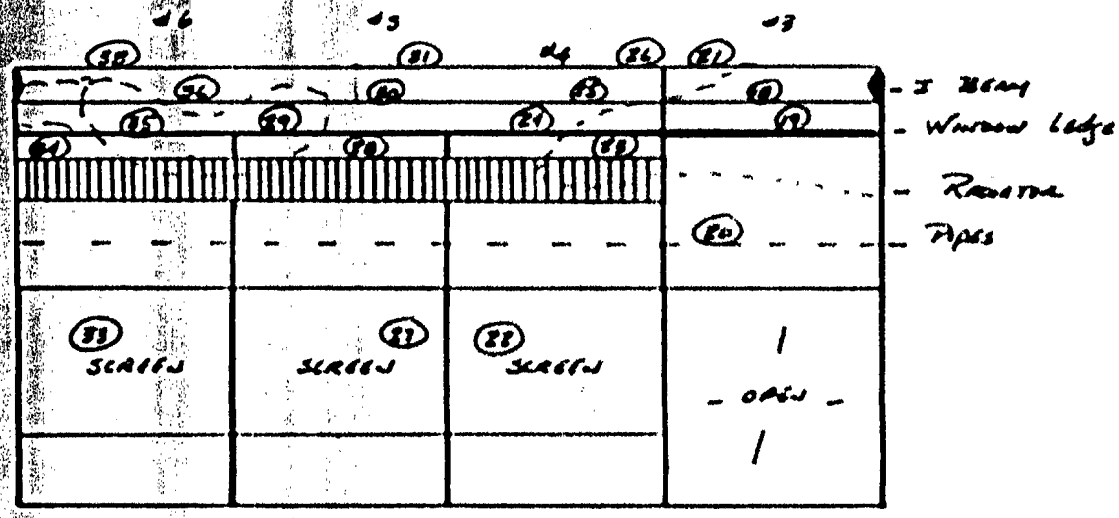
19958

0023

12/4

CHAPMAN VALVE

SOUTH WALL BLDG 23
INTERIOR
B'γ & α SCAN



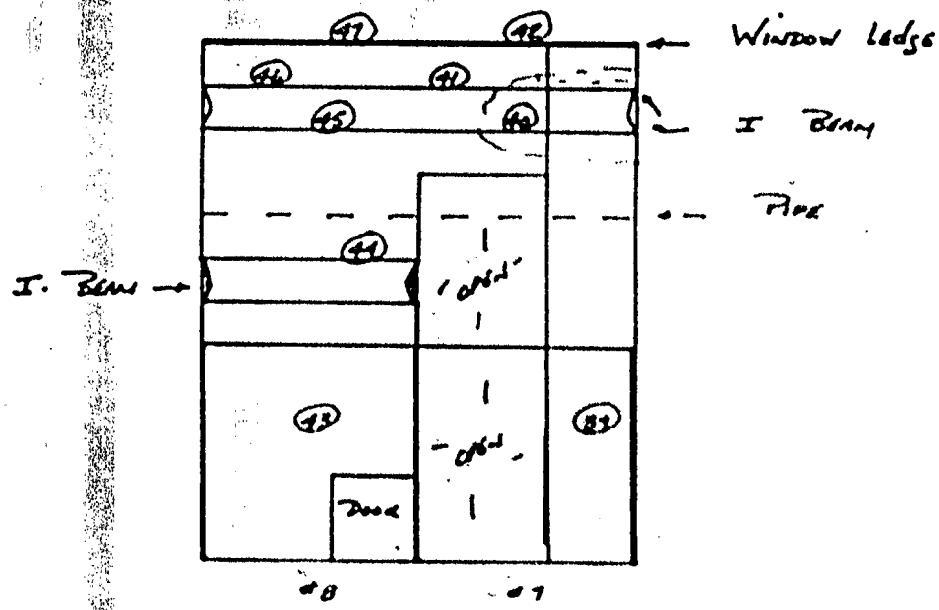
D-2256-D

69

0 0 2 3

SOUTH WALL BLDG #23
(INTERIOR)

5/94
CHAMPION VALVE



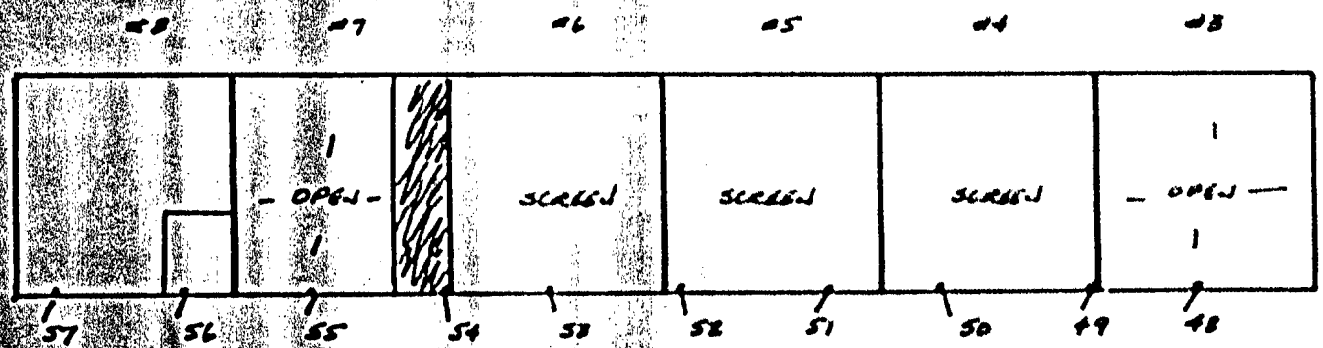
D-196223

70

12950

SOUTH WALL
(INTERIOR)
B'γ's & SCAN GRID INTERCEPTS

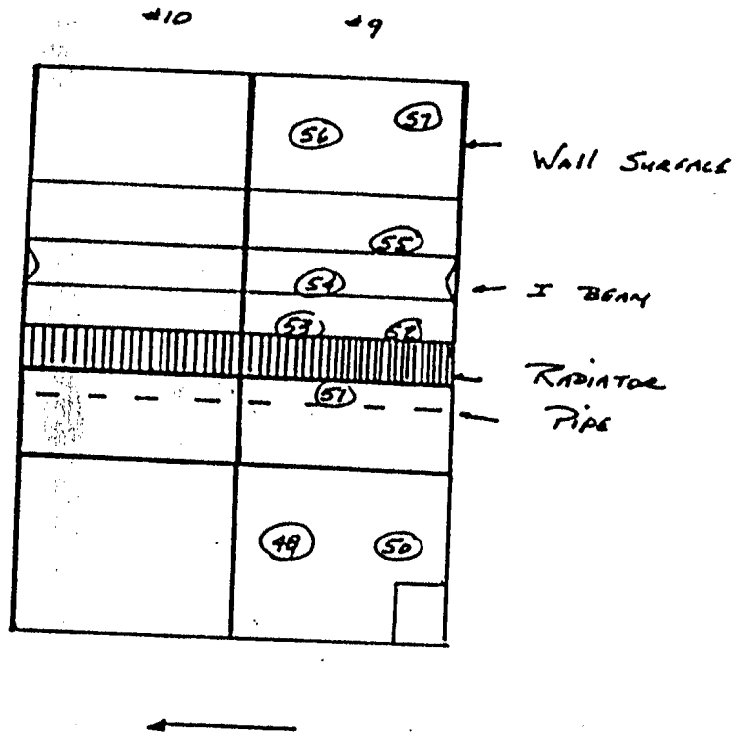
12/7/94
CHAPMAN VALVE



139527

South Wall
Bldg #23 (Interim)

12/6/94
Chapman Value

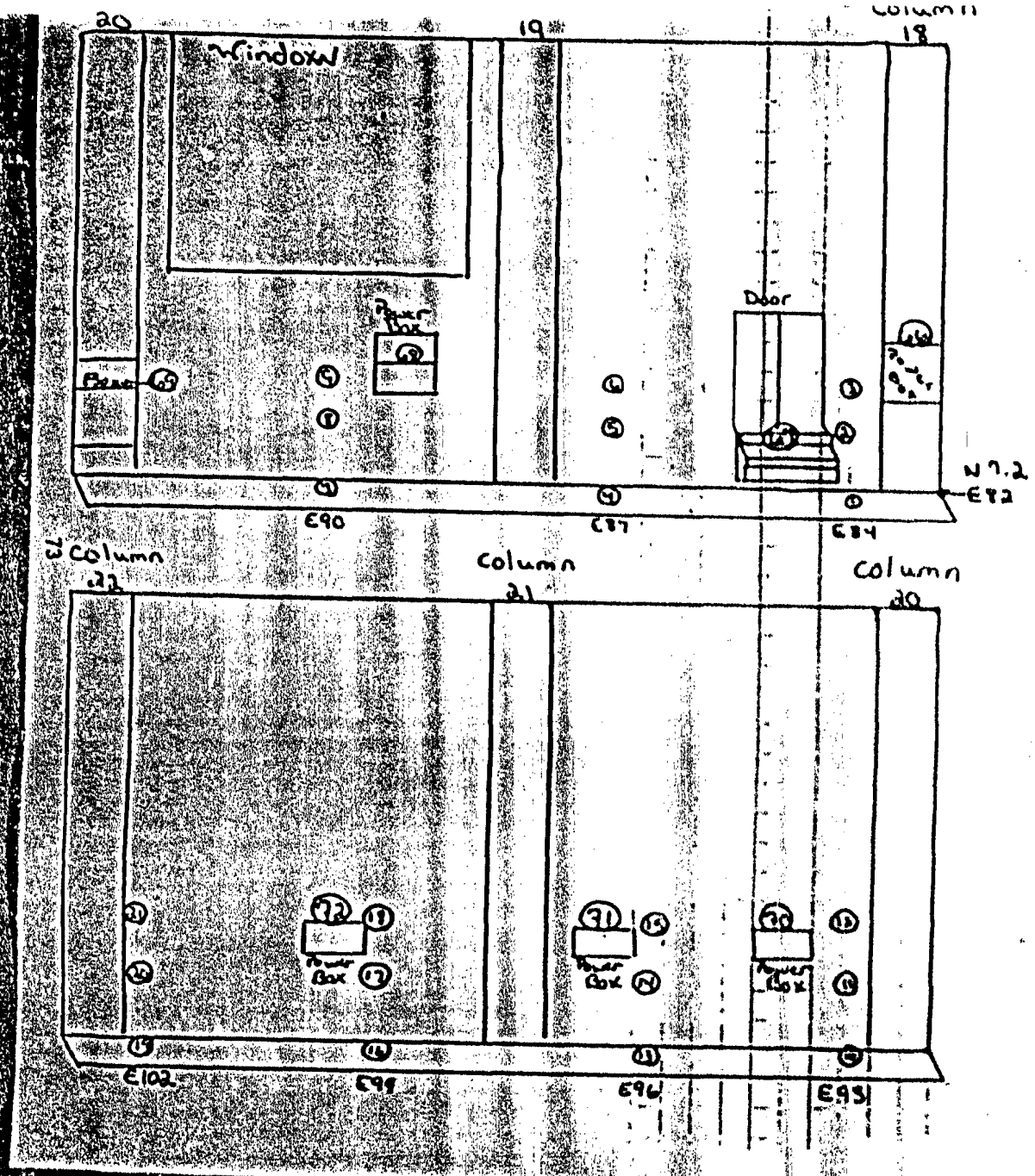


U-15623

72

189377

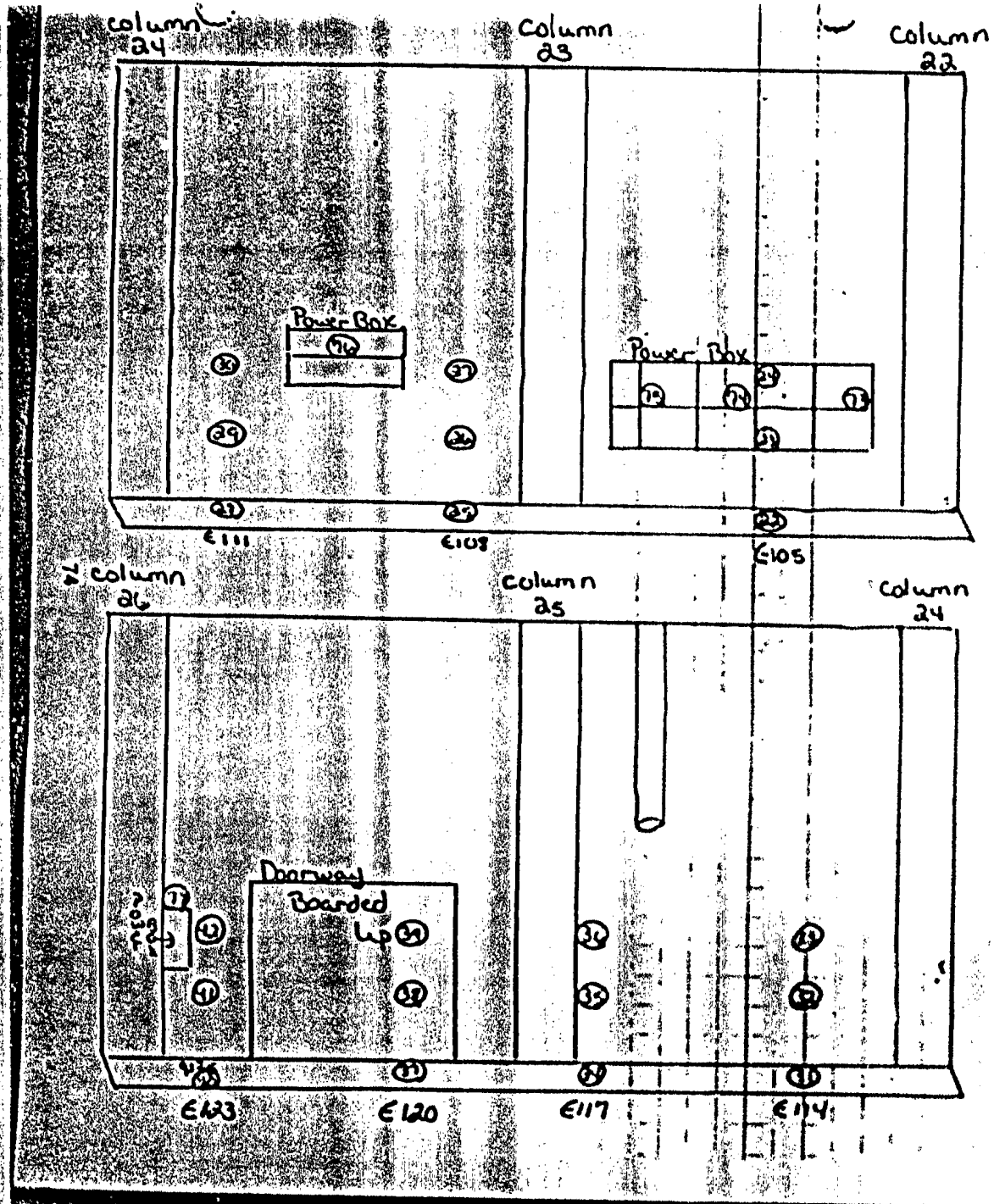
0023



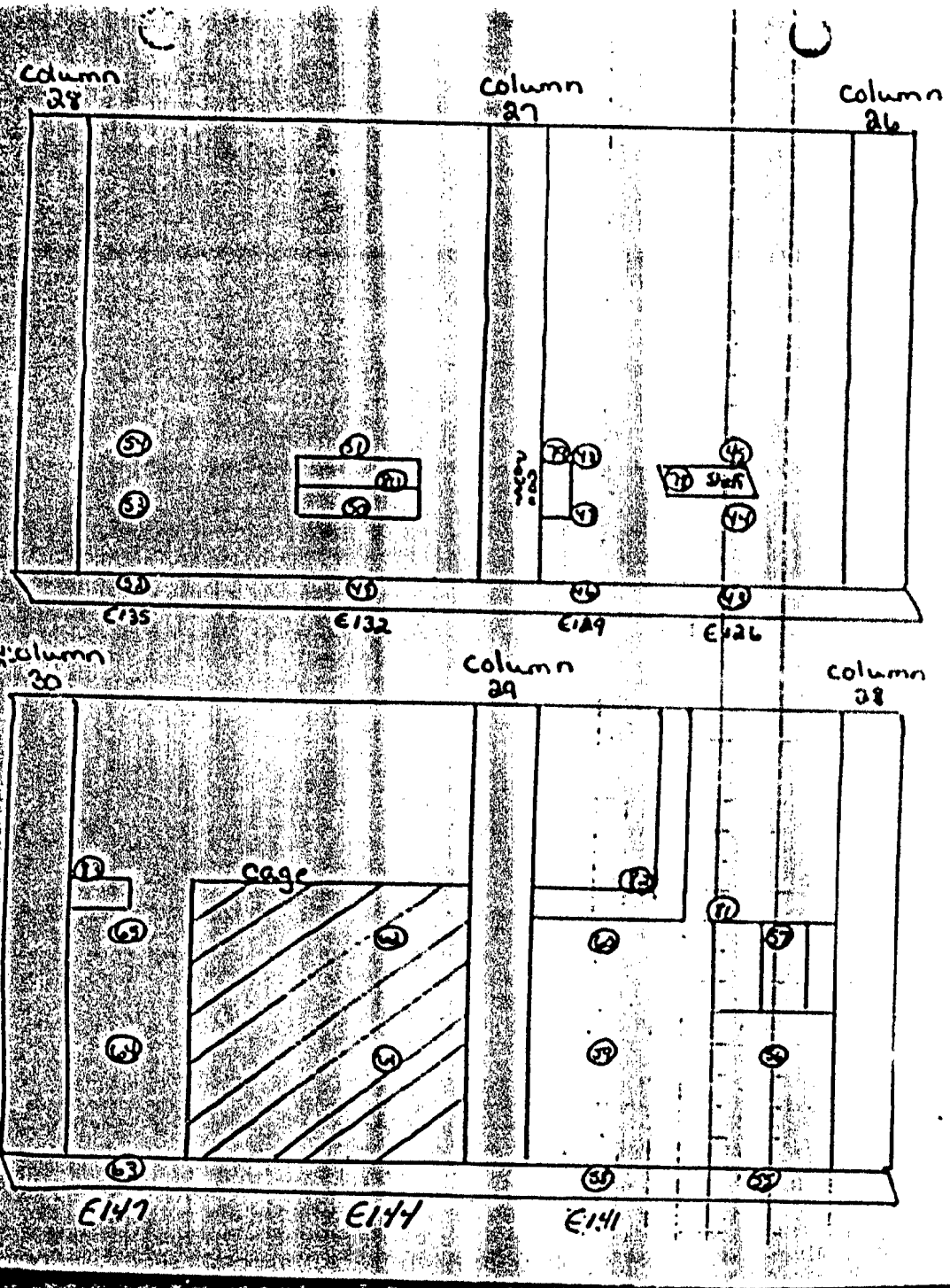
Bldg 23
B-Side Wall

Readings taken on floor, 1 meter and 2 meter height from floor, and horizontal surfaces.

0-19693


column 24
Column 23
Column 22
Power Box
Power Box
E111
E108
E105
column 26
Column 25
Column 24
Doorway Boarded Up
E123
E120
E117
E114

D-19693



On E144 no floor Reading was taken due to raised platform and water on it's surface.

D-19693

129507

North Wall Scan Data

N-WALL.XLS

Bay	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
1	1	80	1189			-1	-8
1	2	294	22506			32	46
1	3	307	5262			5	-8
1	4	-13	6154			7	4
1	5	27	2081			7	-4
1	6	254	367				
1	7	120	3776			5	4
1	8	467	15371			10	-13
1	9	334	1516			-1	29
1	10	120	6630			5	13
1	11	-40	2406			2	46
1	12	160	7462			5	-17
1	13	-107	1219			-1	17
1	14	-40	1516			7	8
1	15	-67	1516			2	-4
1	16	-80	1189			-1	21
1	17	40	1427			-1	25
1	18	93	1665			2	0
1	19	40	1487			5	4
1	20	-27	922			-1	17
1	21	-40	1576			-1	17
1	22	-27	1397			-1	8
1	23	13	1516			-1	-38
2	1	-45	59				
2	2	969	446	8	922	5	4
2	3	168	2616	1529	2200	10	0
2	4	1316	9611	1289	6392	13	4
2	5	248	9068	3398	7522	19	63
2	6	115	2438	2117	11922	-1	4
2	45	99	2943	19	2081	3	33
2	46	99	1843	45	3300	6	-4
2	47	19	8860	-8	773	6	21
3	7	916	0				
3	8	1156	624	1102	416	-1	-4
3	9	1396	3865	1690	4222	2	17
3	10	1449	8027	1556	4787	40	117
3	11	869	624				
3	48	-35	922				
3	49	19	2260	72	1219	3	17
4	12	622	-59				
4	13	355	-89				
4	14	1289	2349	1209	1933	2	-46
4	15	302	-30				
4	16	1423	7462	1049	5143	8	71
4	50	152	1397			9	-8
4	51	45	1219			6	50

0023

12 950

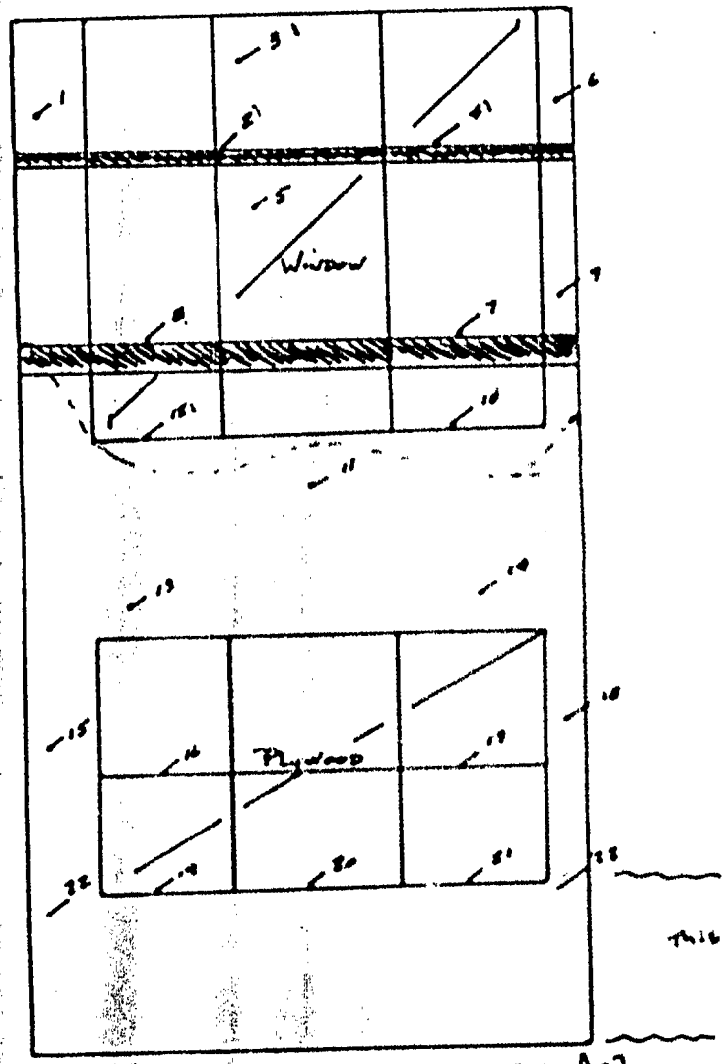
North Wall Scan Data

N-WALL.XLS

Bay	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferable (DPM/100 cm^2) Alpha	Transferable Beta/Gamma
5	17	622	416				
5	18	782	-59				
5	19	996	1635	115	1070	10	4
5	20	-19	1724	-19	1100	5	8
5	21	1022	14609	1903	9187	78	180
5	22	89	-208				
5	52	45	4043	179	1514	33	71
5	53	45	2872	72	2081	-1	-12
5	54	-8	2379	45	1911	3	-4
6	23	916	59				
6	24	1823	1249	-45	25628	8	4
6	25	-45	2854	-72	1784	13	38
6	26	68	-89				
6	27	195	2587	275	1605	2	13
6	55	232	505				
6	56	99	922				
7	28	-58	-59				
7	29	344	1249				
7	30	371	2349				
7	31	504	1457				
7	32	211	5530	771	3003	16	17
7	33	-29	-30				
7	57	72	1578	72	2436	6	28
7	58	-35	565				
8	34	-29	1962	478	1070	2	-4
8	35	24	178				
8	59	99	2587	72	4133	13	-42
8	60	72	-297				
9	36	24	535				
9	37	-89	119				
9	38	104	327				
9	39	51	-208				
9	40	-29	-238				
10	41	157	2022	238	2587		
10	42	284	297				
10	43	157	30				
10	44	-3	-149				
10	61	45	30				
10	62	-35	-208				
10	63	45	-238				

CHAPMAN Bldg NORTH
INTERIOR ~~West~~ WALL (BAY-1)
Bldg # 23

12-29-..
J. DAVIS / J. ROBINSON



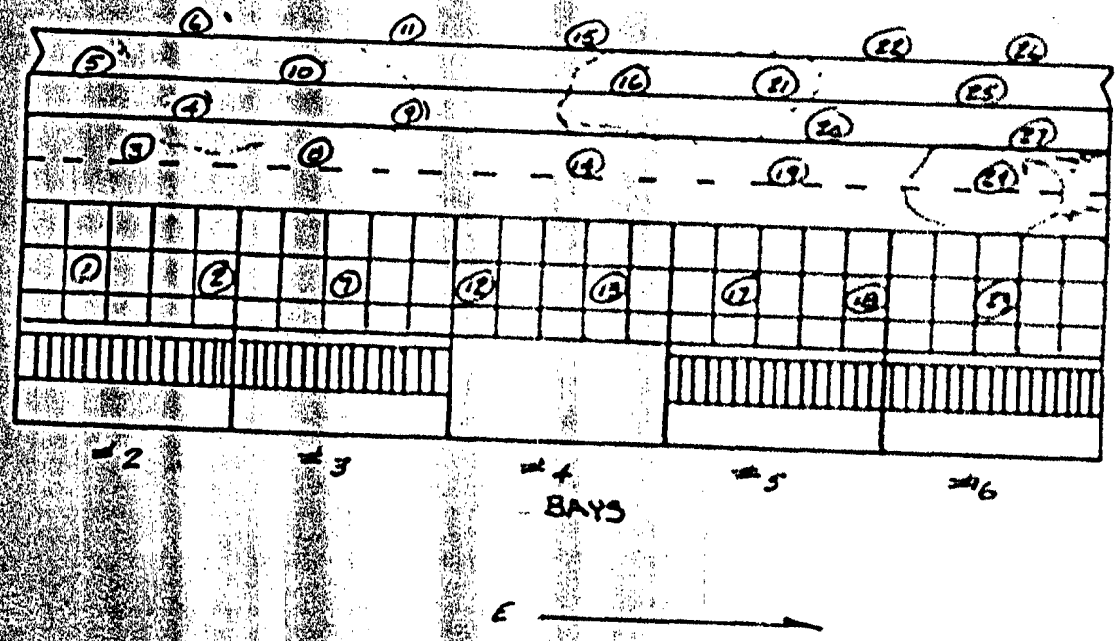
THIS AREA IS COVERED w/ PLASTIC "ACCESS CONTROL"

0-19693

19327

NORTH WALL
INTERIOR SCAN
B'γ & α

-5-94
CHAPMAN VALVE



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1895

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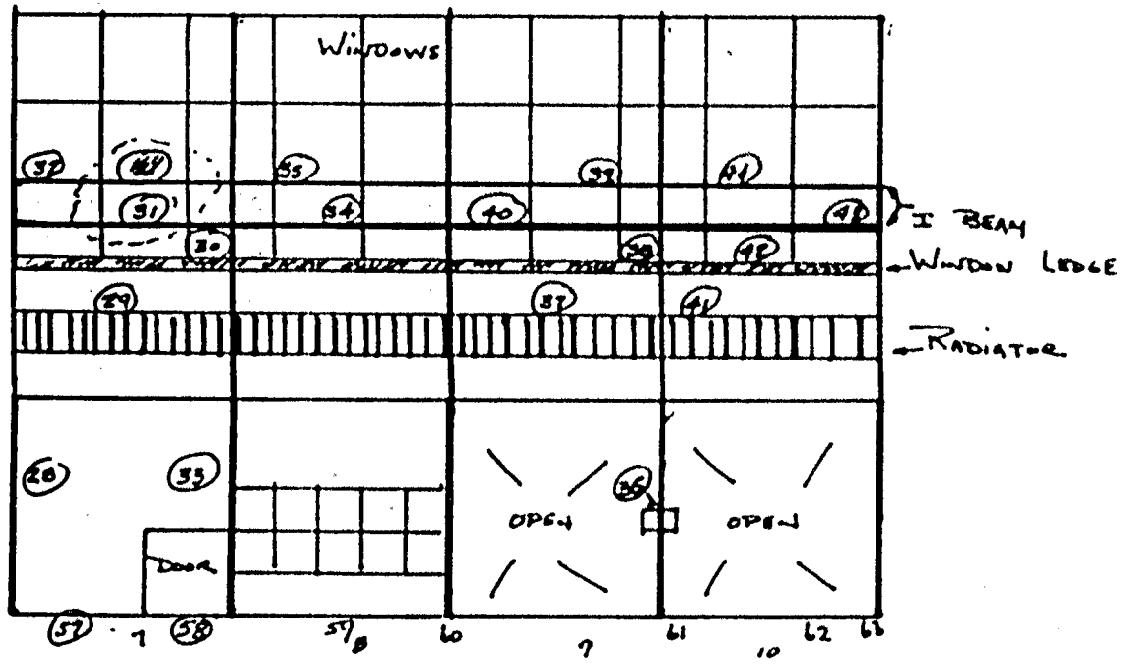
D-19623

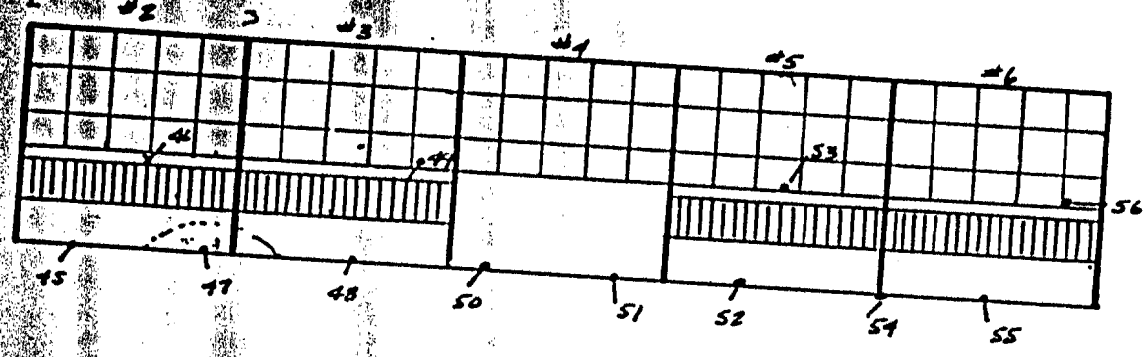
NORTH WALL INTERIOR
B'&C on SCAN



129527

2/7/94
CHAMPAIGN Valve

NORTH WALL
INTERIOR SEAM
8'8½" FLOOR GRID POINTS



D-19623

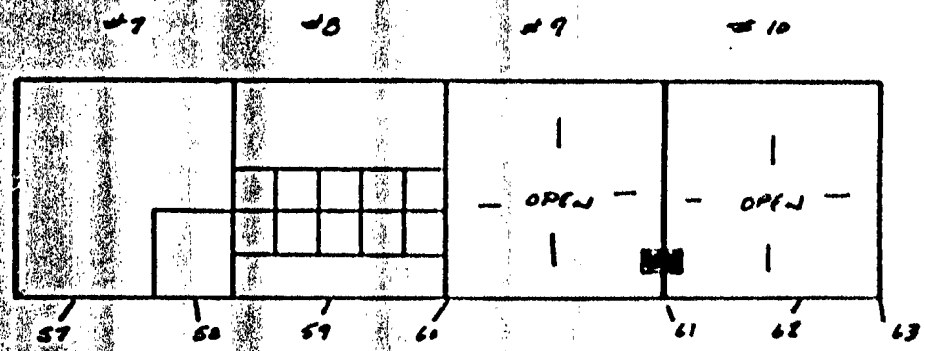
81

12952

0 0 2 3

D-19623

NORTH WALL INTERIOR
B'8 & α SCAN
FLOOR SURVEY AT GRID POINTS



129527

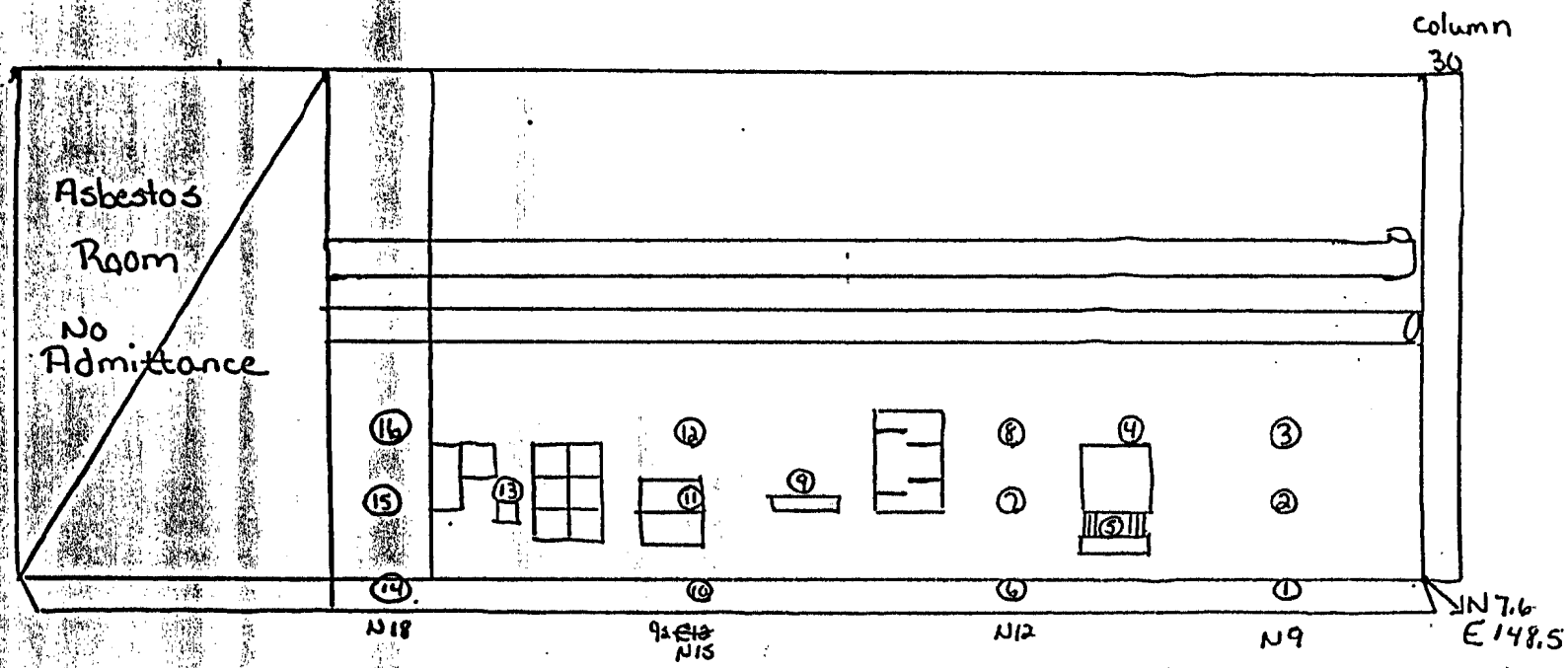
129507

East Wall Scan Data

E-WALL.XLS	Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
Location	Before Dust Removal		After Dust Removal			
Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	-12	799				
2	91	118				
3	222	562				
4	91	-266				
5	-41	148				
6	47	1332	-26	1273	-1	-4
7	32	-178				
8	61	651				
9	-26	-296				
10	-12	710				
11	-18	-326				
12	76	-89				
13	-12	-237				
14	-26	414				
15	3	-296				
16	120	-414				

0023

Bldg 23 East Wall



D-19693

19587

129507

Room AA-1 Scan Data

AA1.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	-3	829				
e	2	-16	-178				
r	3	51	414				
l	4	159	1243			2	25
z	5	226	1361			5	-25
e	6	-43	977				
n	7	213	829				
t	8	91	503				
a	9	65	385				
l	10	-16	355				
	11	226	661				
S	12	-3	178				
u	13	132	651				
r	14	38	947				
f	15	105	829				
a	16	-43	0				
c	17	-70	1006				
e	18	-3	917				
s	19	-56	769				
	20	145	-533				
	21	65	1273				
	22	24	0				
	23	-30	592				
	24	-43	1006				
	25	-56	414				
	26	186	326				
	27	-43	296				
	28	118	1598			-1	38
	29	-30	385				
	30	-3	1421			-1	-13
	31	-16	562				
	32	213	769				
	33	-8	565				
	34	19	-30				
	35	72	1100				
	36	19	743				
	37	19	714				
	38	19	1962	72	1457	7	-4
	39	45	327				
	40	72	476				
	41	152	505				
	42	-35	-387				
	43	19	-327				
	44	-8	714				
	45	19	773				
	46	19	89				
	47	72	1724	99	1397	7	42
	48	-8	149				
	49	-35	-119				

0023

129507

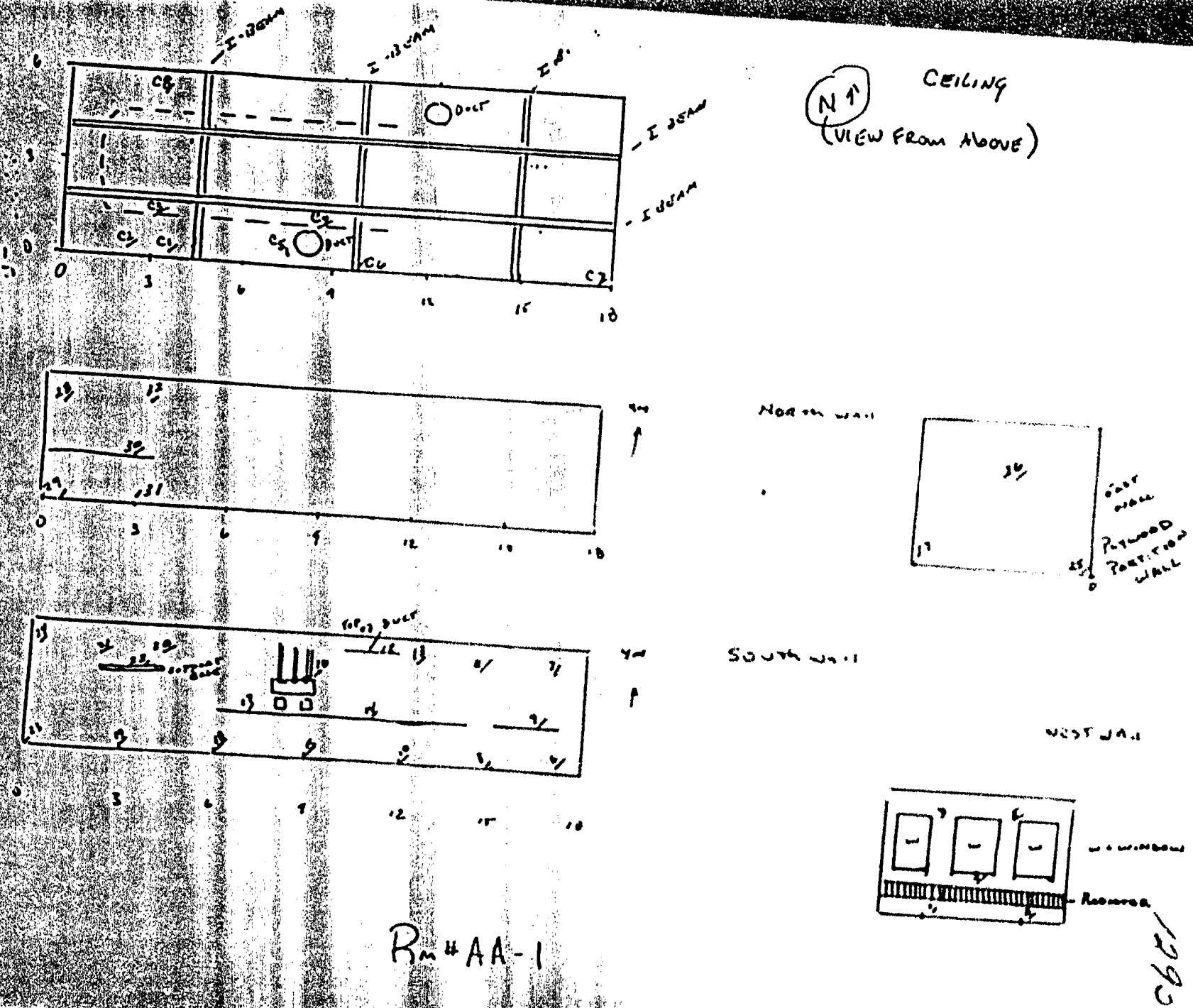
Room AA-1 Scan Data

AA1.XLS

Location	Location Number		Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
	50		-35	-327				
	33	*	111	357				
	34	*	32	1130	-4	1070	5	-4
	35	*	336	1130	270	922	2	-4
	36	*	-21	565				
	37	*	230	624				
	38	*	45	684				
	39	*	283	1249	230	1916	2	-29
	40	*	19	773				
	41	*	270	951	297	624	2	-33
	42	*	19	714				
	43	*	72	803				
	44	*	-34	743				
	45	*	270	862				
	46	*	72	-149				
	47	*	283	1070	270	1724	-1	-21
	48	*	-41	624				
	49	*	98	1665	270	1387	-1	38
	50	*	177	892				
	51	*	124	803				
C	1		65	118				
o	2		132	118				
l	3		-43	2641	283	3433	16	33
l	4		-3	0				
i	5		105	118				
n	6		-56	1460	91	829	5	-13
g	7		78	296				
	8		91	5830			-1	-29
	1	*	-6	832				
	2	*	19	3508	-34	2546	2	8
	3	*	111	1011	111	2468	13	-33
	4	*	19	149				
	5	*	257	2200	72	1338	7	-50
	6	*	85	297				
	7	*	45	832				
	8	*	72	-80				

* = readings were taken during Survey Number 133-DT-037, 12/29/94

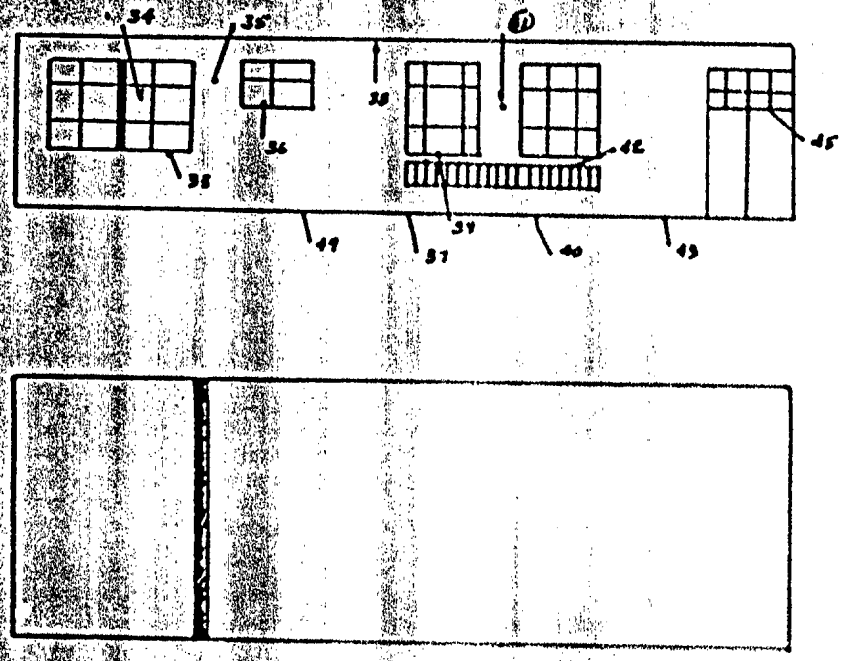
0023 1


N1
CEILING
(VIEW FROM ABOVE)
I-BEAM
DUCT
NORTH WALL
SOUTH WALL
WEST WALL
EAST WALL
PLYWOOD PARTITION WALL
WINDOW
RADIATOR

Rm # AA-1

Rm: AA-1
Bldg # 23

12-94
Champion Valve



U-U

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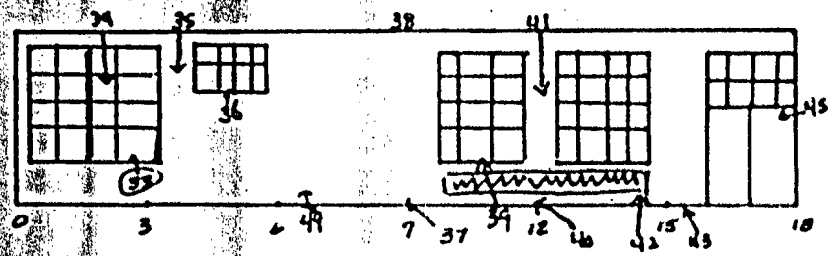
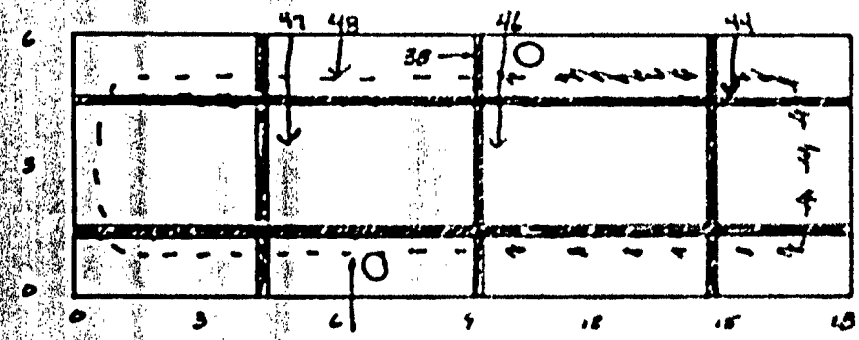
15 9527

#44-I

HP-210 7915

NORTH WALL





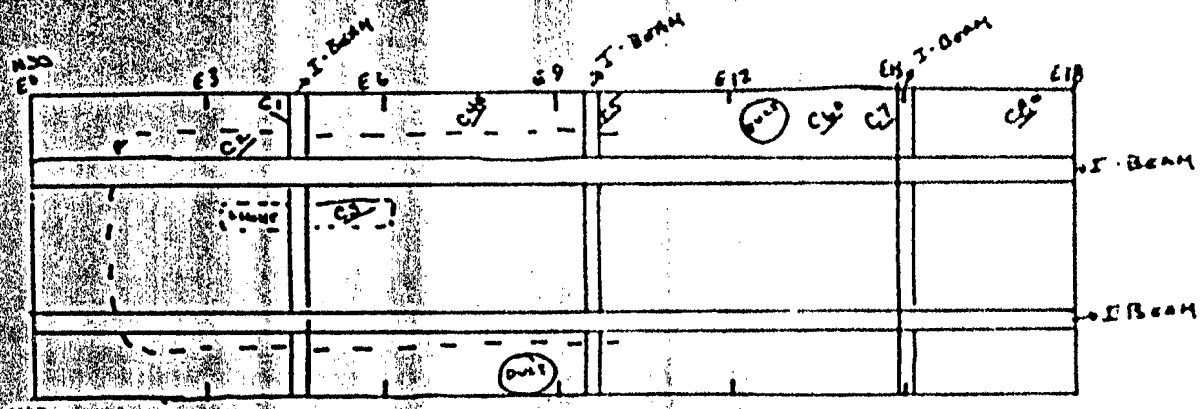
	Pre clean B'γ	α	Post clean B'γ α	
33.	62	4		
34.	42	6		
35.	80	10		
36	68	6		
37.	67	6		
38.	109	6		I-Bay
38A	92	10	5	
39	54	8		
40	59	10		
E. 41	69	16		
42	30	2		
43	32	6		
44	37	4		
45	69	6		
46	46	6		
47	101	10		
48A	90	12	5	
48	48	4		
49	39	2		
50	32	2		

LASCEI

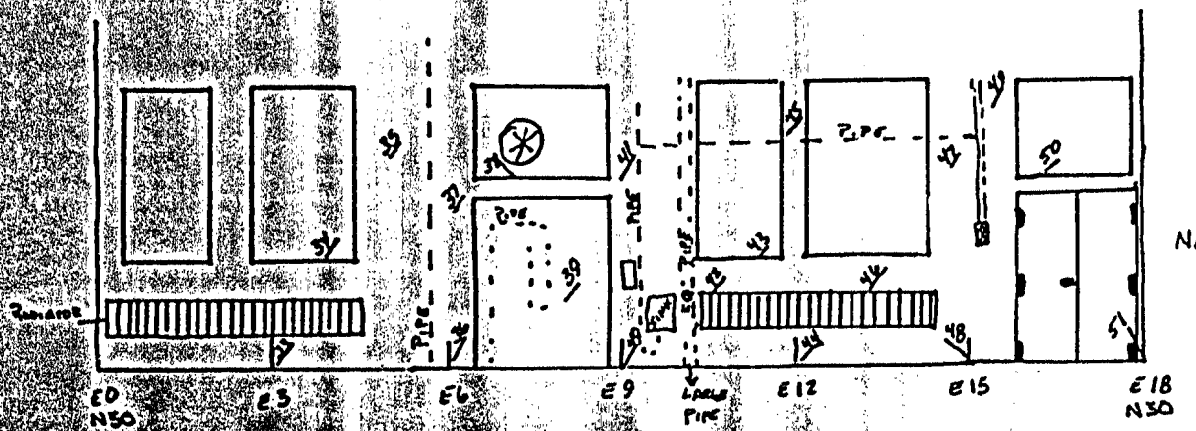
0 0 2 3

(VIEW FROM ABOVE)

CEILING IN ROOM # AA-1 N. WALL SIDE



* → READINGS WERE TAKEN ON TOP OF PIPES



NORTH WALL of ROOM # AA1

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Room AA-5 Scan Data

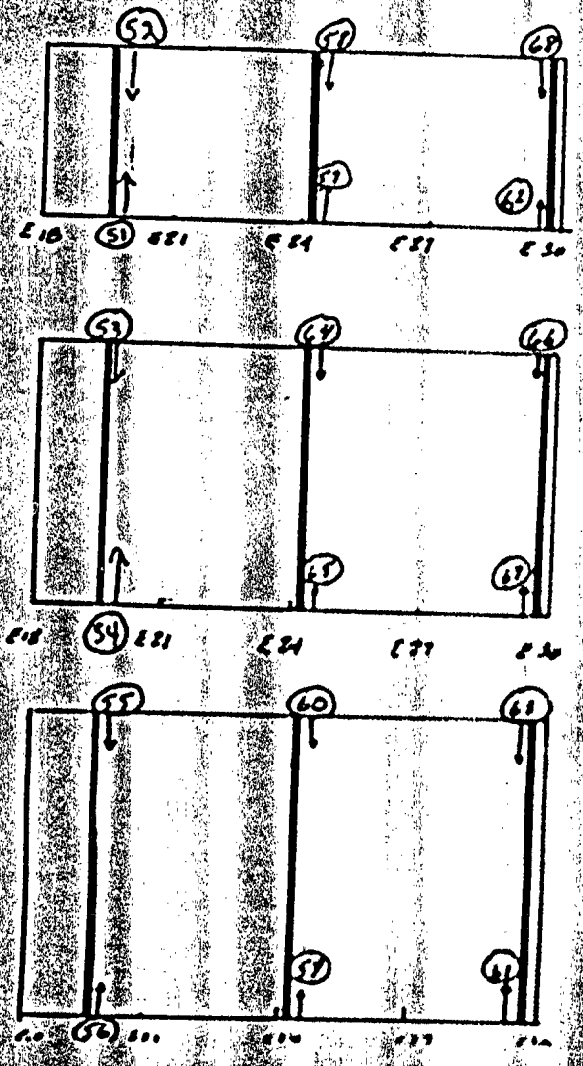
AA5.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	51	73	1605			2	-4
o	52	-28	1160			-1	-46
r	53	73	714				
i	54	301	1724	377	1308	13	29
z	55	-40	30				
o	56	-28	0				
n	57	-28	892				
t	58	-40	832				
a	59	-15	1011			2	-4
l	60	-28	595				
	61	-28	119				
S	62	-40	297				
u	63	-15	357				
r	64	-40	476				
f	65	-28	1070			-1	-42
a	66	-3	1427			-1	-33
c	67	-15	1665			2	-4
e	68	-40	178				
s	1	-81	1114		960	5	0
	2	-81	-192				
	3	-81	863				
	4	-56	653				
	5	-68	-192				
	6	-56	-115				
	7	-81	-499				
	8	-81	192				
	9	-68	-192				
	10	-81	1728		1498	-1	50
	11	-43	1498		2189	2	-29
	12	-43	-115				
	13	-68	269				
	14	-81	-192				
	15	-81	863				
	16	-56	-499				
	17	-43	-192				
	18	-94	576				

0023

PRS-I - 7859

43-5 - 7170



North Wall

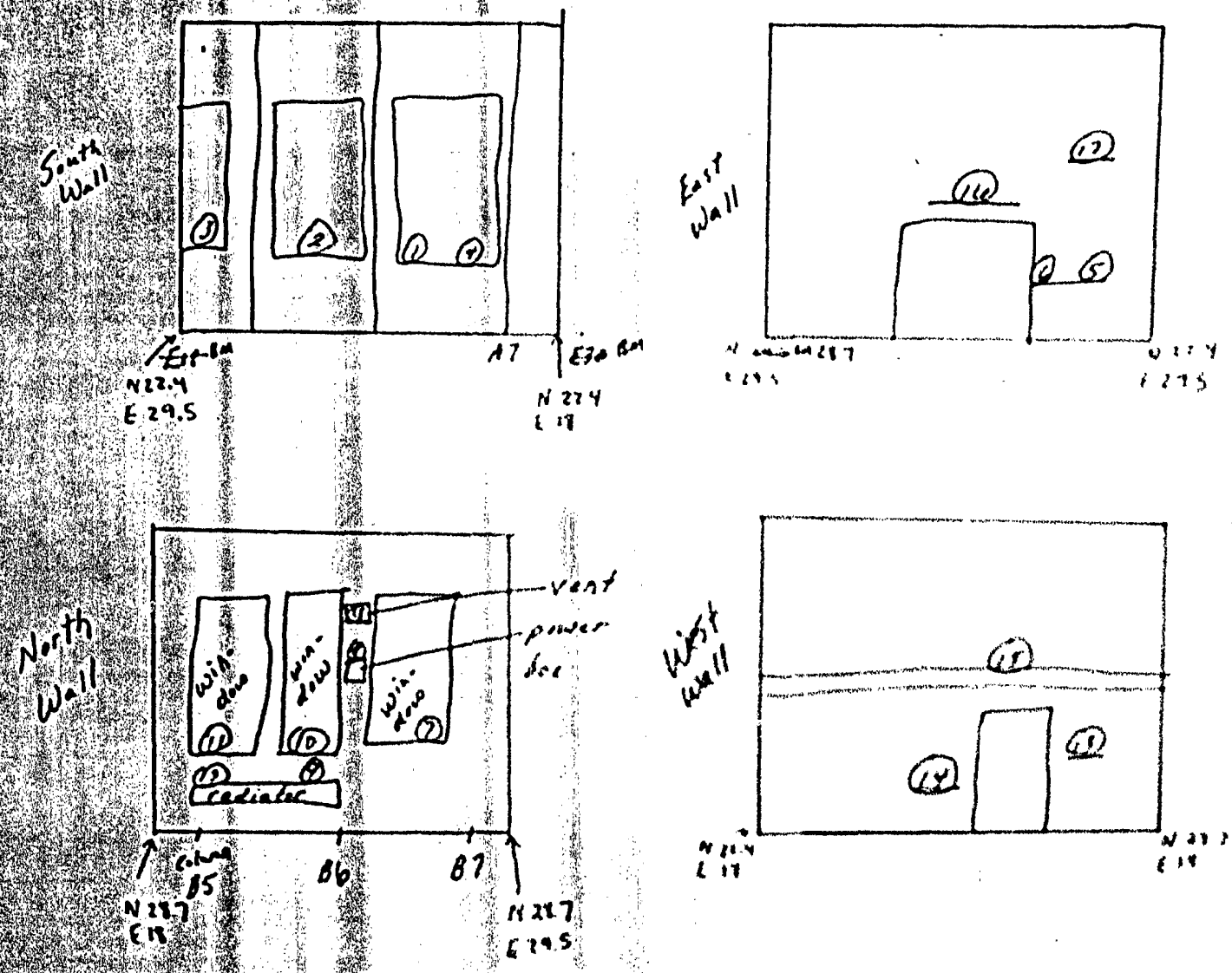
Ceiling

South Wall

	B'8	∝
51	88	10
52	73	2
53	58	10
54	92	28
54A	78	34
55	1	
56	2	
57	2	
58	1	
59	3	
60	2	
61	2	
62	1	
63	5	
64	1	
65	2	
66	4	
67	3	
68	1	

129527

Room AA-3



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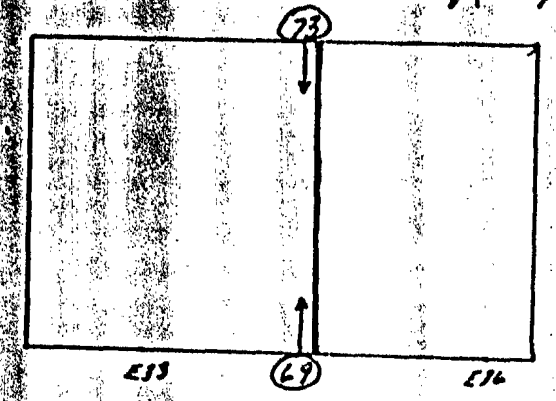
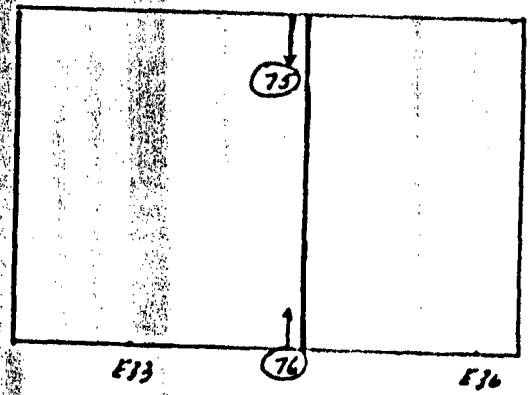
Room AA-7 Scan Data

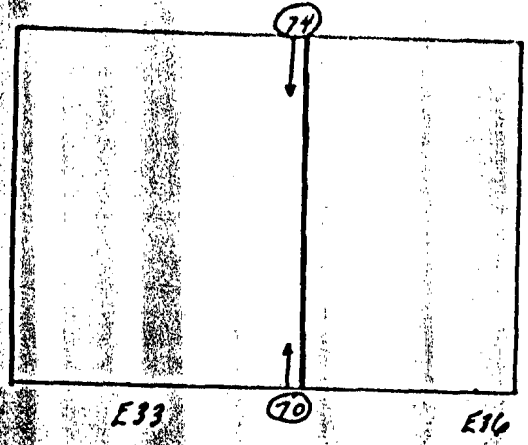
AA7.XLS

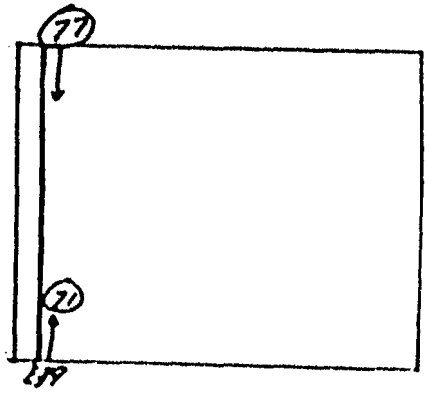
Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferable (DPM/100 cm^2) Alpha	Transferable Beta/Gamma
H	69	-3	1130			2	8
e	70	-28	0				
r	71	-40	714				
i	72	-28	-357				
z	73	-28	714				
o	74	-15	0				
n	75	-3	1130			2	8
t	76	-40	772				
a	77	-28	1011			5	13
l	78	-40	832				
	79	-15	1189			-1	0
S	80	-3	832				
u							
r							
f							
a							
c							
e							
s							

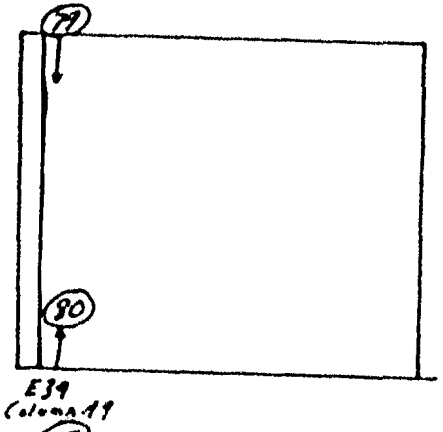
Moo, nAA-1

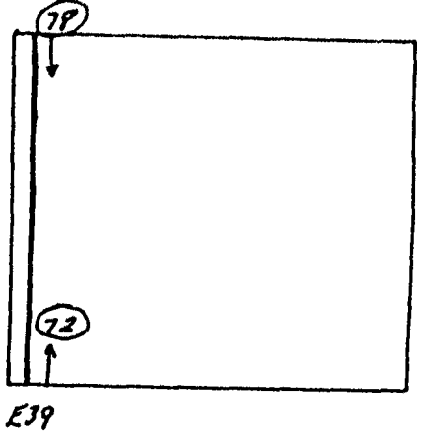












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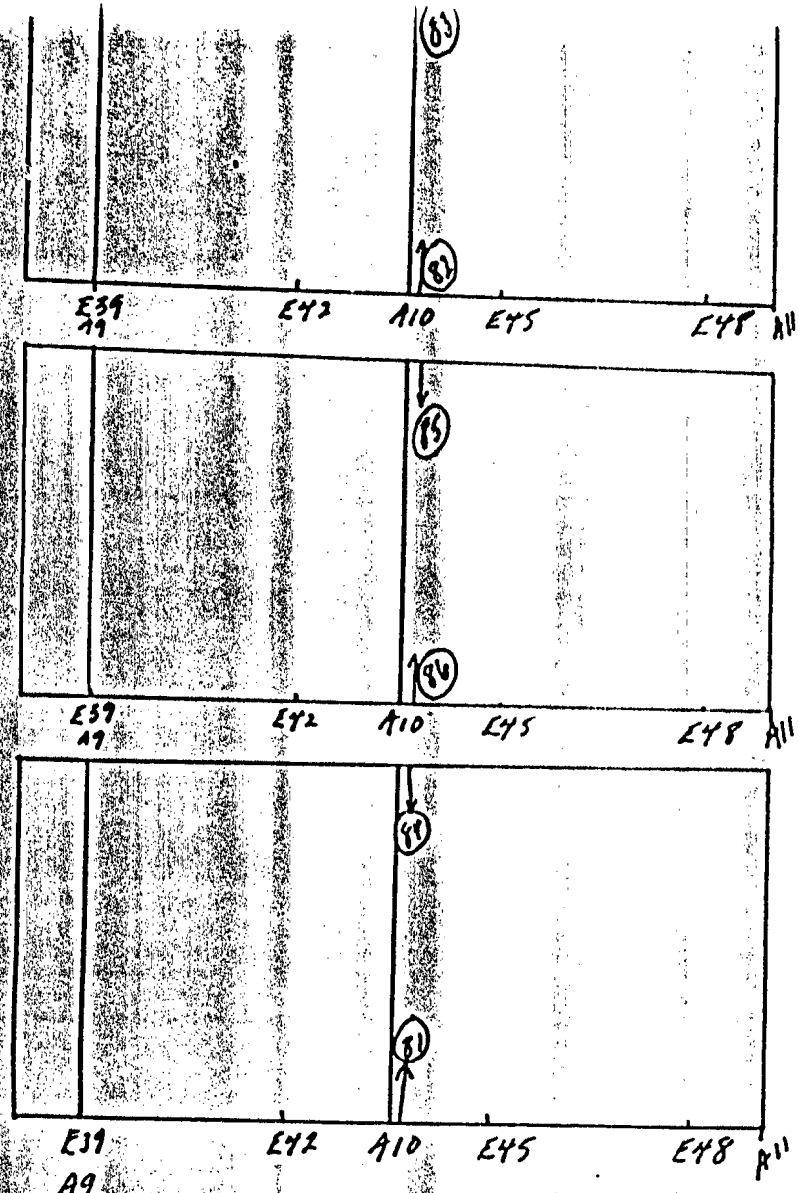
Room AA-9 Scan Data

AA9.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	81	-23	59				
o	82	94	535				
r	83	-53	1070			2	-4
i	84	6	-535				
z	85	-23	595				
o	86	-53	714				
n							
t							
a							
l							
S							
u							
r							
f							
a							
c							
e							
s							



Note: Truss A9 was done on a previous survey.

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Room AA-11 Scan Data

AA11.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferable (DPM/100 cm^2) Alpha	Transferable Beta/Gamma
H	1	18	59				
o	2	39	207				
r	3	8	266				
i	4	29	178				
z	5	-2	89				
o	6	-22	385				
n	7	-2	178				
t	8	69	1065	39	562	1	-21
a	9	49	769				
l	10	29	148				
	11	39	266				
S	12	-12	296				
u	13	-2	621				
r	14	-2	621				
f	15	100	-444				
a	16	39	296				
c	17	59	947				
e	18	59	30				
s	19	-2	296				
	20	-2	-207				
	21	69	148				
	22	69	681				
	23	18	1036	18	740	-1	-4
	24	59	769				
	25	-22	30				
	26	100	-30				
	27	29	30				
	28	29	178				
	29	49	385				
	30	120	89				
	31	59	-30				
	32	69	-296				
	33	-12	59				
	34	29	474				
	35	80	-444				
	36	172	-355				
	37	49	562				
	38	8	917				
	39	49	503				
	40	29	1273	-12	59	-1	-17
	41	59	799				
	42	-22	1332	-12	651	-1	-13
	43	49	769				
	44	8	651				
	45	29	799				
	46	49	-148				
	47	29	1213	100	1125	-1	25
	48	59	592				

129509

Room AA-11 Scan Data

AA11.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Direct (DPM/100 cm^2) Before Dust Removal Beta/Gamma	Direct (DPM/100 cm^2) After Dust Removal Alpha	Direct (DPM/100 cm^2) After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable (DPM/100 cm^2) Beta/Gamma
	49	-12	582				
	50	120	148				
	51	69	69				
	52	-12	1008	-12	696	2	17
	53	69	-256				
	54	8	429				
	55	49	582				
	56	18	681				
	57	18	-148				
	58	80	-49				
	59	29	474				
	60	49	-582				
	61	39	-356				
	62	59	69				
	63	18	-30				
	64	29	237				
	65	29	-365				

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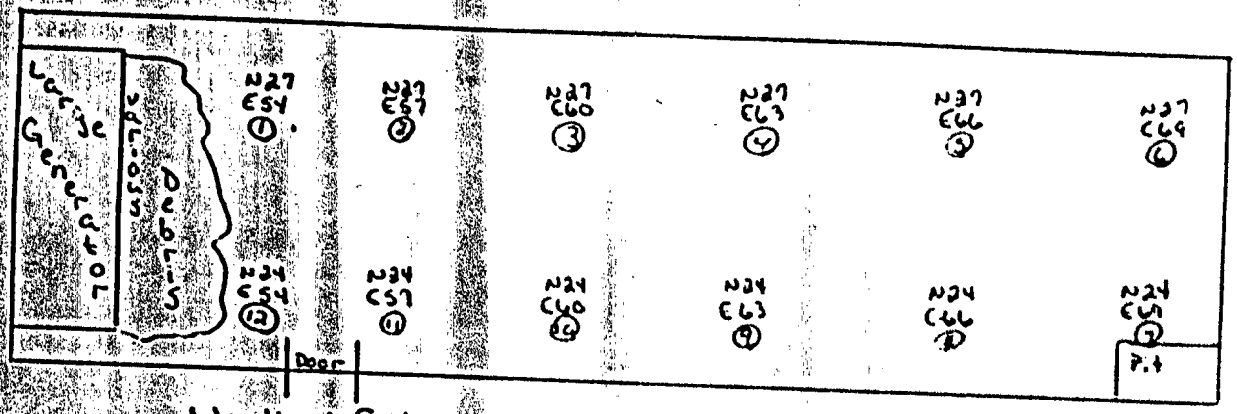
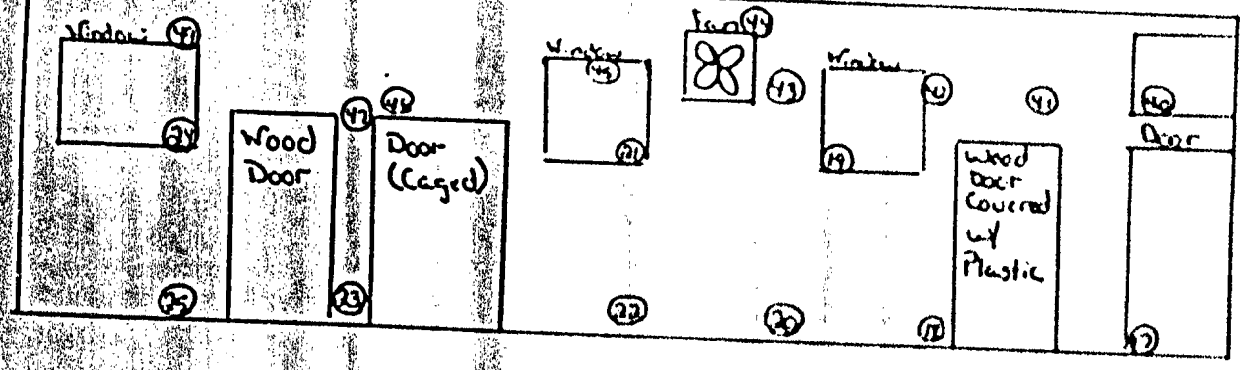
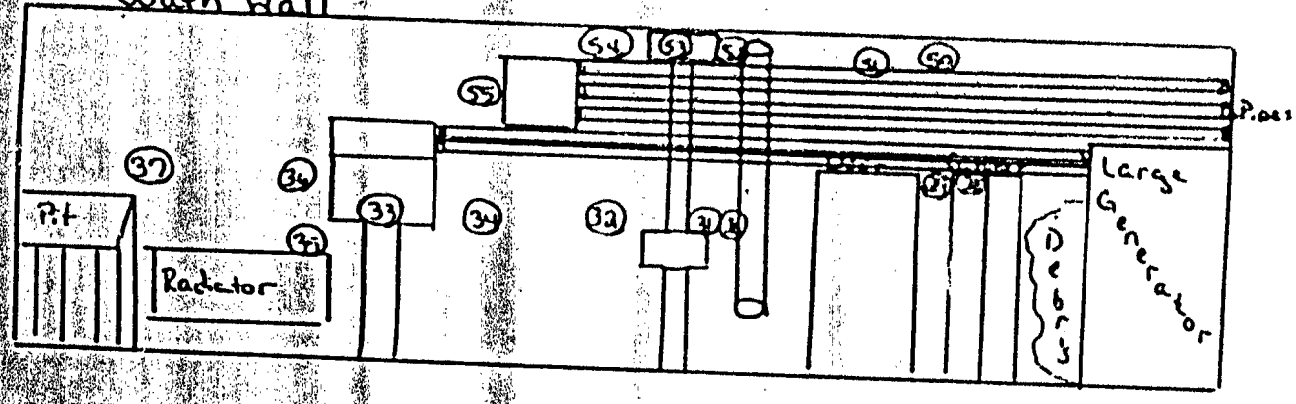
Floor Room AA-11



North Wall



South Wall



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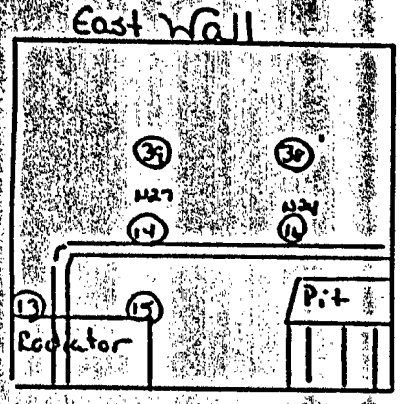
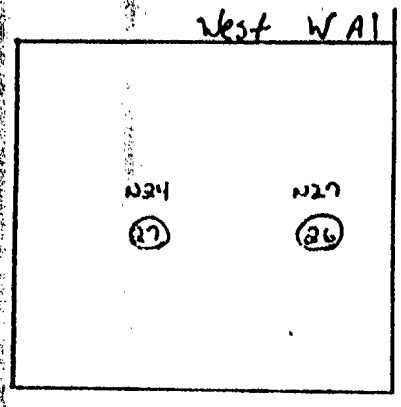
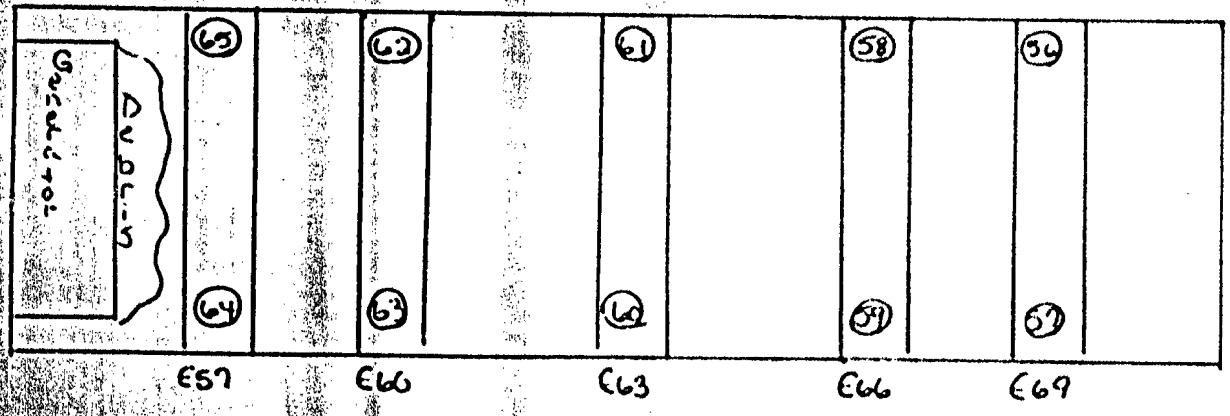
Room AA-11





No upper readings done due to inability to get a ladder behind the generator.

Ceiling (Top View Facing North)



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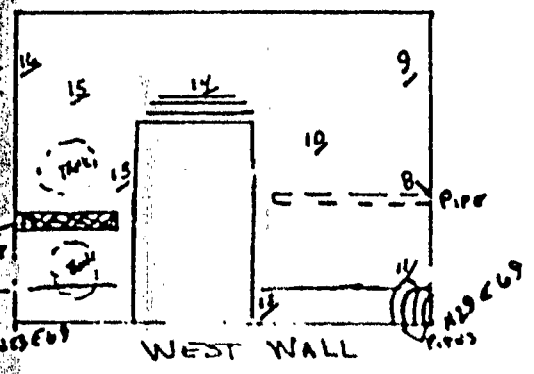
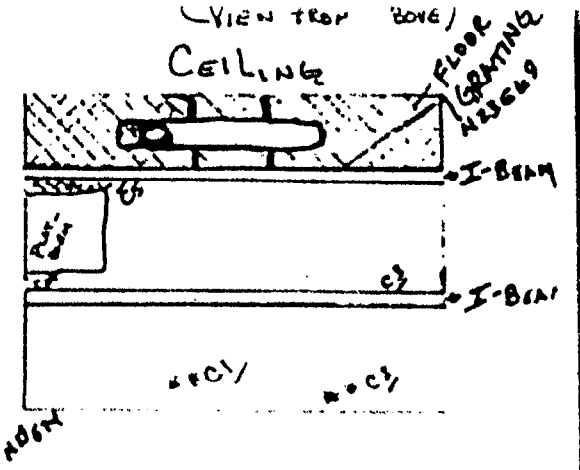
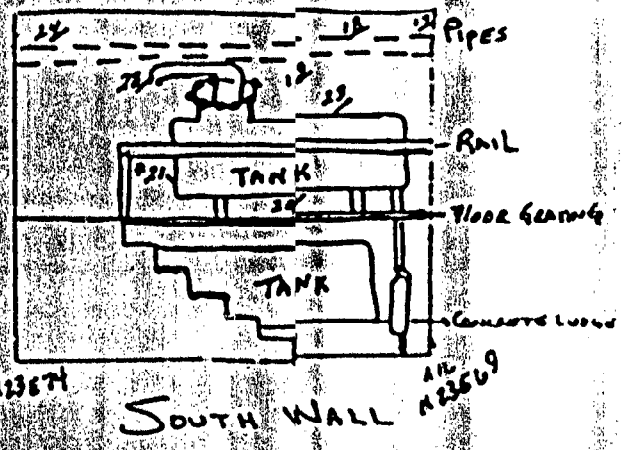
Room AA-16 Scan Data

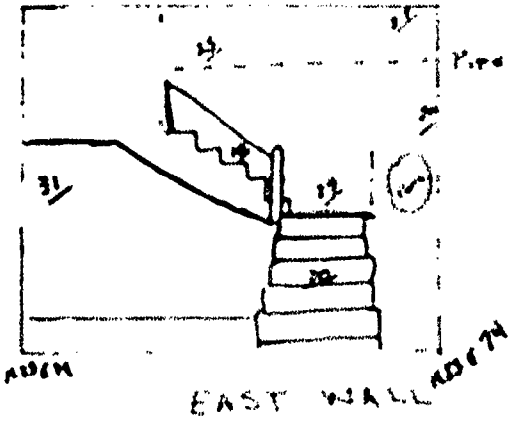
AA16.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	32	951	151	614	-1	35
o	2	204	1188	350	892	-1	4
r	3	257	505				
i	4	244	476				
z	5	151	535				
o	6	-21	476				
n	7	138	536				
t	8	177	327				
a	9	350	1070	98	476	5	71
l	10	257	565				
	11	-48	476				
S	12	-61	595				
u	13	310	743				
r	14	191	1160	151	922	-1	13
f	15	124	505				
a	16	191	595				
c	17	244	535				
e	18	164	981	98	892	2	3
s	19	164	624				
	20	244	297				
	21	-8	119				
	22	-34	89				
	23	-87	535				
	24	177	981	98	892	-1	-31
	25	-48	505				
	26	19	1397	-61	624	2	0
	27	-101	595				
	28	-61	-208				
	29	-8	238				
	30	5	-119				
	31	5	505				
C	1	85	-149				
e	2	98	0				
i	3	72	1070	72	416	-1	29
l	4	5	149				
l	5	85	238				
n							
g							









* #21 READINGS TAKEN ON SUPPORT BEAM FOR TANK (NOT SHOWN)

** C1 & C2 READINGS WERE TAKEN ON BOTTOM SIDE OF STAIR CASE.

LESLEY

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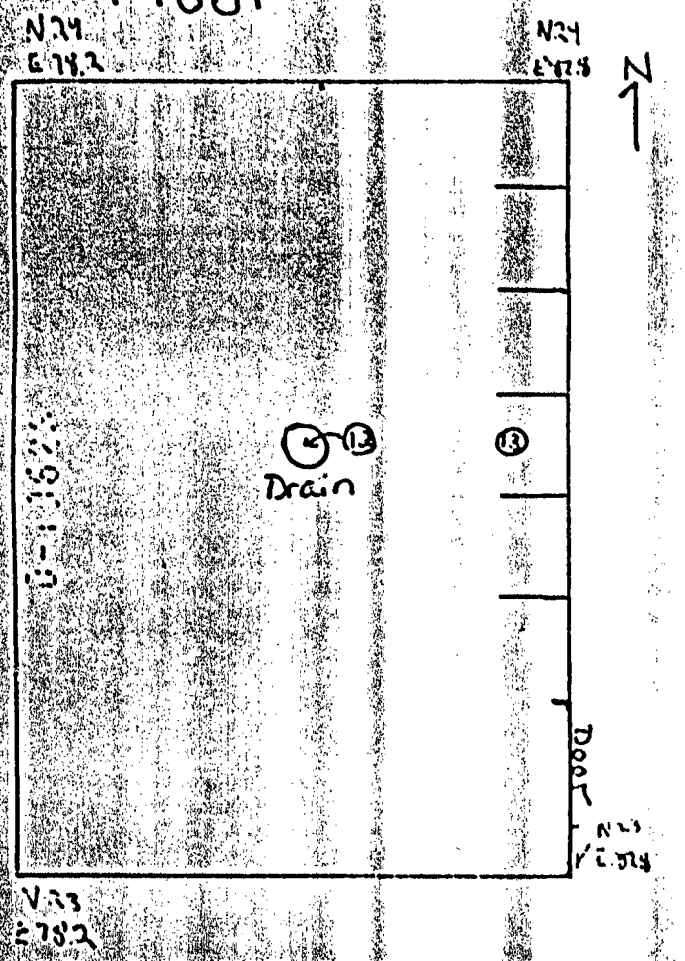
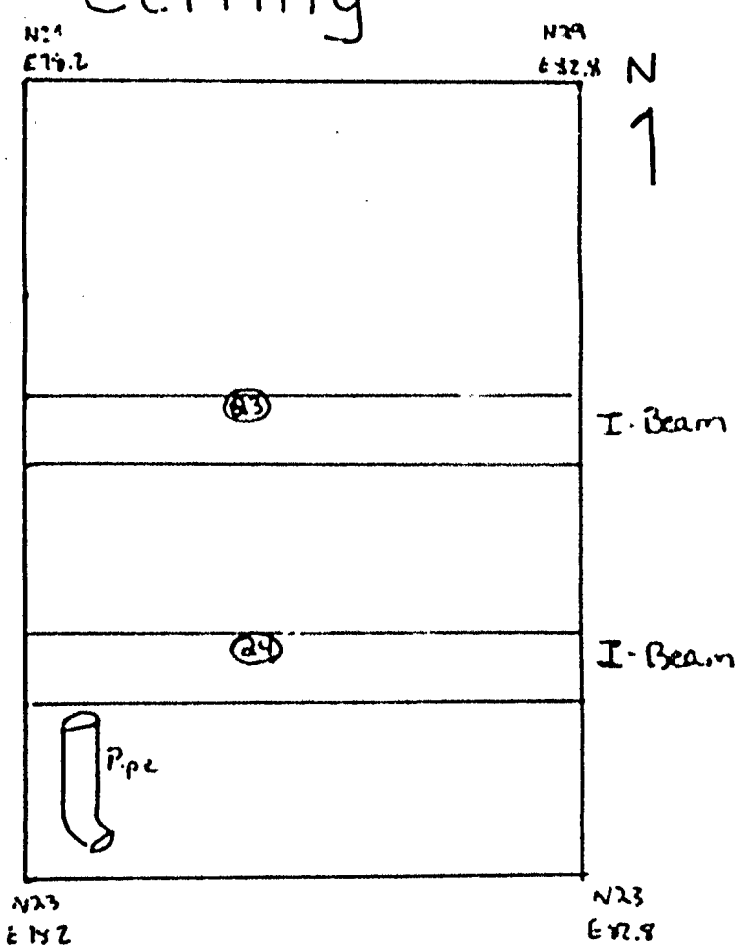
Room AA-17 Scan Data

AA17.XLS

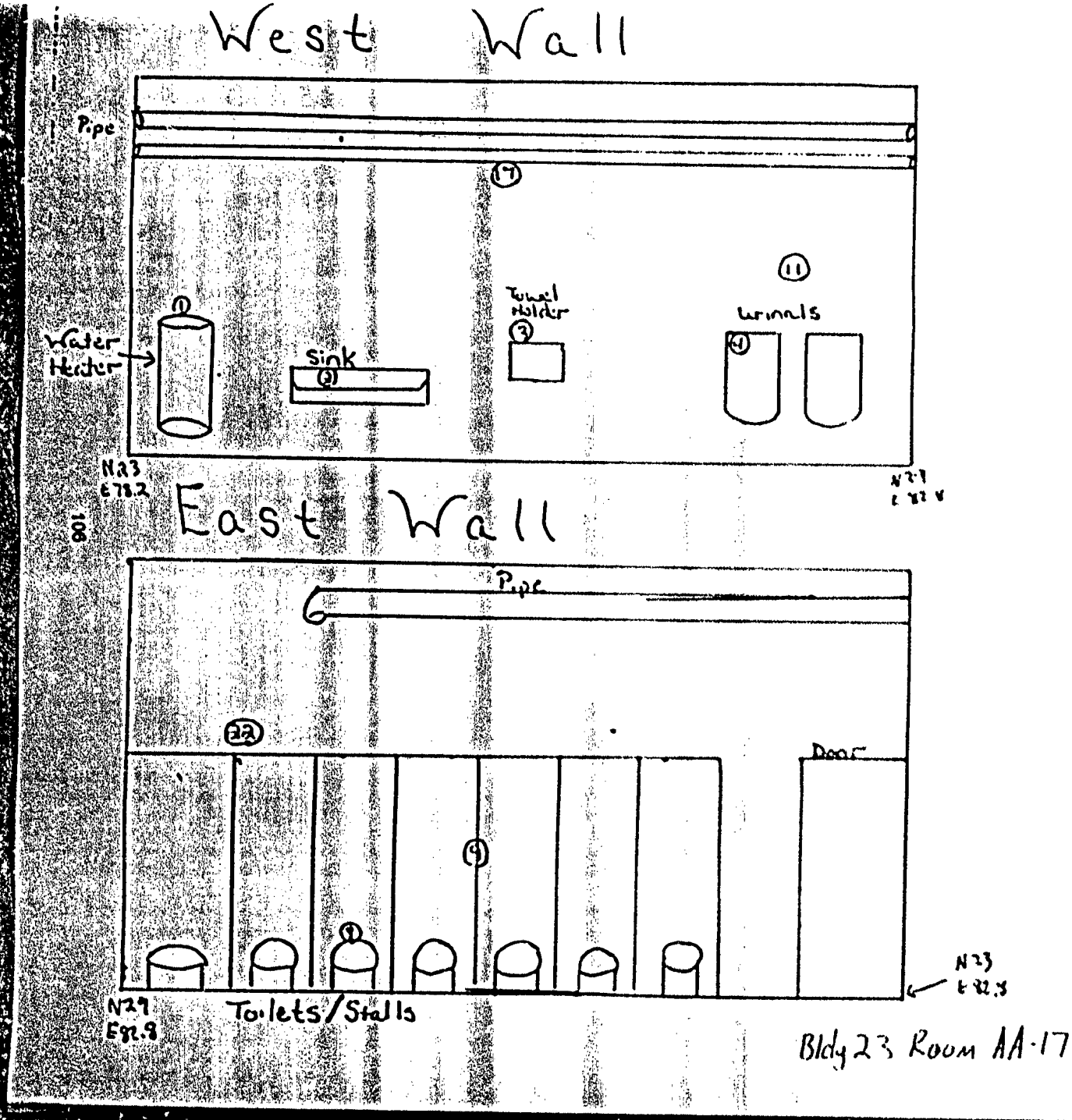
Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	23	-149				
o	2	-6	297				
r	3	-64	714				
i	4	23	476				
z	5	-6	297				
o	6	9	714				
n	7	23	-238				
t	8	-35	862				
a	9	-79	178				
l	10	111	119				
	11	155	535				
S	12	-64	178				
u	13	-6	595				
r	14	155	416				
f	15	140	-119				
a	16	111	446				
c	17	38	565				
e	18	82	476				
s	19	53	59				
	20	23	505				
	21	-64	387				
	22	-6	714				
	23	23	-89				
	24	23	119				
	25	-35	149				
	26	111	654				
	27	9	178				



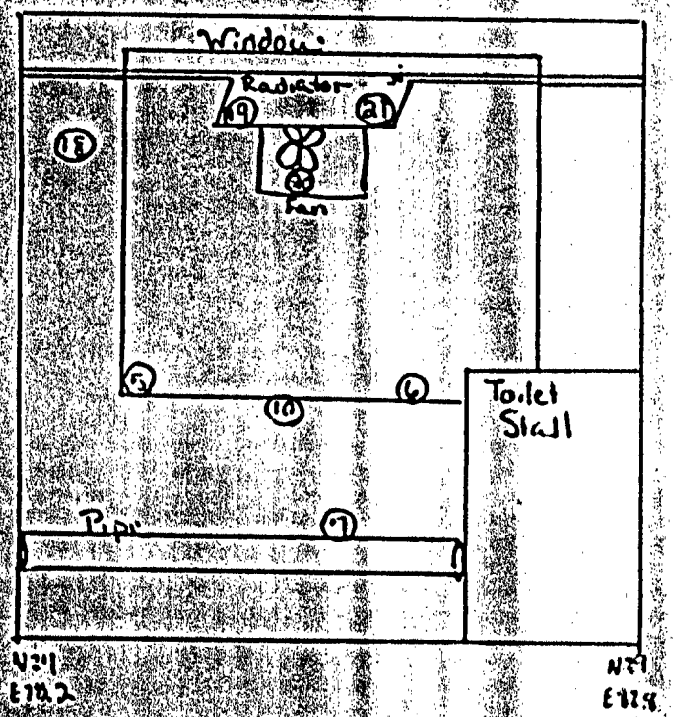


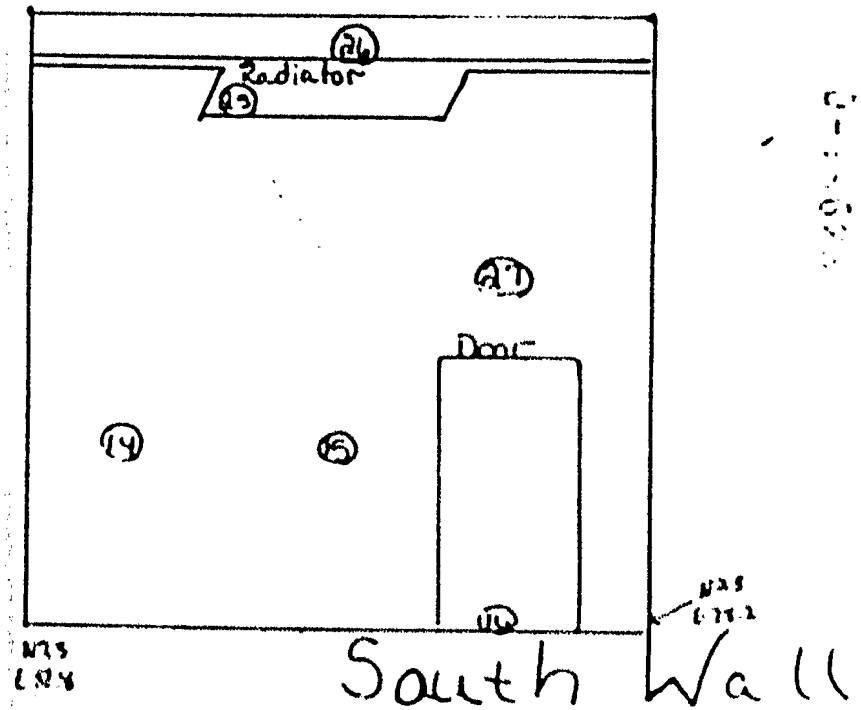
Bldg 23 Room AA-17

LASBER



Bldg 23 Room AA-17





Bldg 23 Room AA-17

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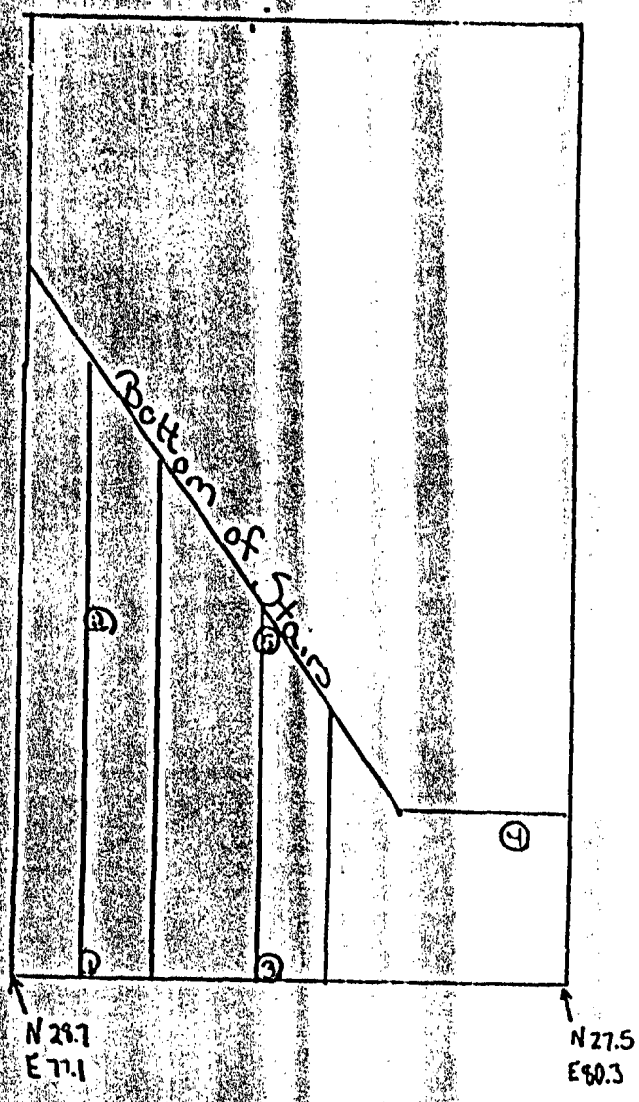
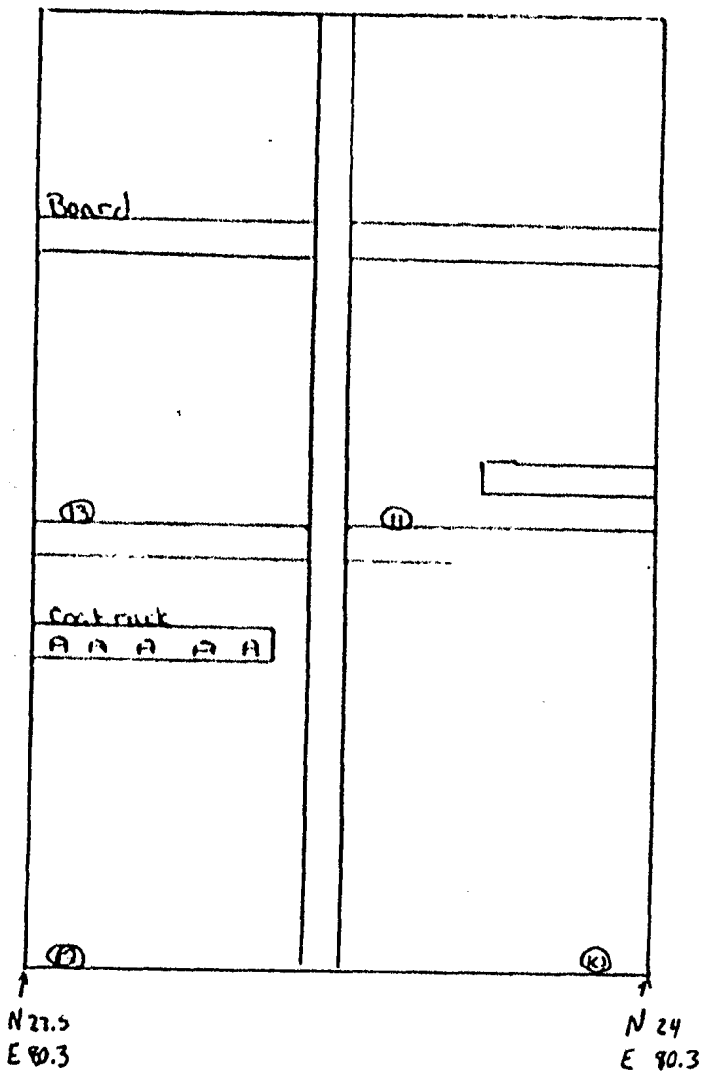
Room AA-18 Scan Data

AA18.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	-9	-119				
o	2	196	1160	167	238	2	0
r	3	-23	565				
i	4	181	0				
z	5	210	654				
o	6	64	476				
n	7	-23	149				
t	8	94	-367				
a	9	94	149				
l	10	20	357				
	11	50	416				
S	12	-9	208				
u	13	6	-208				
r	14	35	476				
f	15	6	-89				
a	16	76	327				
c	17	123	-327				
e							
s							





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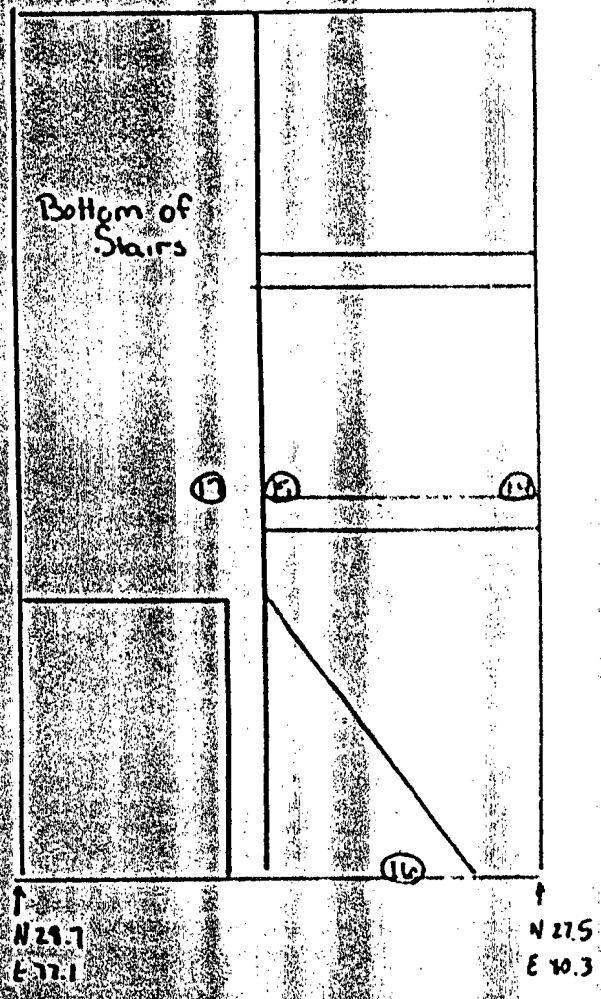
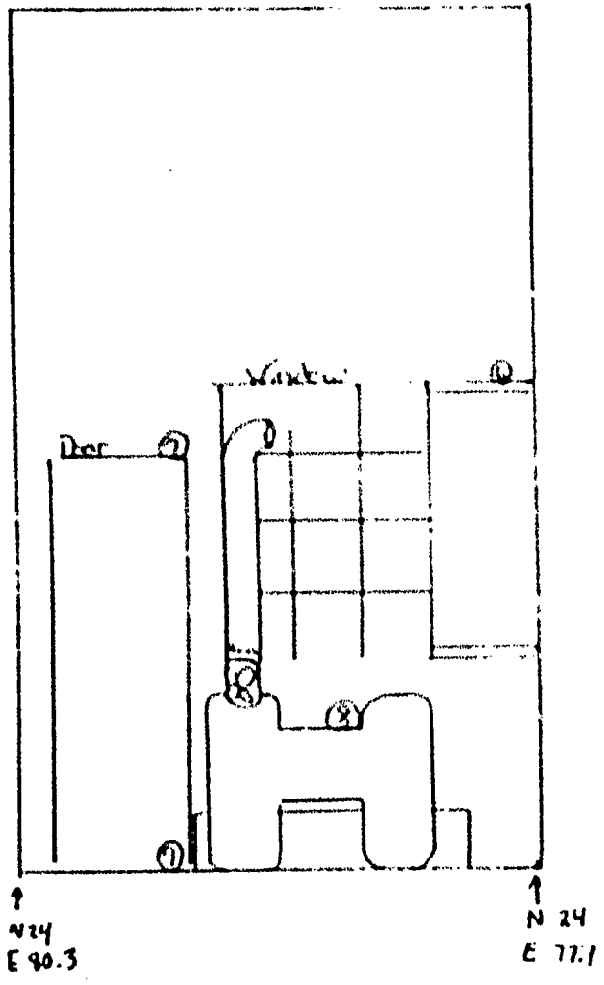
Room AA-18

North Wall



South Wall



0 0 2 3

129507

Room AA-20 Scan Data

AA20.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	254	416				
o	2	94	-327				
r	3	64	-505				
i	4	299	-119				
z	5	-9	59				
o	6	-9	-297				
n	7	20	-892				
t	8	313	535				
a	9	35	208				
l	10	152	-89				
	11	20	-59				
S	12	20	59				
u	13	6	-30				
r	14	94	327				
f	15	106	565				
a	16	-23	535				
c	17	20	387				
e	18	6	89				
s	19	-23	-178				
	21	50	-30				
	21	50	-535				
	22	6	-30				
	23	50	-238				
	24	-38	-238				
	25	-9	327				
	26	64	-89				
	27	20	-89				
	28	123	595				
	29	20	505				
	30	-53	297				
	31	-67	-268				
	32	50	446				
	33	35	-357				
	34	-23	-208				
	35	20	892				
	36	-23	149				
	37	20	-208				
	38	210	327				
	39	6	0				
	40	50	-416				
	41	-50	89				
	42	23	268				
	43	38	0				
	44	9	743				
	45	-35	624				
	46	38	297				
	47	67	297				
	48	38	446				

129507

Room AA-20 Scan Data

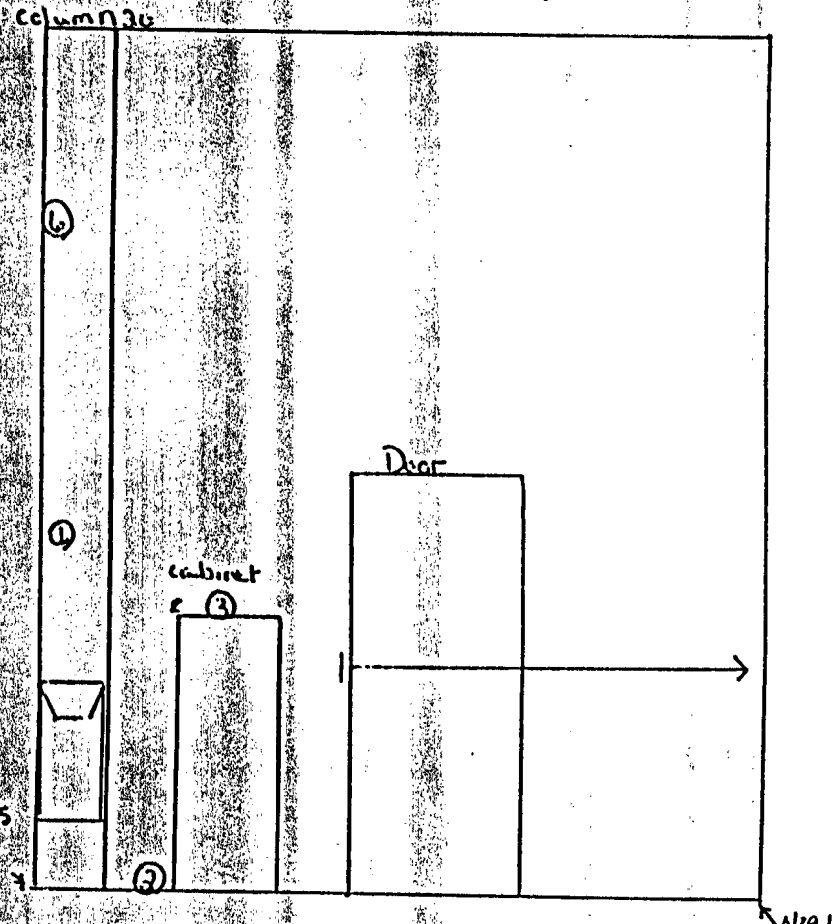
AA20.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
	49		892				
	50	23	89				
	51	38	30				
	52	9	773				
	53	67	357				
	54	53	476				
	55	23	951				
	56	9	178				
	57	169	387				
	58	23	-446				
	59	-35	654				
	60	184	327				

0023

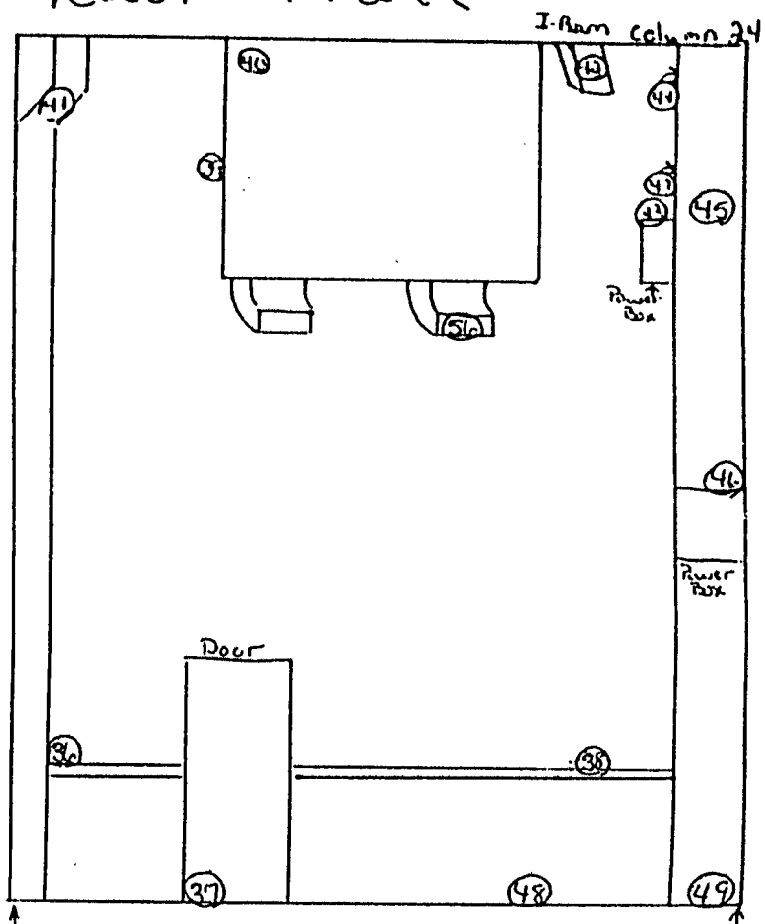
Room AA-20

West Wall

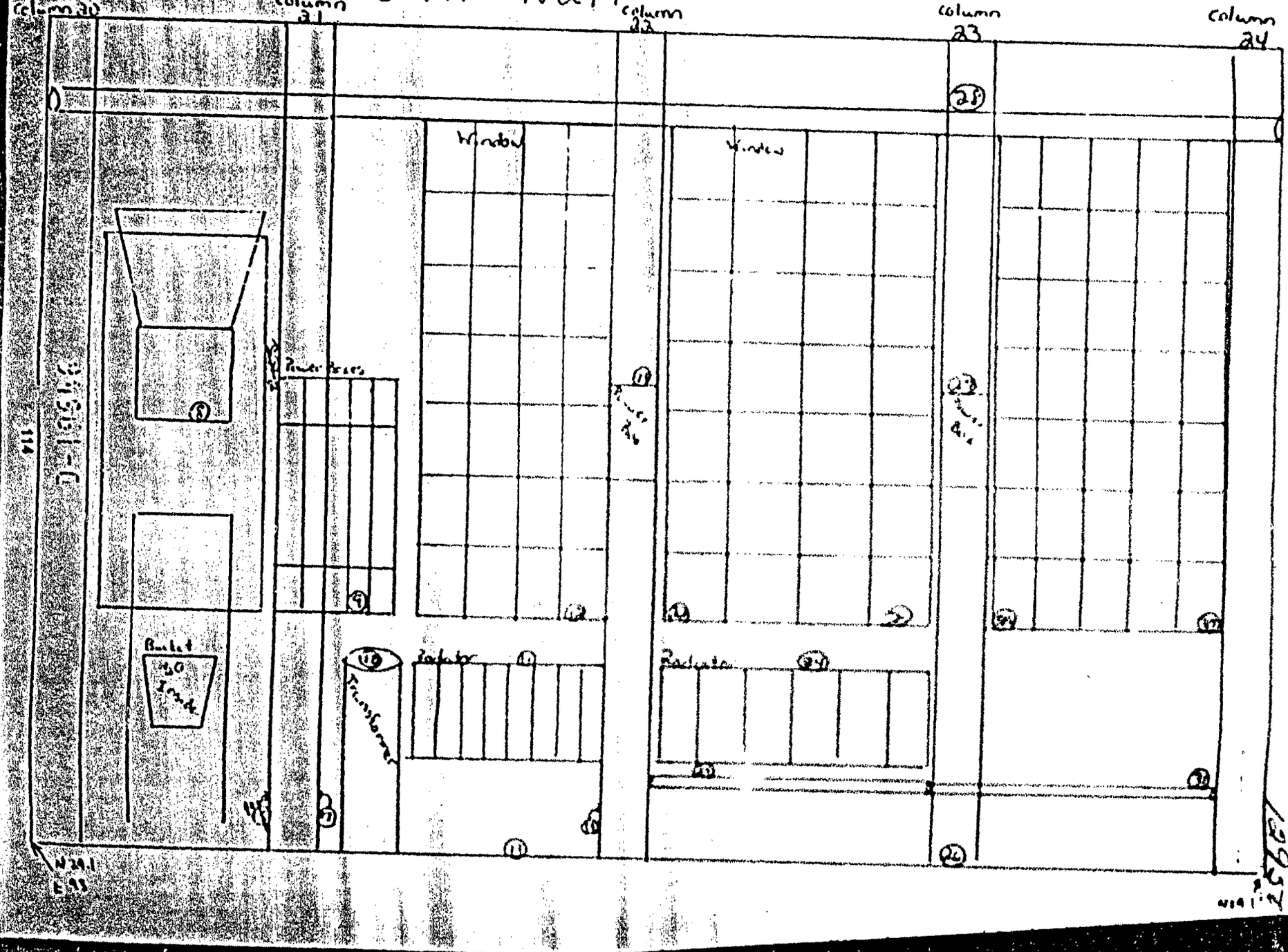


No readings from Door on
because of wooden planks over
water.

East Wall

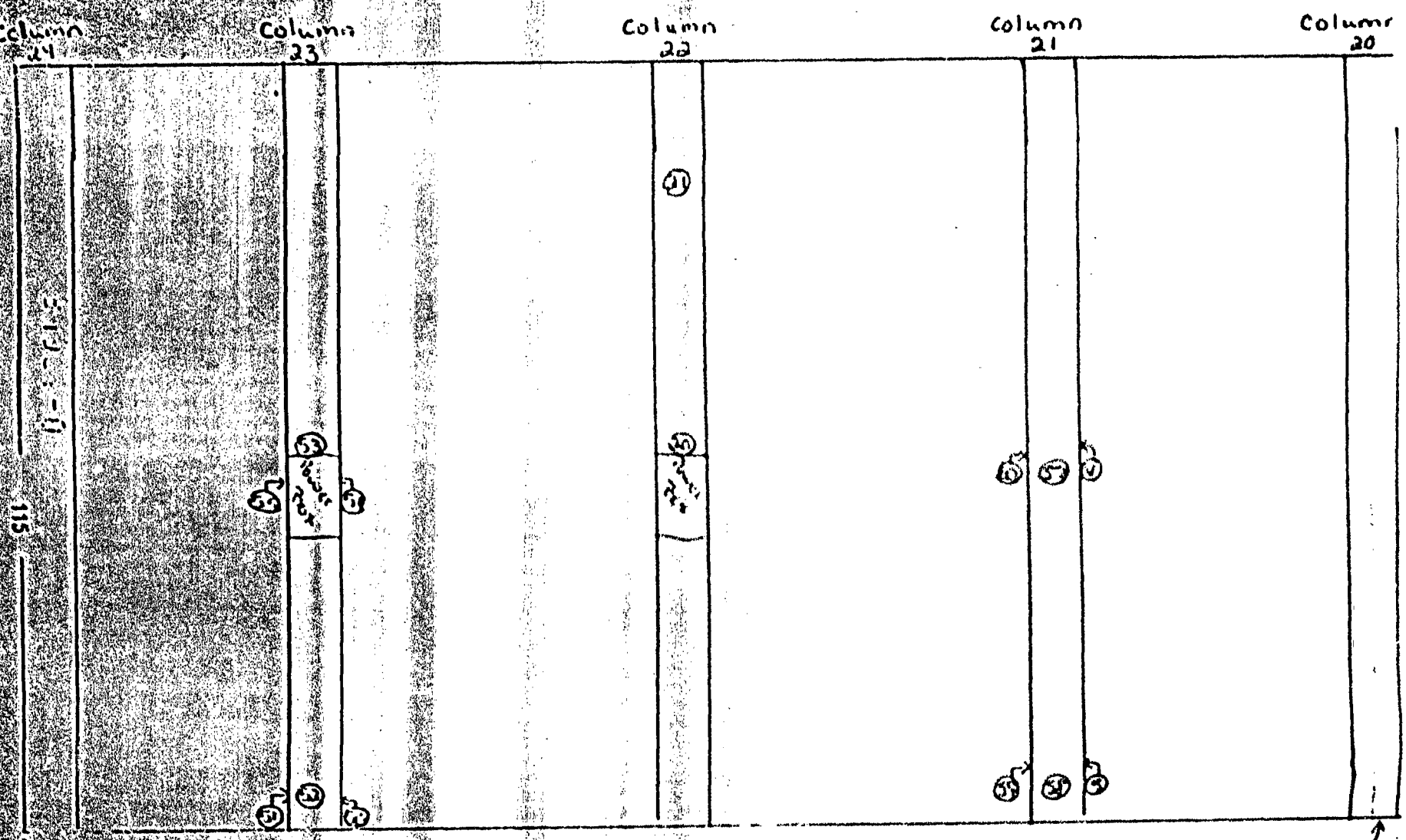


North Wall Room AA-20



0023

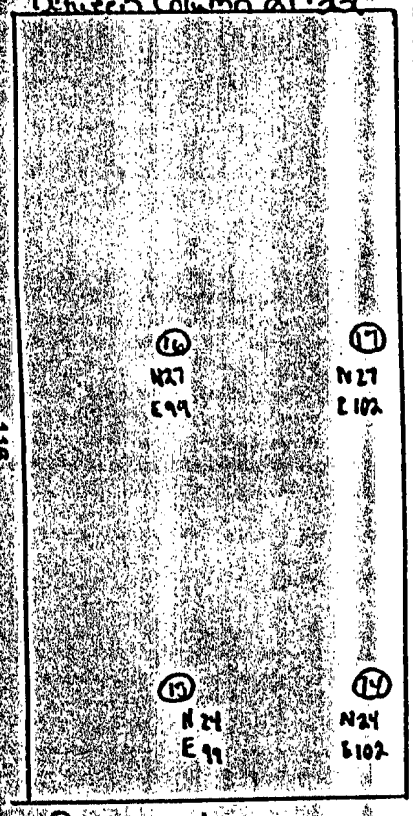
Room AA-20



View from Inside Room Looking Out

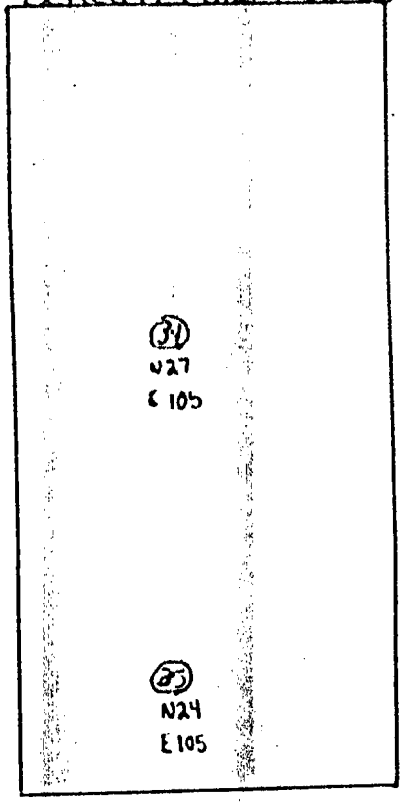
Room AA-20

Floor
Between Columns 21+22

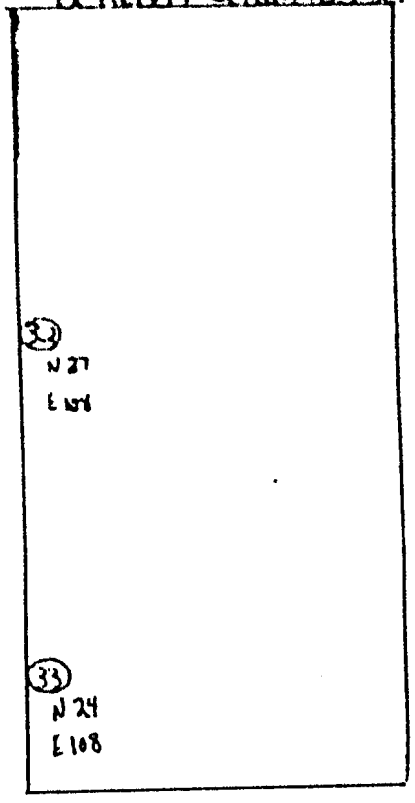


Readings taken on
Grid Points

Floor
Between Columns 22+23



Floor
Between Columns 23+24



116

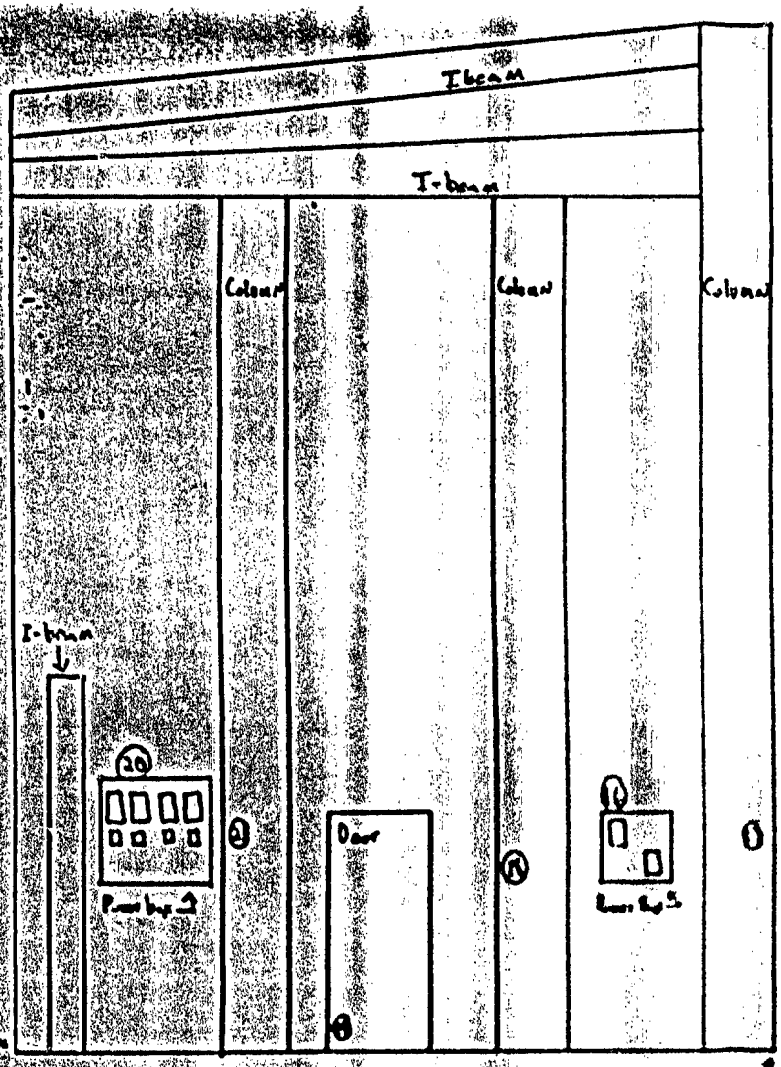
129507

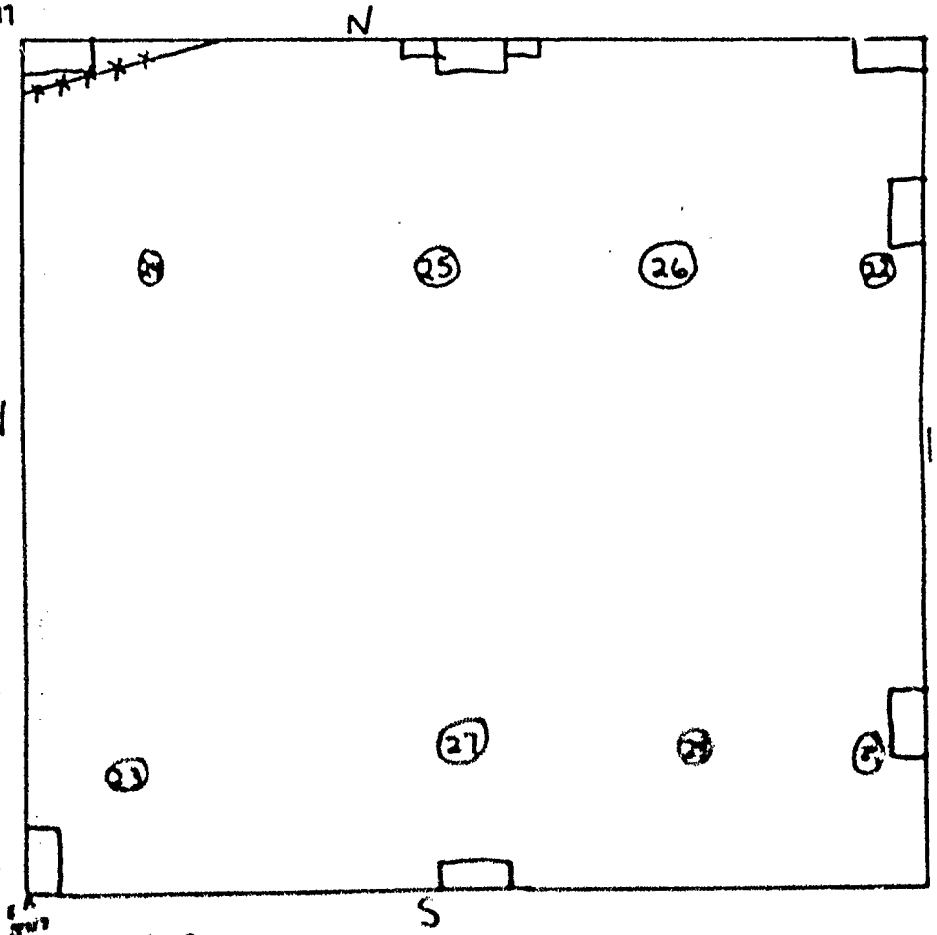
0023

Room AA-25 Scan Data

AA25.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	67	-59				
e	2	53	208				
r	3	-44	684				
i	4	199	505				
z	5	-50	119				
e	6	-6	505				
n	7	23	446				
i	8	-20	476				
a	9	67	179				
l	10	111	505				
	11	-20	30				
S	12	67	149				
u	13	67	743				
r	14	23	446				
f	15	-20	714				
a	16	38	268				
c	17	82	714				
e	18	243	1605	184	1278	-1	-29
s	19	-20	208				
	20	-35	416				
	21	82	951				
	22	-6	0				
	23	-20	178				
	24	-6	119				
	25	9	327				
	26	23	565				
	27	-6	-89				
	28	-50	0				
	29	-6	387				
	30	169	1011	286	654	-1	-17
	31	213	714				
	32	-20	1070	111	446	-1	-21
	33	96	595				
	34	53	1070	53	1457	2	-59
	35	140	387				





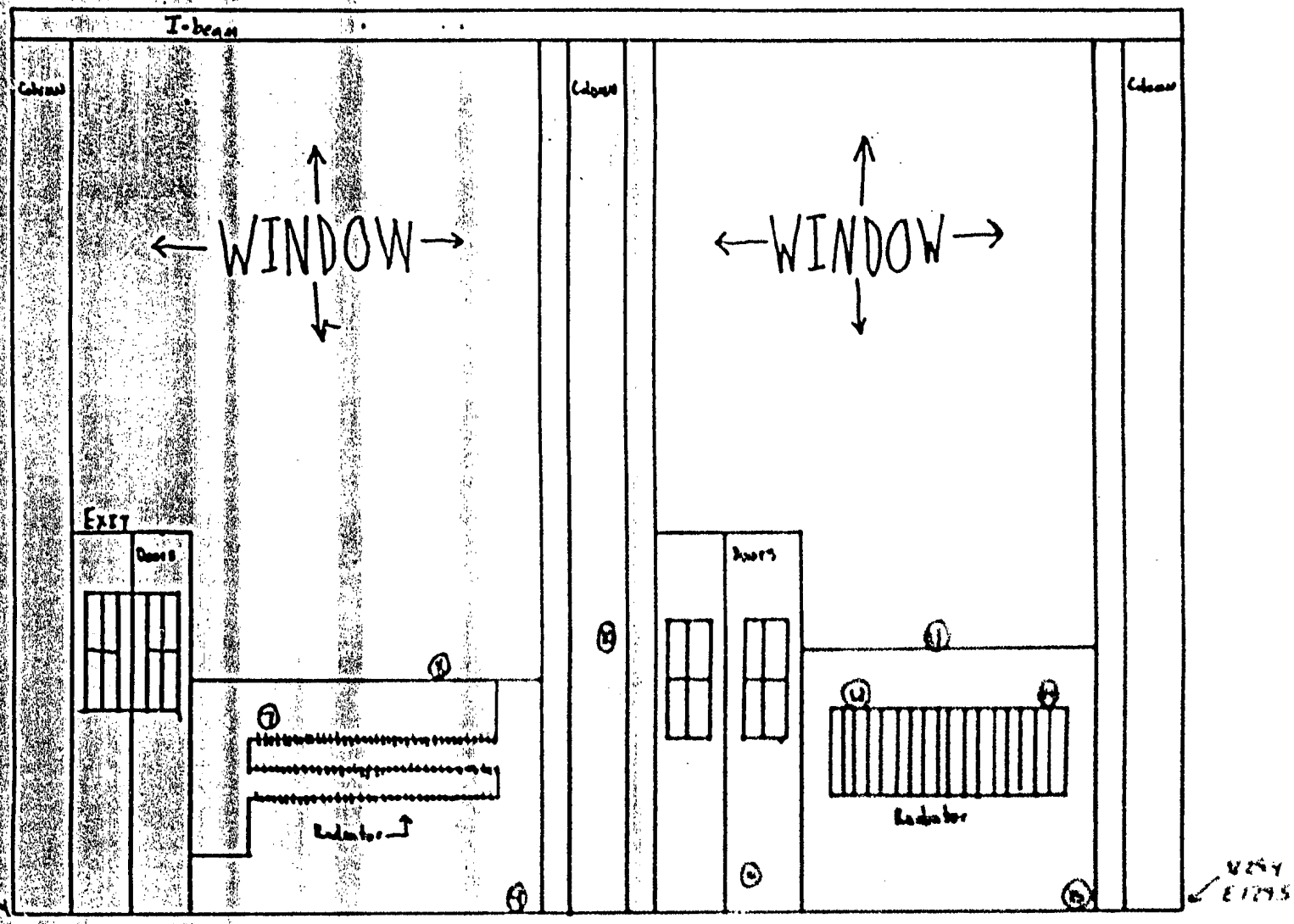
Legend

XXXX = Contamination Area Boundary

= column

129327

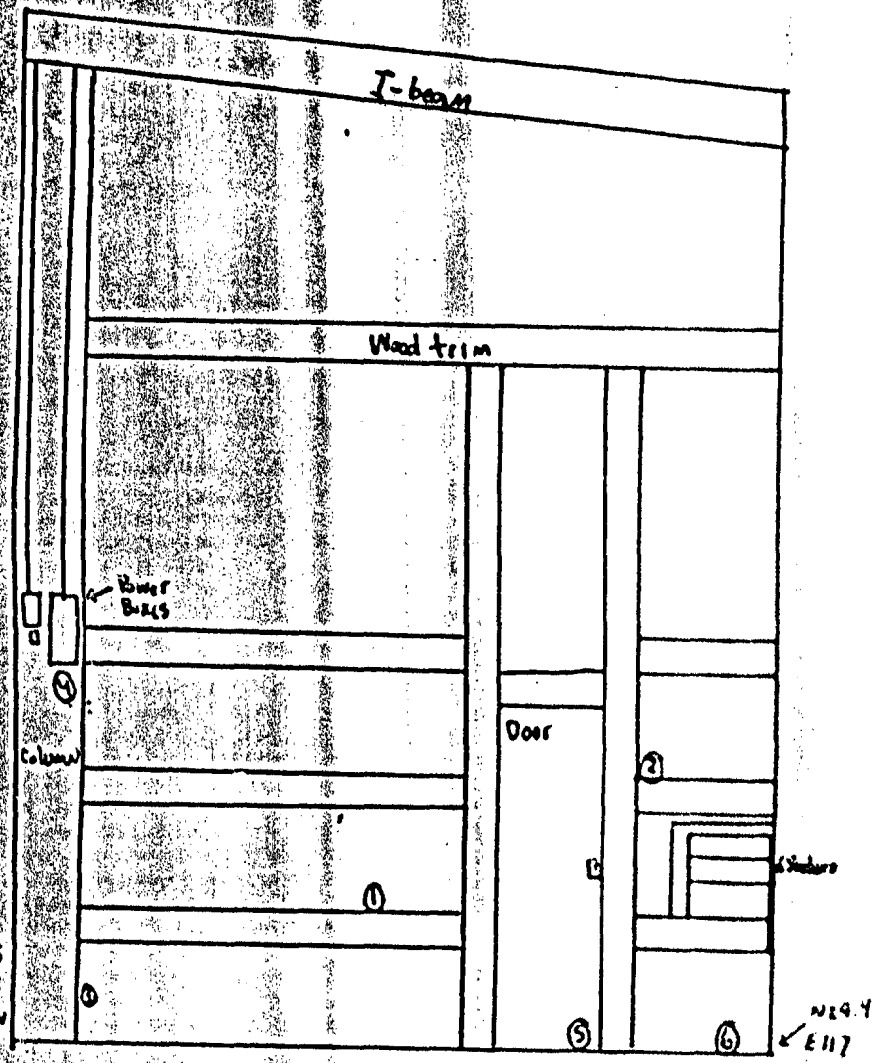
0023



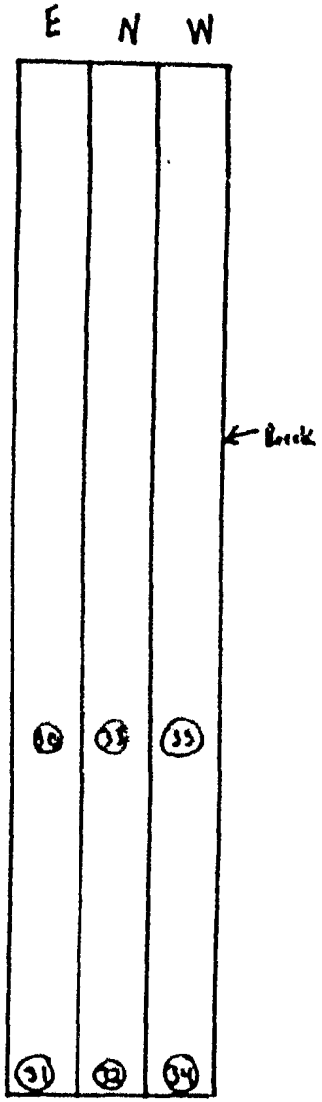
Bldg 23 N. Wall
Room 1A25

119

LAS 661



Bldg 23
W. Wall Room AA-25



Bldg 23
S. Side Middle Column
Room AA25

19957

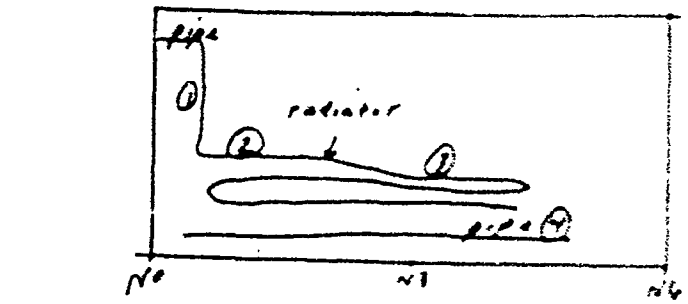
129507

Room B1 Scan Data

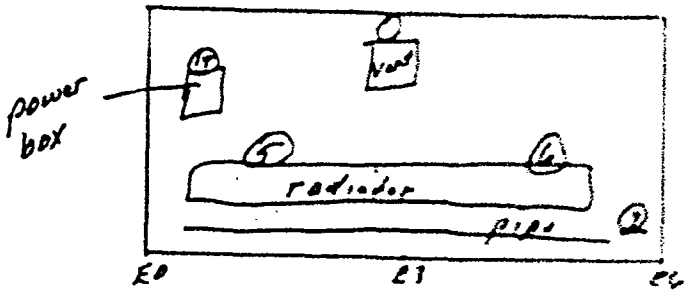
B1.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	327	1959		1267	2	-42
o	2	94	-38				
r	3	-82	154				
i	4	6	614				
z	5	-23	9101		8487	7	13
o	6	-23	9332		7873	2	0
n	7	-82	998				
t	8	-53	115				
a	9	-82	-499				
l	10	-82	346				
	11	64	-115				
S	12	-82	-38				
u	13	210	1114				
r	14	35	960				
f	15	-23	730				
a	16	35	883				
c	17	123	-38				
e	18	-82	576				
s	19	-82	883				
	20	6	806				
C	1	121	6630	147	5530	2	-46
e	2	147	922	226	862	2	-8
i	3	252	862				
l	4	305	-59				
i	5	68	2081	42	119	7	-13
n	6	226	1189	121	119	13	-4
g	7	95	862	42	59	-1	21
	8	68	238				
	9	200	1368	121	357	5	-21
	10	42	357				
	11	68	416				
	12	121	416				

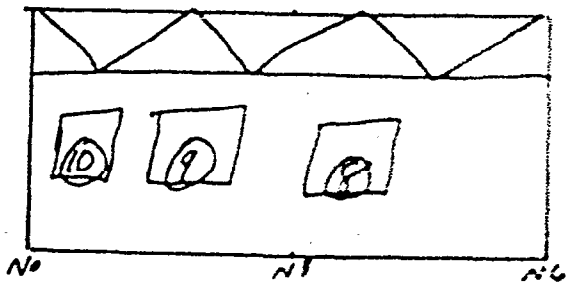
129507



West Wall

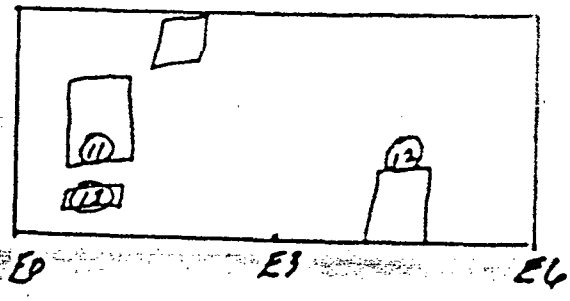


South Wall

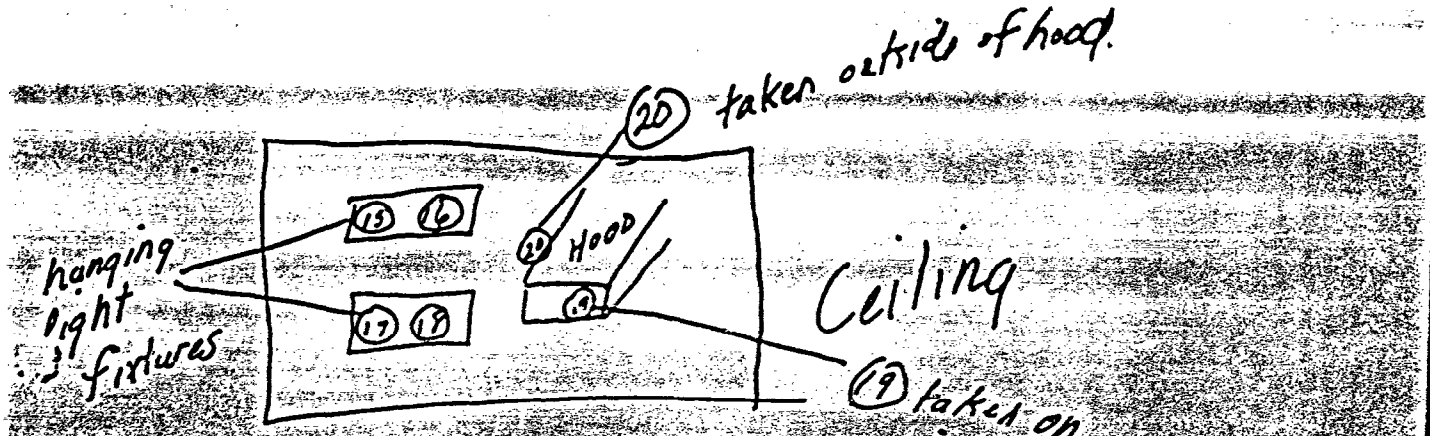


East Wall

Room B1

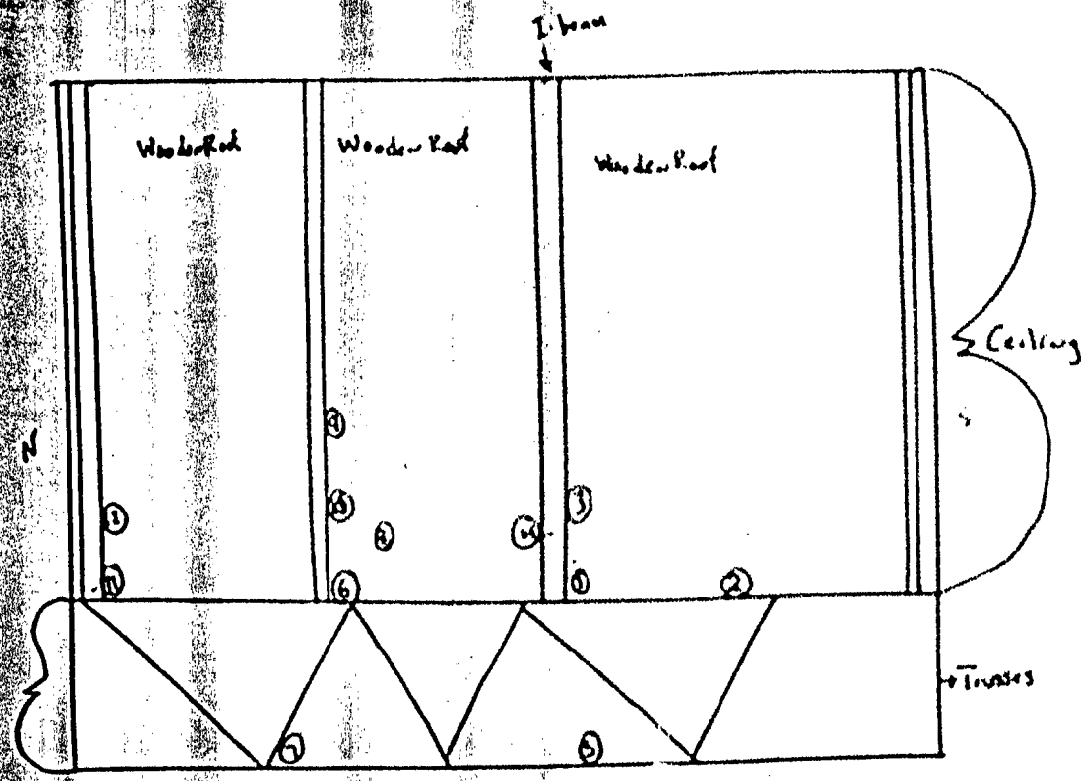


North Wall



Ceiling

0023



Room B1

12956

U-329623

0023

129507

Room B2 Scan Data

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	72	1605			-1	-4
o	2	-4	1011				
r	3	-23	0			-1	-4
i	4	81	832				
z	5	15	6838			2	-13
o	6	-4	1843			5	-17
n	7	62	773				
t	8	34	505				
a	9	6	862				
l	10	44	832				
	11	-13	387				
S	12	15	1011			-1	4
u	13	290	27174			-1	63
r	14	-13	298				
f	15	72	892				
a	16	-13	892				
c	17	44	743				
e	18	-13	862				
s	19	25	684				
	20	44	505				
	21	-13	1070			-1	-33
	22	25	1011			2	-48
	23	53	1189			-1	0
	24	25	862				
	25	81	654				
	26	-4	1189			-1	-39
	27	-13	951			-1	8
	28	252	15787			-1	-25
	29	72	476				
	30	34	-59				
	31	34	0				
	32	6	89				
	33	44	59				
	34	-13	30				
	35	44	-208				
	36	15	268				
	37	6	-89				
	38	-23	59				
	39	-4	0				
	40	62	-238				
C.	1	-23	892				
e	2		21168				
i	3	15	714				
l	4	62	3181			-1	-21
l	5	15	476				
n	6	119	3568			-1	-25
g	7	-4	297				

129507

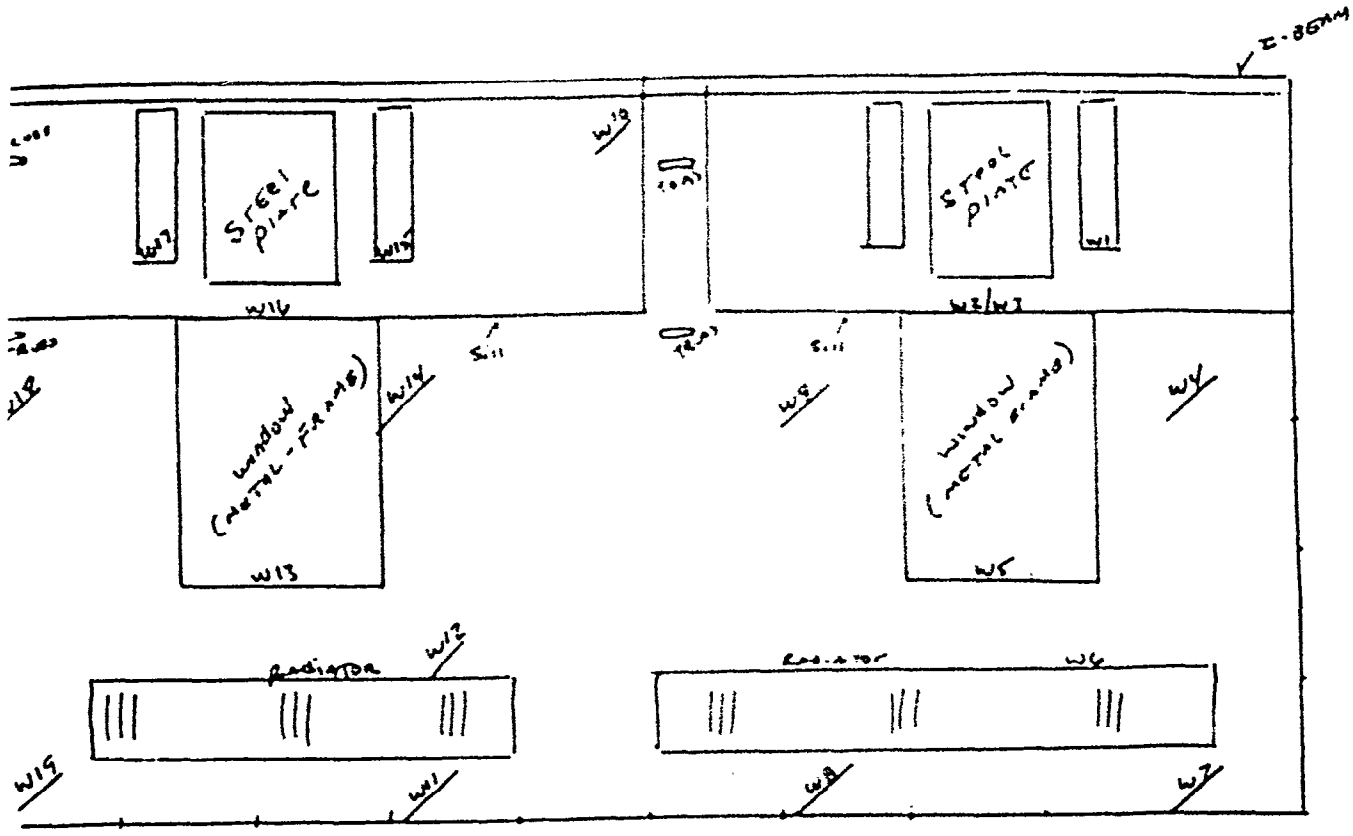
Room B2 Scan Data

B2.XLS		Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
	Location	Before Dust Removal		After Dust Removal			
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
C	8	-4	357				
e	9	6	357				
l	10	44	476				
l	11	25	4757				
l	12	-13	476				
n	13	271	6622		9573	-1	-17
g	14	34	-446				
	15	44	327				
	16	25	535				
	17	15	743				
	18	6	654				
	19	91	951			5	-17
	20	15	387				
	21	53	1130				
	22	-4					
	23	44					
	24						
	25						
	26						
	27	25					

0023

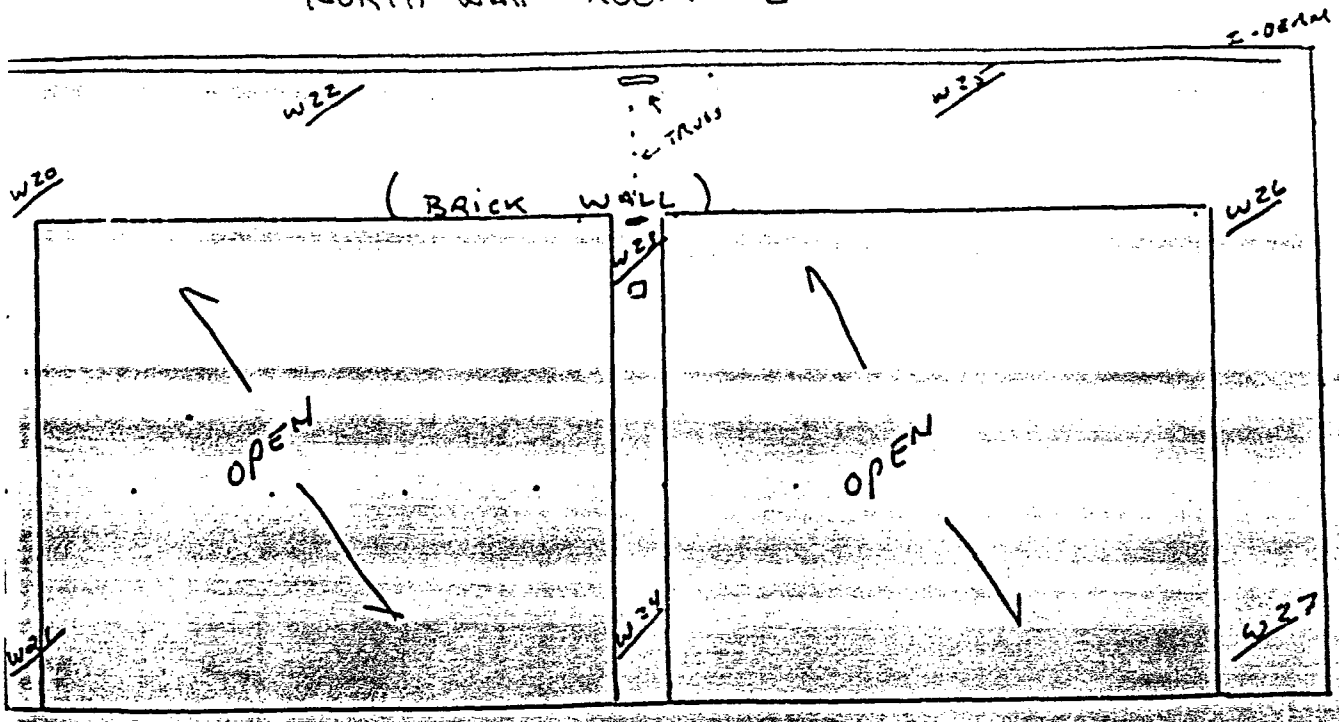
Survey # 133-DT-043
129507

SOUTH WALL - ROOM B-2



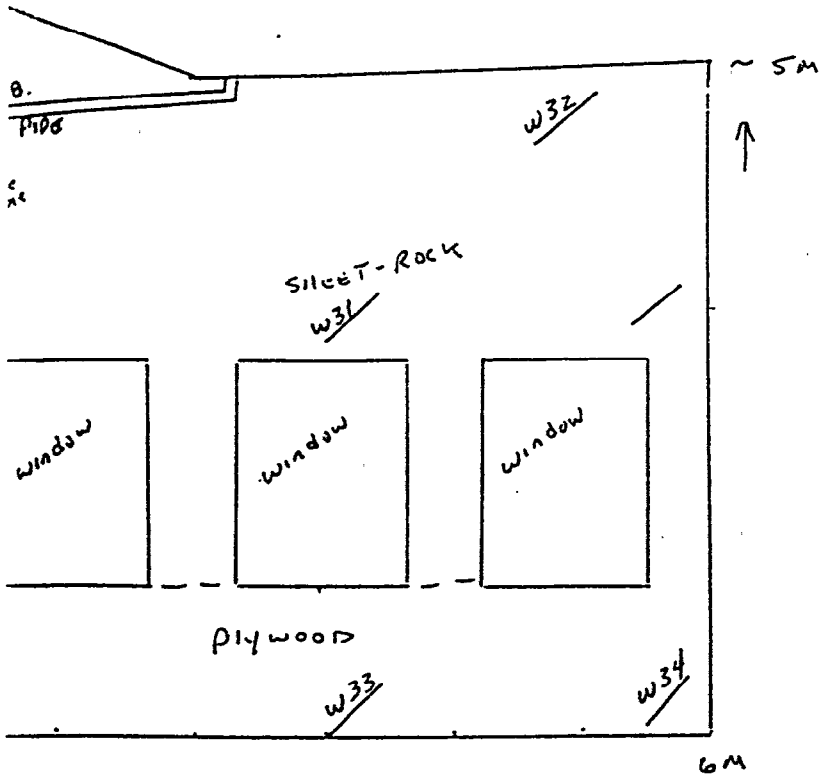
10M

NORTH WALL - ROOM # 2

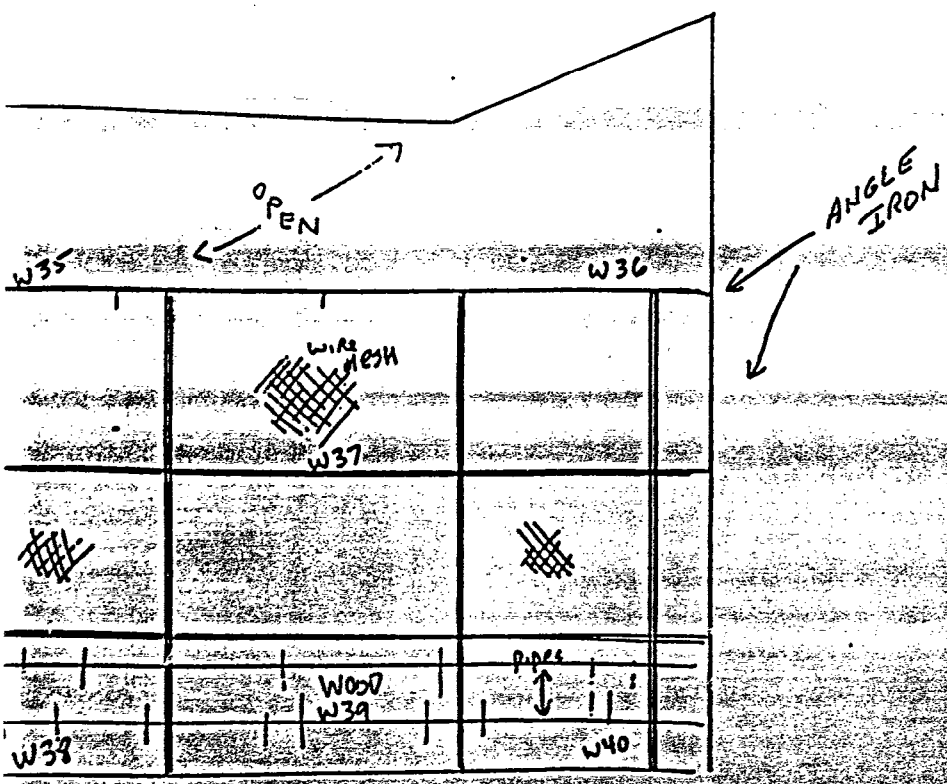


Survey # 133 DT 045
129507
D-19646

WEST WALL - ROOM #B2

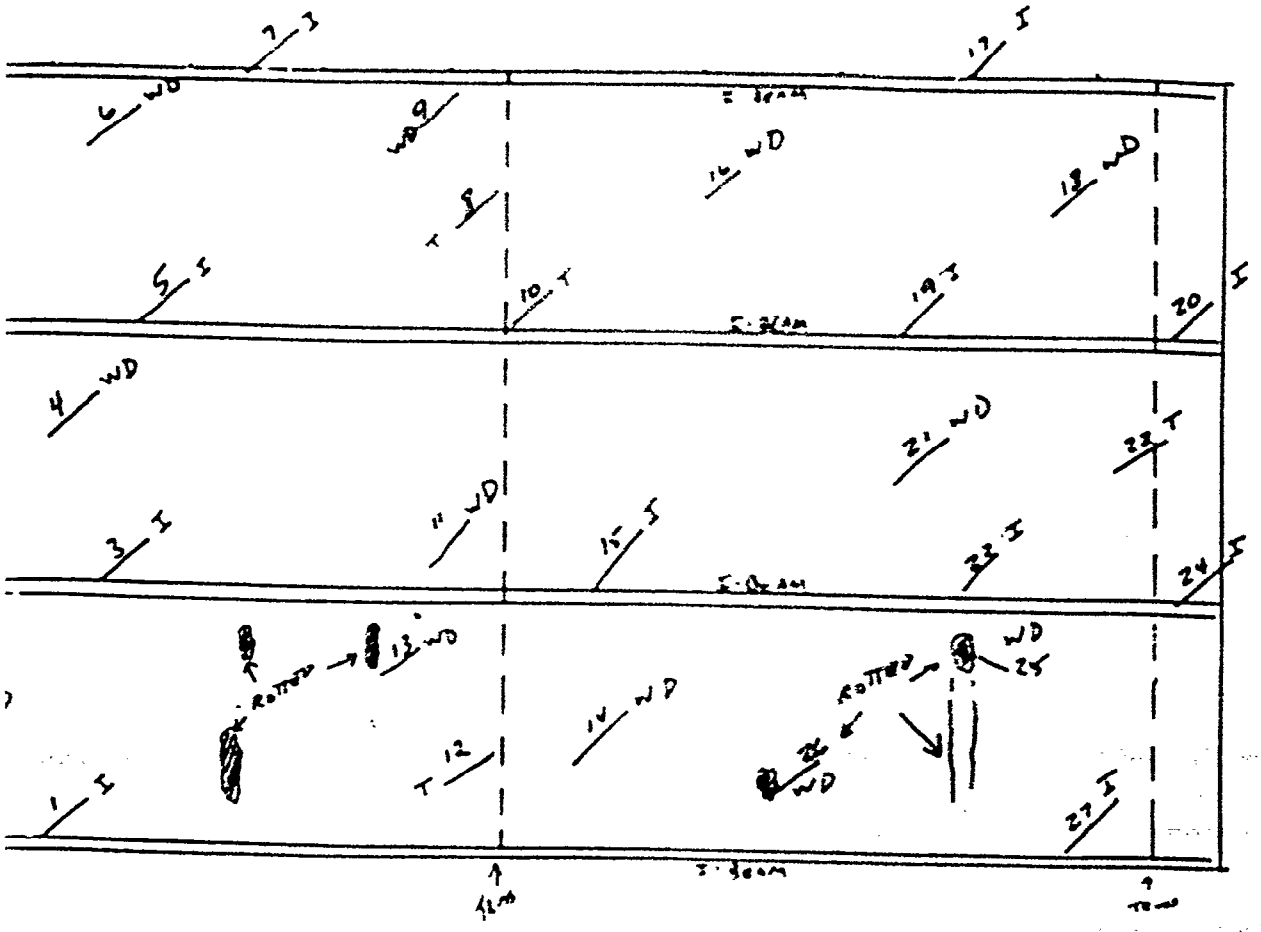


EAST WALL



Survey #
133-DT-043 129507

Ceiling - Room #B2
(VIEW FROM ABOVE)



I = IBEAM/TOP
WD = WOOD
T = TRUSS/TOP

129507

Room B4 Scan Data

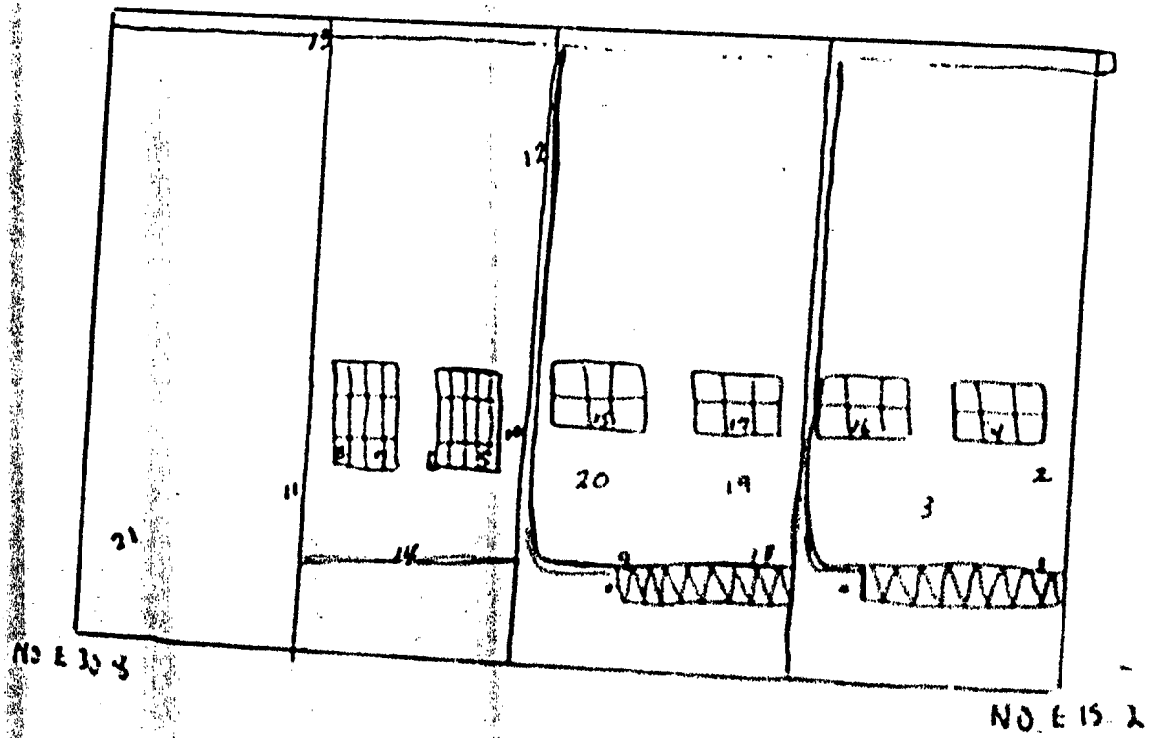
B4JD.3		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
H	1	-99	357				
e	2	115	1130	-99	832	2	-17
r	3	141	1308	115	1160	2	-21
i	4	88	7165	88	7076	5	13
z	5	-19	1427	-45	714	0	8
o	6	-99	892				
n	7	-45	773				
t	8	115	1188	8	951	5	-8
a	9	-19	1070	-72	773	5	-50
l	10	8	1011				
	11	-72	1011				
S	12	-45	714				
u	13	141	565				
r	14	-19	773				
f	15	88	1903	41	3895	5	-21
a	16	-19	2468	-72	2708	5	8
c	17	-72	3360	-99	2587	0	25
e	18	-99	505				
s	19	8	1219	-45	773	0	-4
	20	-99	2111	-99	1487	2	-46
	21	-45	1011				
	22	-19	476				
	23	-19	832				
	24	-45	1130	-72	714	2	-54
	25	248	1219	195	535	2	-13
	26	88	714				
	27	-45	46				
	28	275	1189	141	1665	0	-21
	29	-99	0				
	30	-72	-238				
	31	8	1011	-19	1011	0	46
	32	-99	297				
	33	-99	357				
	34	-45	-119				
	35	8	1487	-19	1249	5	-33
	36	275	-178				
	37	-99	-59				
	38	-99	178				
	39	-99	119				
	40	8	416				
	41	-99	-59				
	42	-72	-297				
	43	-45	238				
	44	-72	238				
	45	-19	357				
	46	-72	178				
	47	-99	714				

129501

Room B4 Scan Data

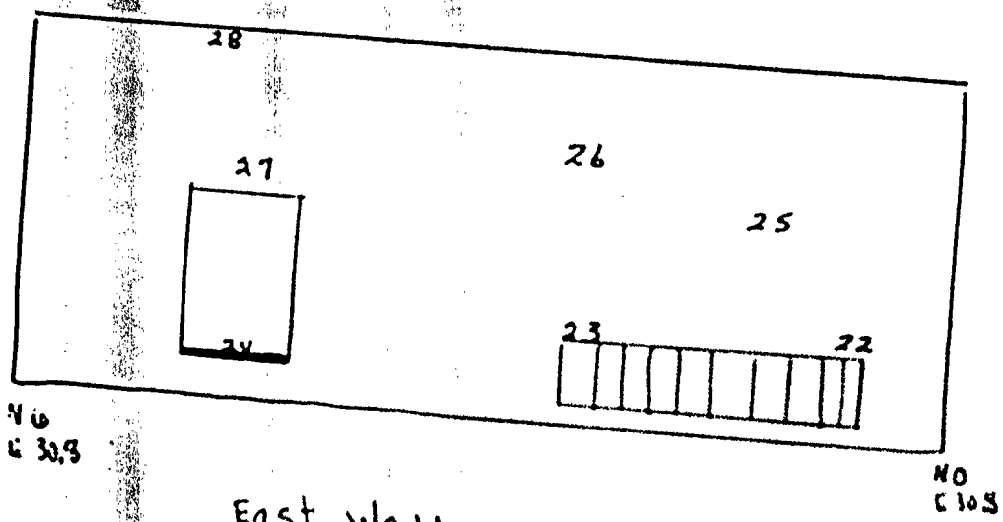
B4.XLS		Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
	Location	Before Dust Removal		After Dust Removal			
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
	48	-72	238				
	49	-99	684				
C	50	-19	3270	[illegible]	4400	5	17
e	51	-19	9560	-72	[illegible]	[illegible]	[illegible]
l	52	-45	[illegible]	-72	[illegible]	[illegible]	[illegible]
l	53	-19	[illegible]	-45	[illegible]	[illegible]	[illegible]
l	54	8	5054	-72	[illegible]	[illegible]	[illegible]
n	55	-99	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
g	56	115	[illegible]	-45	[illegible]	[illegible]	[illegible]
	57	61	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	58	115	[illegible]	-45	[illegible]	[illegible]	[illegible]
	59	275	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	60	61	4490	61	[illegible]	[illegible]	[illegible]
	61	[illegible]	[illegible]	61	[illegible]	[illegible]	[illegible]

Room B4 South Wall



* denotes Radiator

0-13623
131
139527
0023

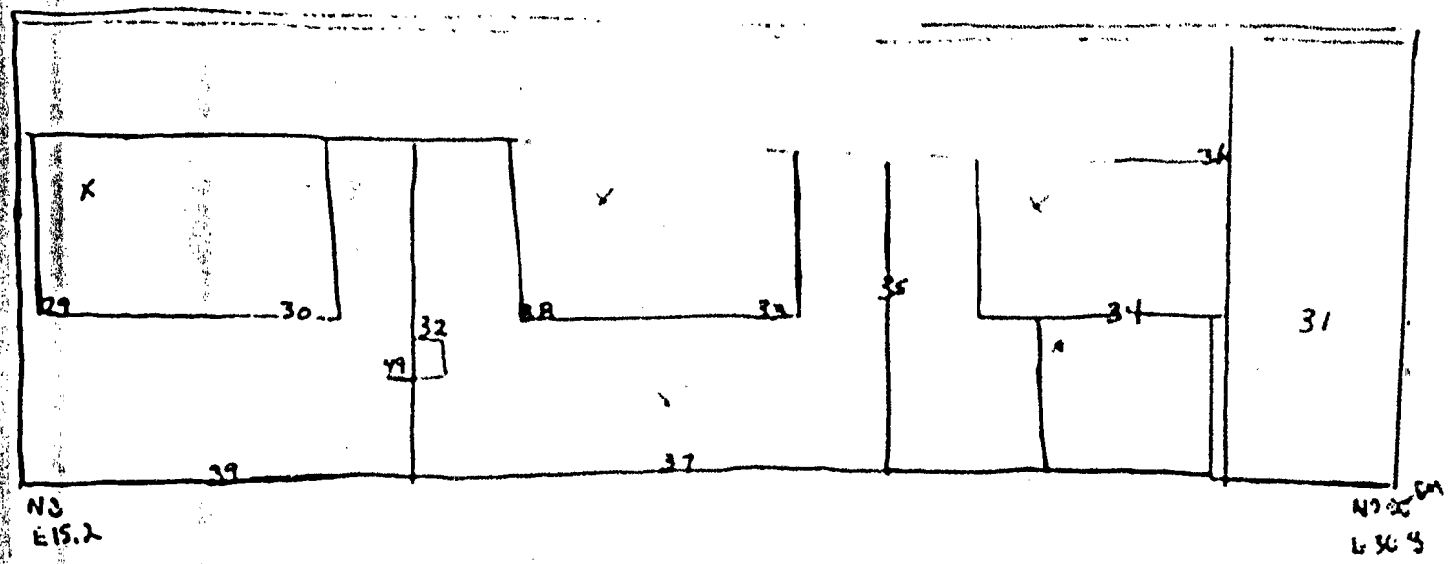


East Wall

Room B4

132

0023



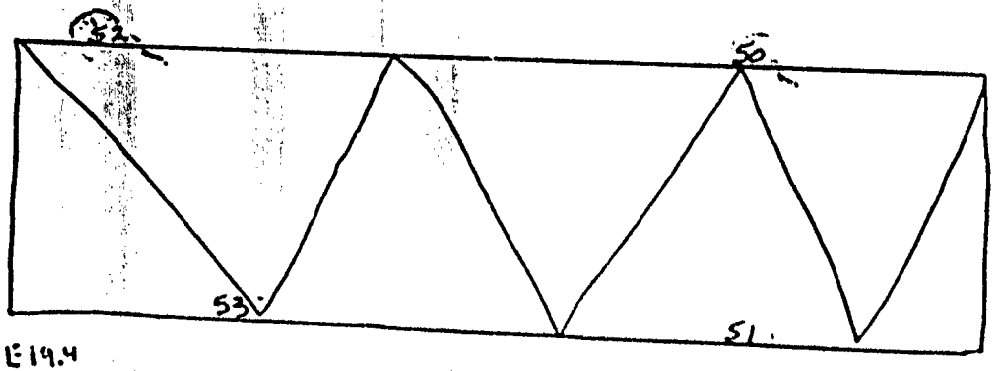
NO WALL Room 04

X denotes Wire Mesh

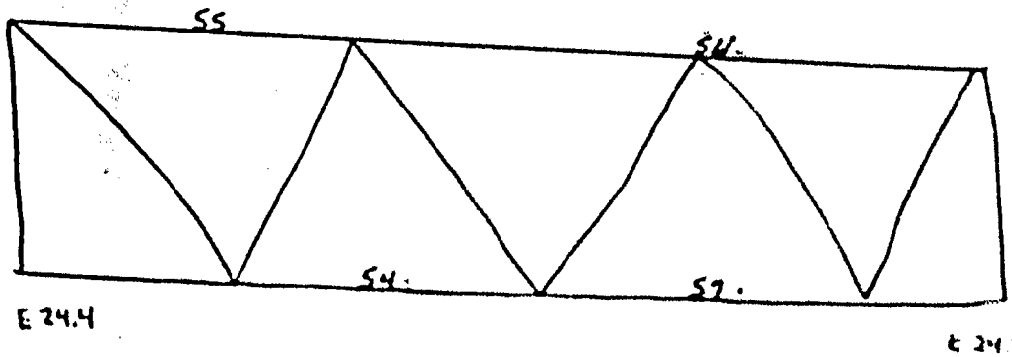
* door

12956-D
133
12956-7
0023

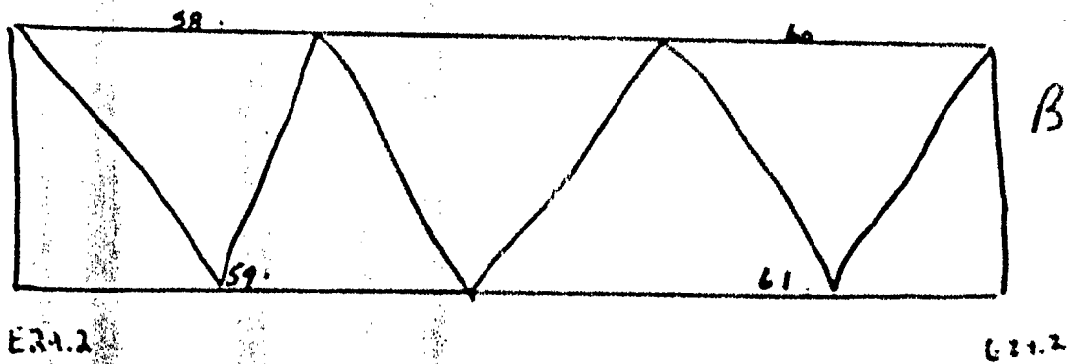
Room B4 Ceiling



B5 Truss



B6 Truss

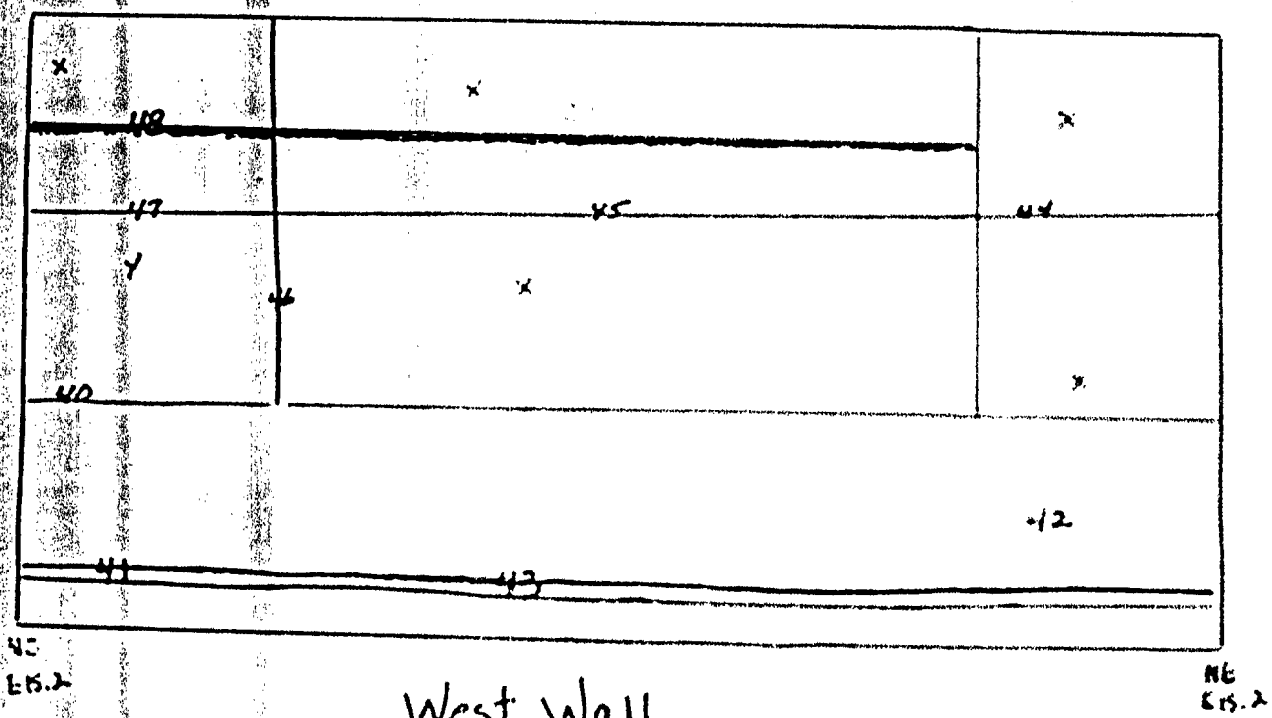


B7 Truss

D-18623

0023

Room B4



West Wall

X denotes wire mesh

D-1596-2
135
129527

129507

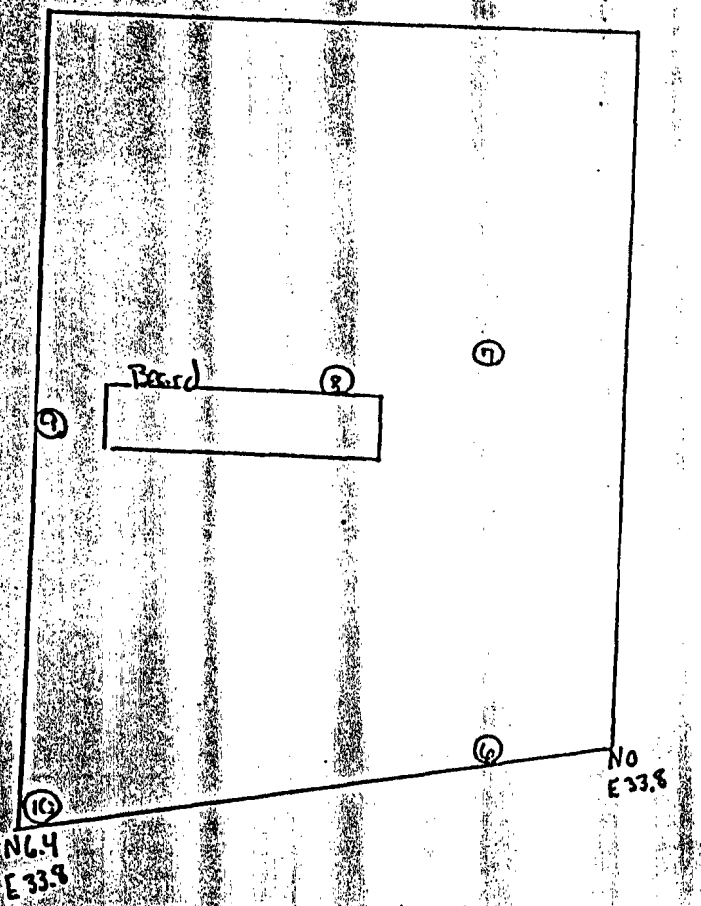
Room B7 Scan Data

B7.XLS

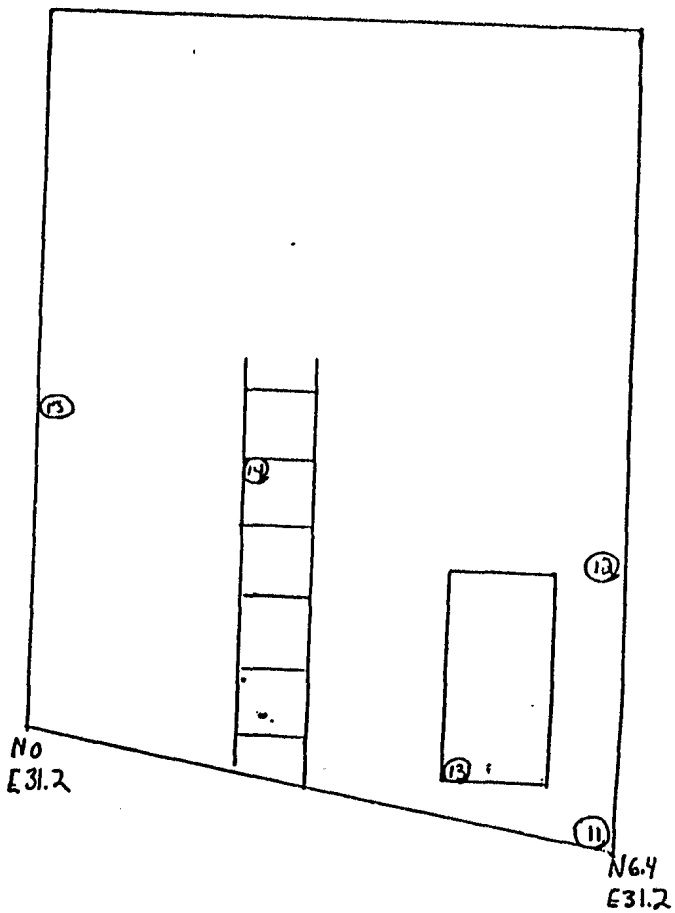
Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	6	2111	50	2269	5	-60
o	2	-9	387				
r	3	50	1130	-38	1427	5	-21
i	4	254	-327				
z	5	240	149				
o	6	-9	1546	6	1519	2	0
n	7	123	1784	20	1427	-1	-21
t	8	64	1814	-9	1511	-1	-13
a	9	283	1100	254	1533	-1	-17
l	10	50	2022	20	1160	-1	-38
	11	268	2170	20	2054	7	8
S	12	327	1248	400	1339	-1	-4
u	13	35	3231	6	6511	-1	-25
r	14	259	1219	181	862	-1	-13
f	15	210	1397	283	951	7	4
a							
c							
e							
s							

Room B-7

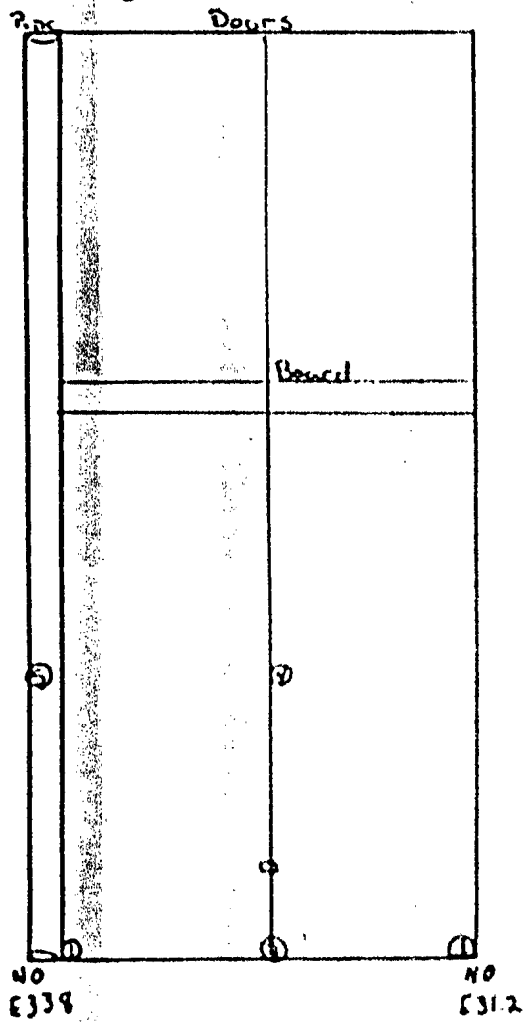
East Wall



West Wall



South Wall Room B-1



129507

129507

Room B8 Scan Data

B8.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	106	1593			2	8
e	2	75	6209			0	-8
r	3	35	135				
l	4	14	297			2	13
z	5	85	1350			2	-8
o	6	95	1728			2	67
n	7	-6	621				
t	8	45	810			2	4
a	9	14	1512			2	0
l	10	55	1350			2	-8
	11	116	1026			5	4
S	12	55	891				
u	13	-8	1539			0	-29
r	14	106	2187			0	-17
f	15	4	837				
a	16	-8	837				
c	17	-6	918			2	-4
e	18	116	594			2	-59
s	19	14	972			0	13
	20	24	1134			0	-4
	21	24	675			0	-25
	22	106	1404			0	29
	23	85	2160			0	0
	24	-6	1107			0	8
	25	4	756				
	26	45	378			0	-17
	27	75	1134			0	-17
	28	85	648				
	29	24	1080			0	-34
	30	177	783				
	31	65	621			8	0
	32	197	1242			0	17
	33		5129			2	29
	34		5264			8	71
	35		2564			0	0
C	1	95	1011			0	-8
e	2	85	5530			0	-21
l	3	170	684				
l	4	66	476				
l	5	57	3003			8	71
n-	6	95	2854			5	8
g	7	57	446				
	8	0	862			0	4
	9	47	1992			0	13
	10		3241		142	5	0
	11	66	1070			0	-4
	12	19	4252			0	8

129507

Room B8 Scan Data

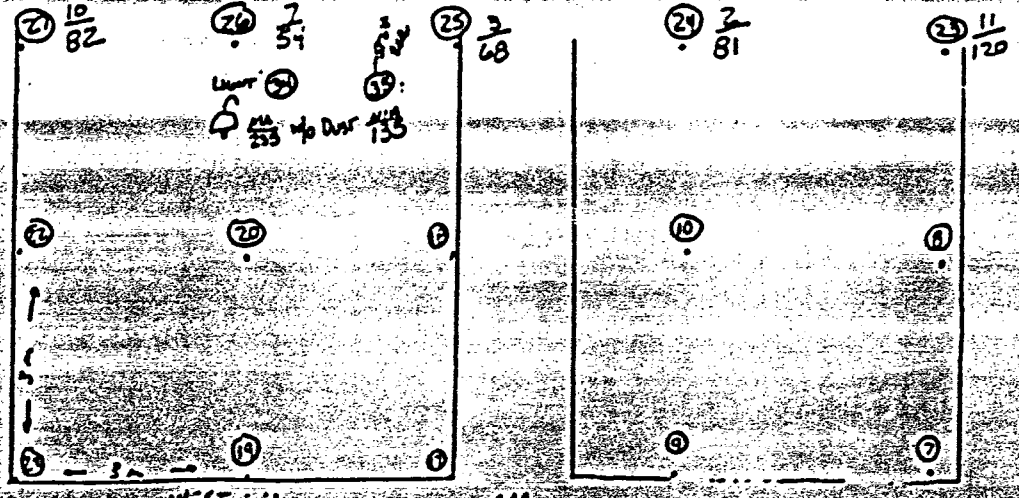
B8X1.5		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
C	13		6836			5	-4
e	14	47	1330			5	4
l	15	19	4876	0	862	11	46
l	16		3300			0	-21
i	17	246	4468			8	29
n	18		8285			11	-13
g	19	189	3360			19	8
	20	170	684			14	-8
	21	57	2319			2	-29

0023

Warm Up/Decon Rm Wall Survey

D-19394

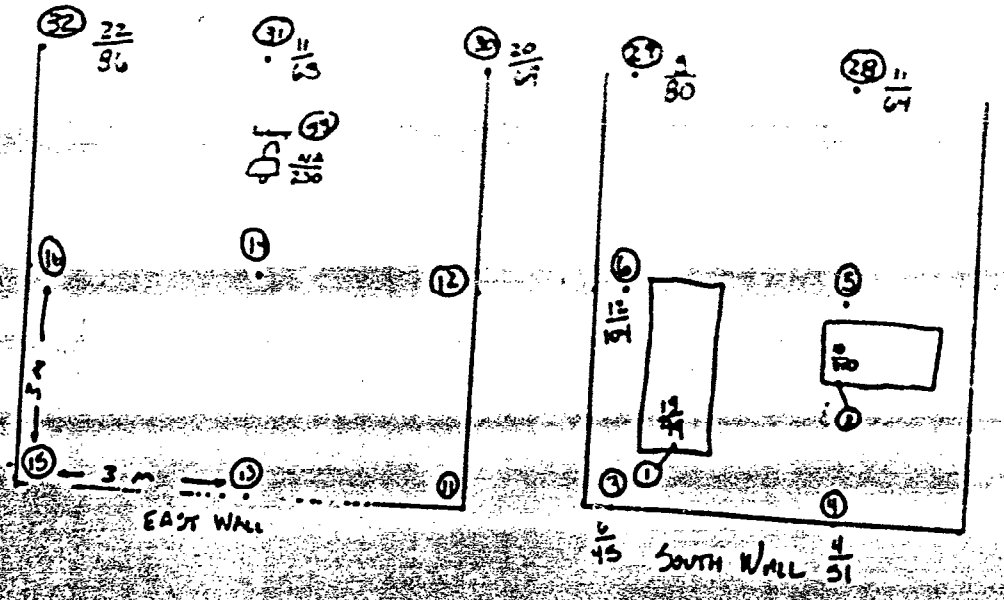
129507



Warm Up/Decon Rm Wall Survey

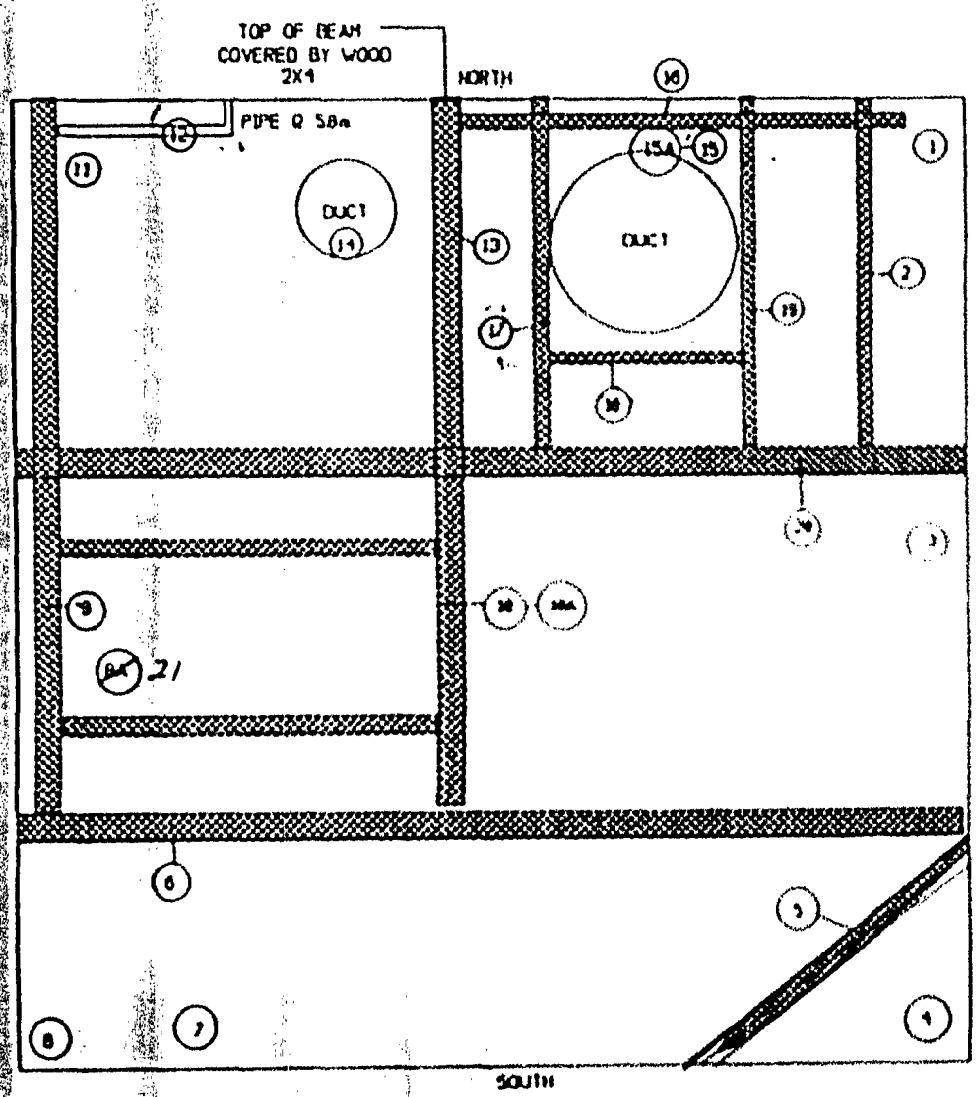
12950

D-19394



CHAPMAN VALVE SITE

WARM UP/DECON RM
BLDG #23
CEILING SURVEY

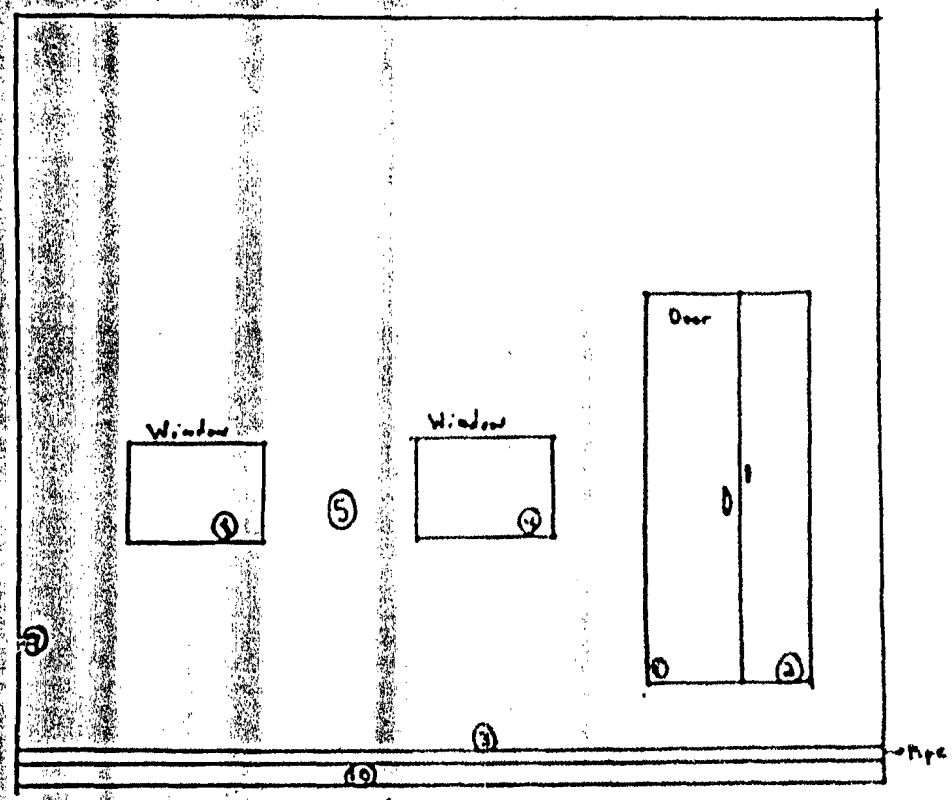


1985-1

129507

Room B10 Scan Data

B10XLS		Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
	Location	Before Dust Removal		After Dust Removal			
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
H	1	3	416				
e	2	16	297				
r	3	42	-59				
i	4	121	654				
z	5	68	1437	42	1724	-1	-21
o	8	42	982				
n	9	173	297				
t	10	16	1011	14	595	-1	35
a	11	121	595				
l	12	16	476				
	13	121	297				
S	14	16	178				
u	15	42	238				
r	16	-11	-178				
f	17	42	1011	16	357	2	4
a	18	42	1437	95	-178	-1	13
c	19	147	-238				
e	20	68	178				
s	21	-11	1011	-11	595	-1	-33
	22	16	-178				
C	6	-11	-178				
e	7	3	-59				
i							
l							
i							
n							
g							



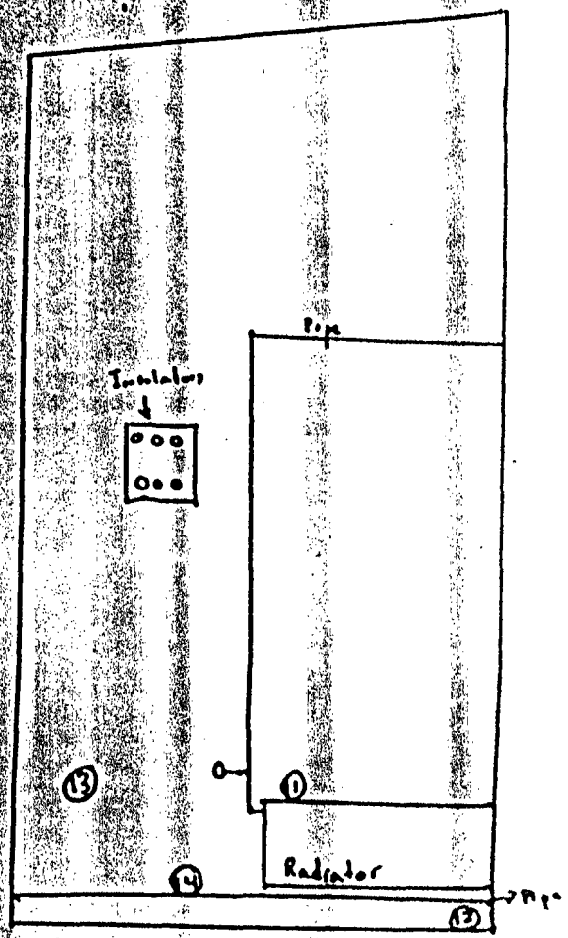
S. Wall Room BH 11/29/94
B 10

1 of 57
0-19623

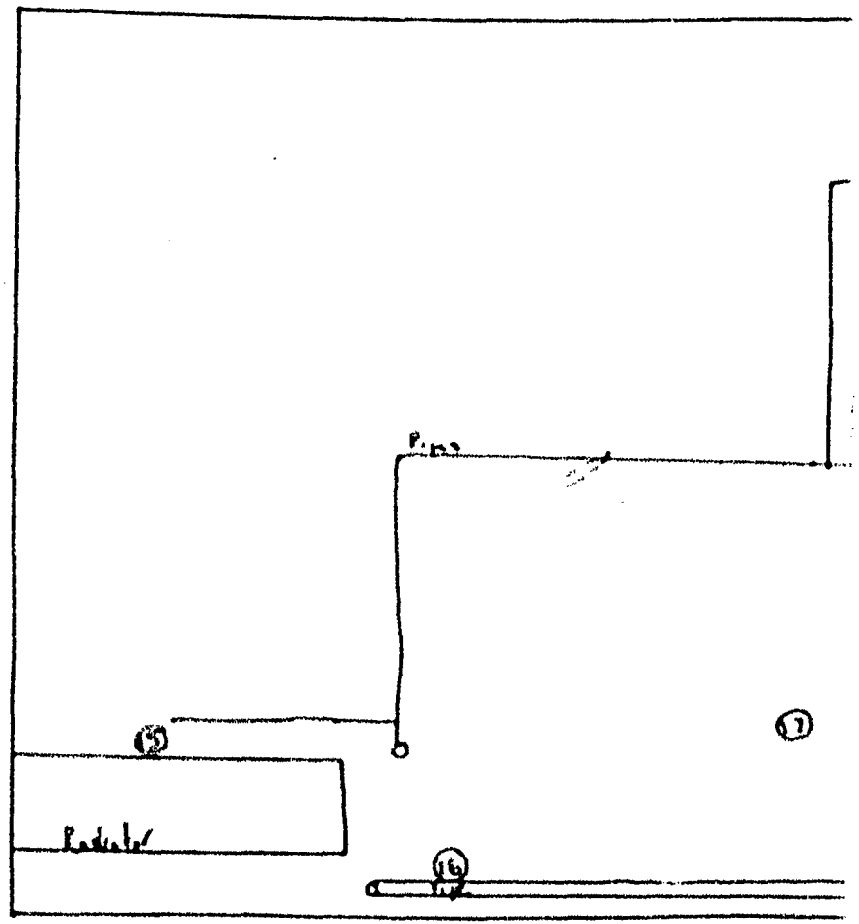
E ~~W~~ Wall - Room ~~811~~ 810

Ceiling - Room ~~811~~ 810

U-5523



W. Wall Room BH B10



N. Wall Room BH B10

147

0-18528

189571

0 0 2 3

129507

Room B11 Scan Data

B11XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	46	829		414	2	-13
o	2	-35	-296				
r	3	-143	1006		621	2	-13
i	4	-89	710		207	-1	13
z	5	-116	1421		592	7	4
o	6	-116	651		769	-1	17
n	7	19	1421		769	5	0
t	8	-143	1184		1065	2	-8
a	21	-22	296				
l	22	-8	266				
	23	-48	385				
S	24	-62	1391		148	2	-50
u	25	-35	533				
r	26	-35	207				
f							
a							
c							
e							
s							
C	9	-62	681		533	2	13
e	10	-116	2811		1450	5	-46
i	11	-116	888		1065	-1	-46
l	12	-62	2456		2486	2	17
l	13	-35	1805		474		
n	14	-22	0				
g	15	288	681				
	16	167	207				
	17	-89	651				
	18	-62	1273		1213	10	-63
	19	-62	207				
	20	-102	503				

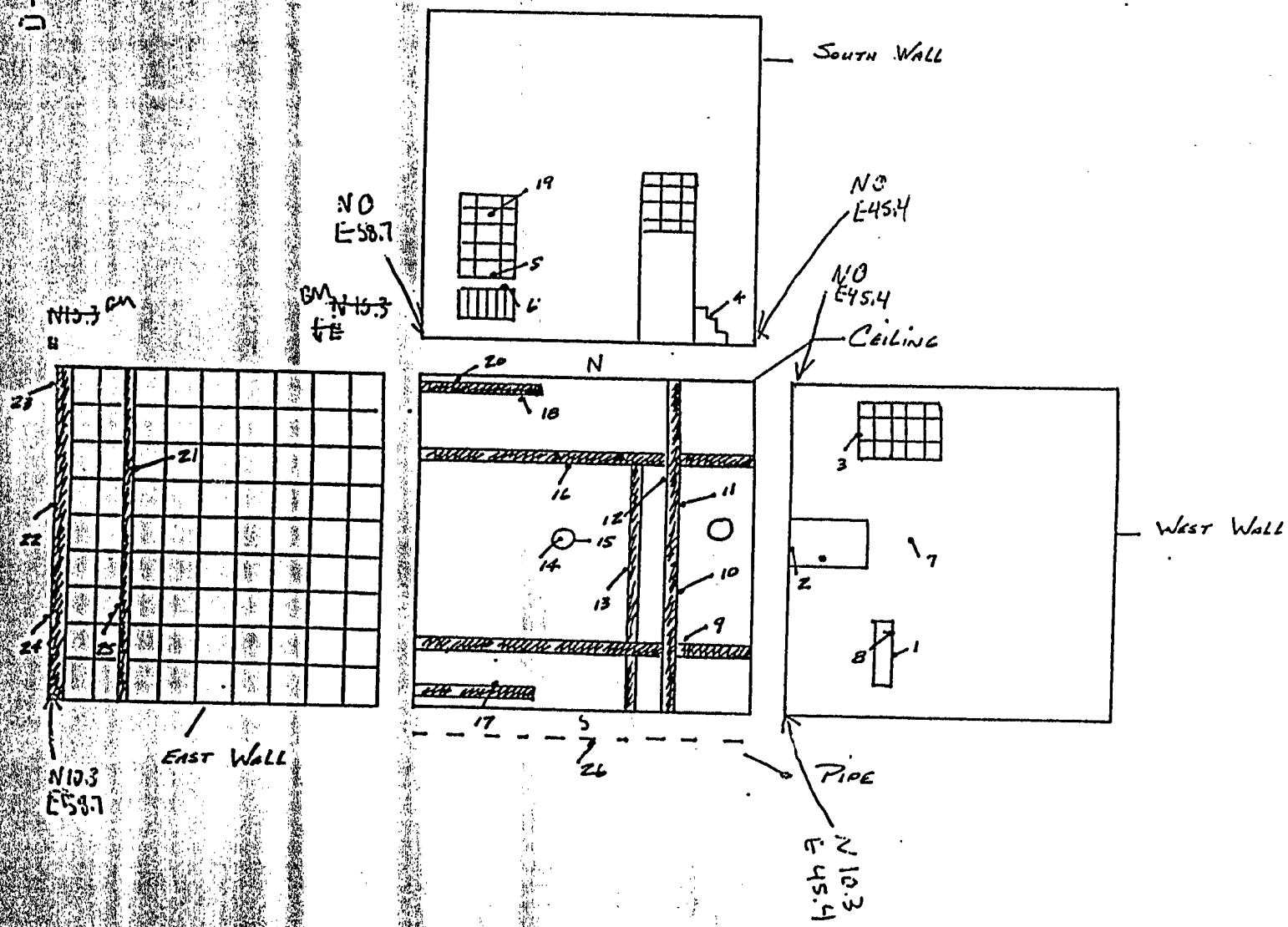
Bldg # 23

RM: ~~B-T5~~ B-11

INTERIOR

SURVEYORS: J. DAVIS
J. ROBINSON

12. 1-94
CHAPMAN VALVE



D-19623

19851

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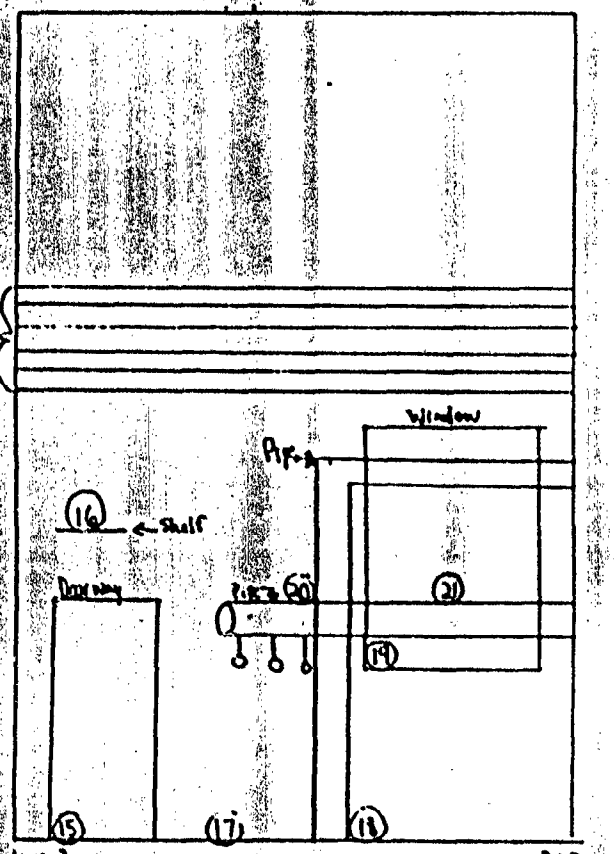
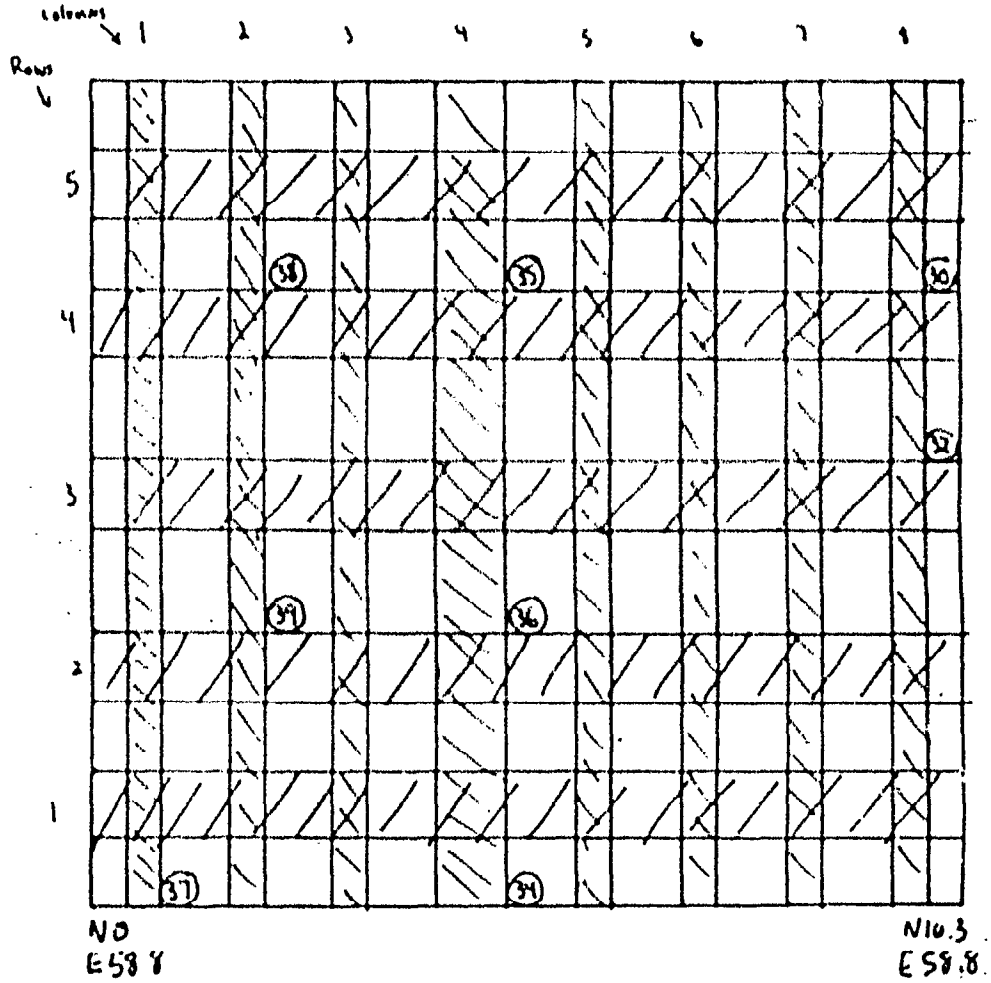
Room B13 Scan Data

B13.XLS		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
H	1	-9	684				
o	2	152	-149				
r	3	-82	89				
i	4	35	654				
z	5	6	297				
o	6	79	1070	-53	1576	0	-38
n	7	-82	595				
t	8	6	-30				
a	9	35	149				
l	10	123	565				
	11	167	-387				
S	12	6	178				
u	13	-23	505				
r	14	64	1576	94	149	8	21
f	15	50	595				
a	16	20	714				
c	17	35	387				
e	18	-53	238				
s	19	35	684				
	20	-9	297				
	21	-23	-178				
	22	64	208				
	23	35	1070	-9	1397	0	-33
	24	20	565				
	25	342	1368	327	1308	0	-13
	26	79	832				
	27	50	149				
	28	20	59				
	29	123	1308	123	714	2	0
	30	108	0				
	31	35	-89				
	32	6	297				
	33	269	-357				
	34	-38	-30				
	35	-67	-178				
	36	-38	-119				
	37	64	-119				
	38	50	59				
	39	50	-178				

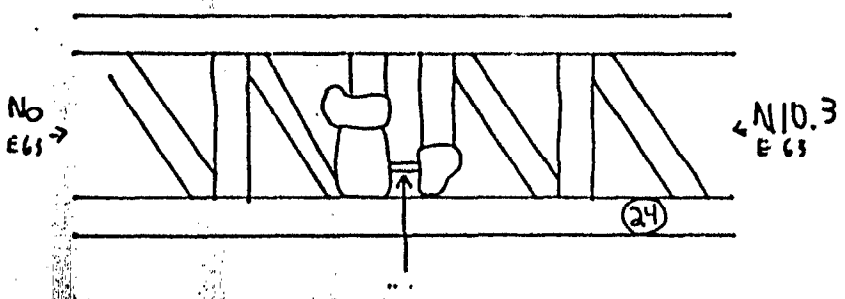
Bld 23 Room B-15 East Wall



Bldg 23 Room B-13 West Wall

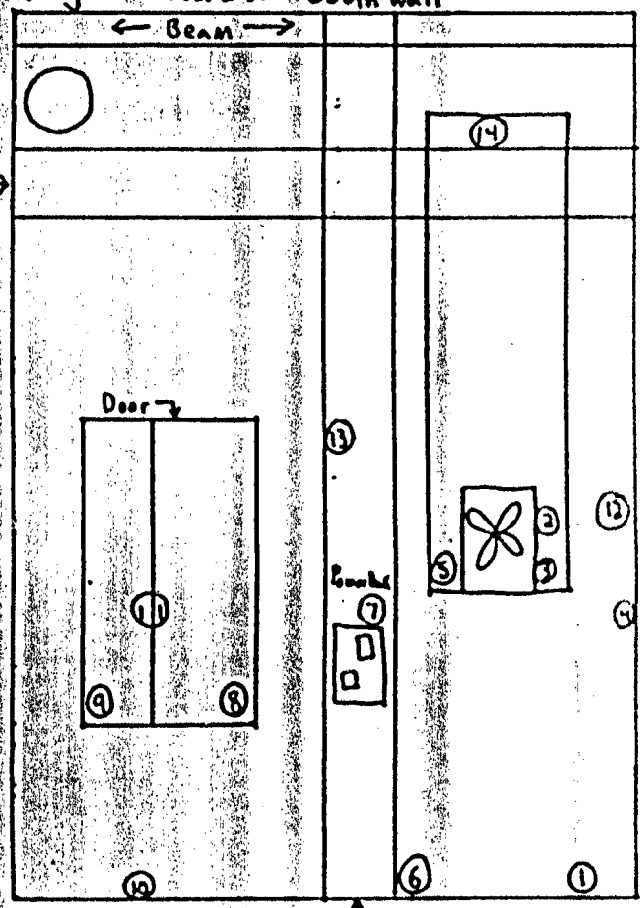


Top view of crane near ceiling in center of room →

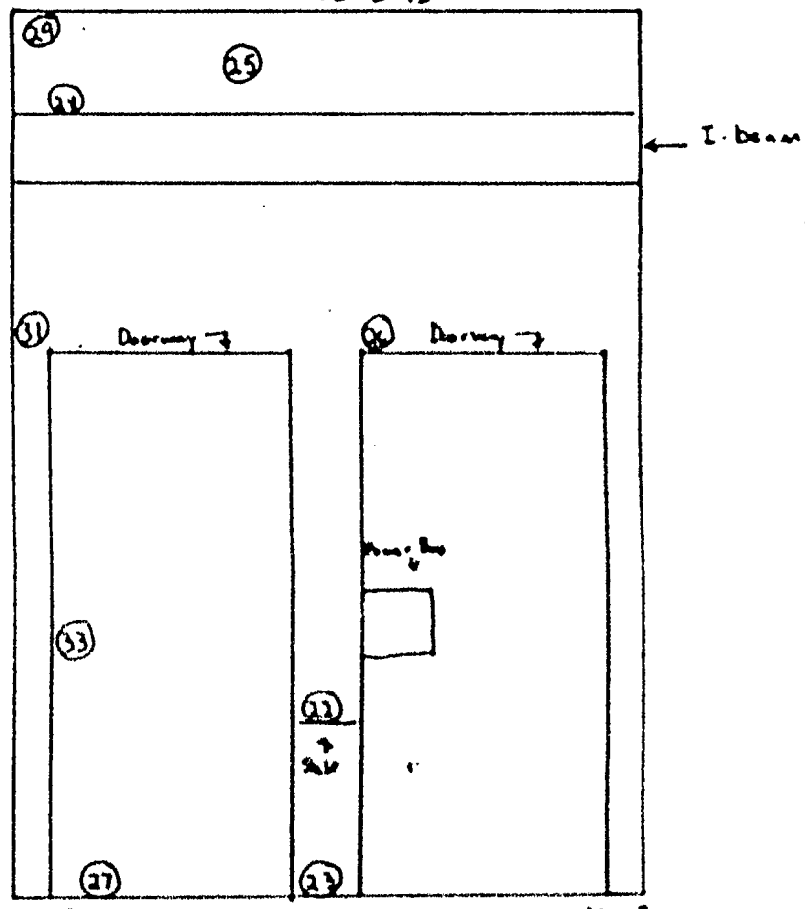


Lasser

Bldg 23 Room ~~B-15~~ B-13 South Wall



Bldg 23 Room ~~B-15~~ B-13 North Wall



129507

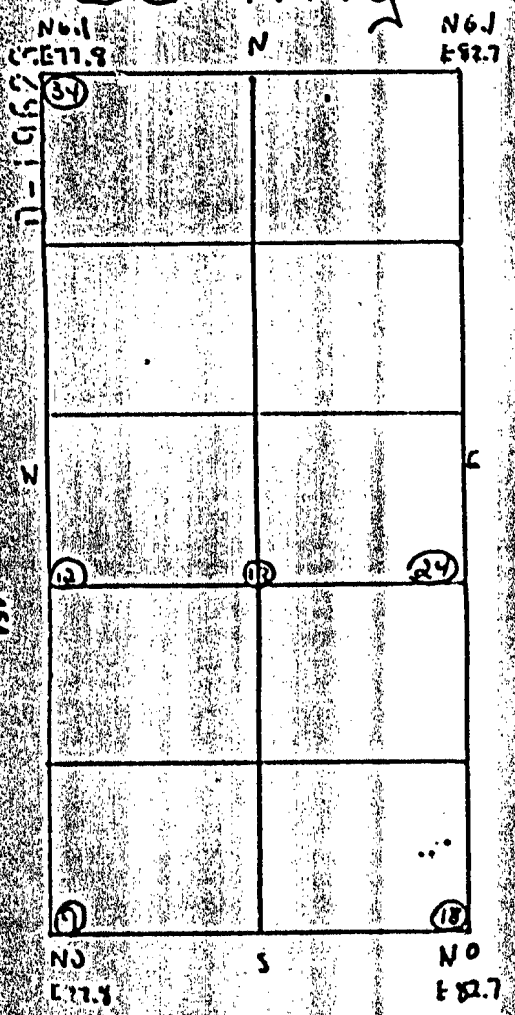
Room B17 Scan Data

B17.XLS

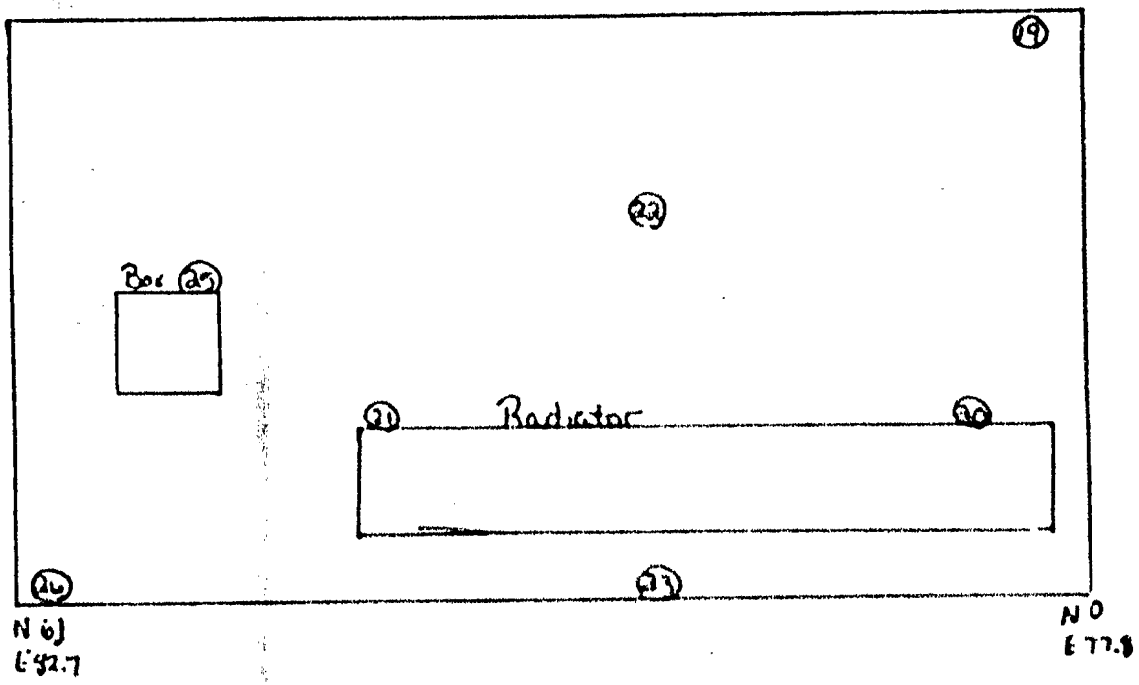
Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
H	1	-20	474				
o	2	184	-297				
r	3	-20	59				
i	4	67	414				
z	5	-35	474				
o	6	23	595				
n	7	23	414				
t	8	23	267				
a	10	23	238				
l	11	4	-149				
	14	9	-238				
S	15	23	178				
u	16	38	743				
r	17	9	0				
f	19	4	327				
a	20	-50	-30				
c	21	-20	-48				
e	22	38	595				
s	23	-35	624				
	25	-50	357				
	26	-4	535				
	27	-4	416				
	28	-50	505				
	29	-35	535				
	30	-64	268				
	31	-79	0				
	32	82	-357				
	33	-20	832				
	35	-20	0				
C	9	53	-238				
e	12	-20	59				
i	13	9	89				
l	18	-50	446				
l	24	9	238				
n	34	67	59				
g							

0023

Ceiling



East Wall Room B-17



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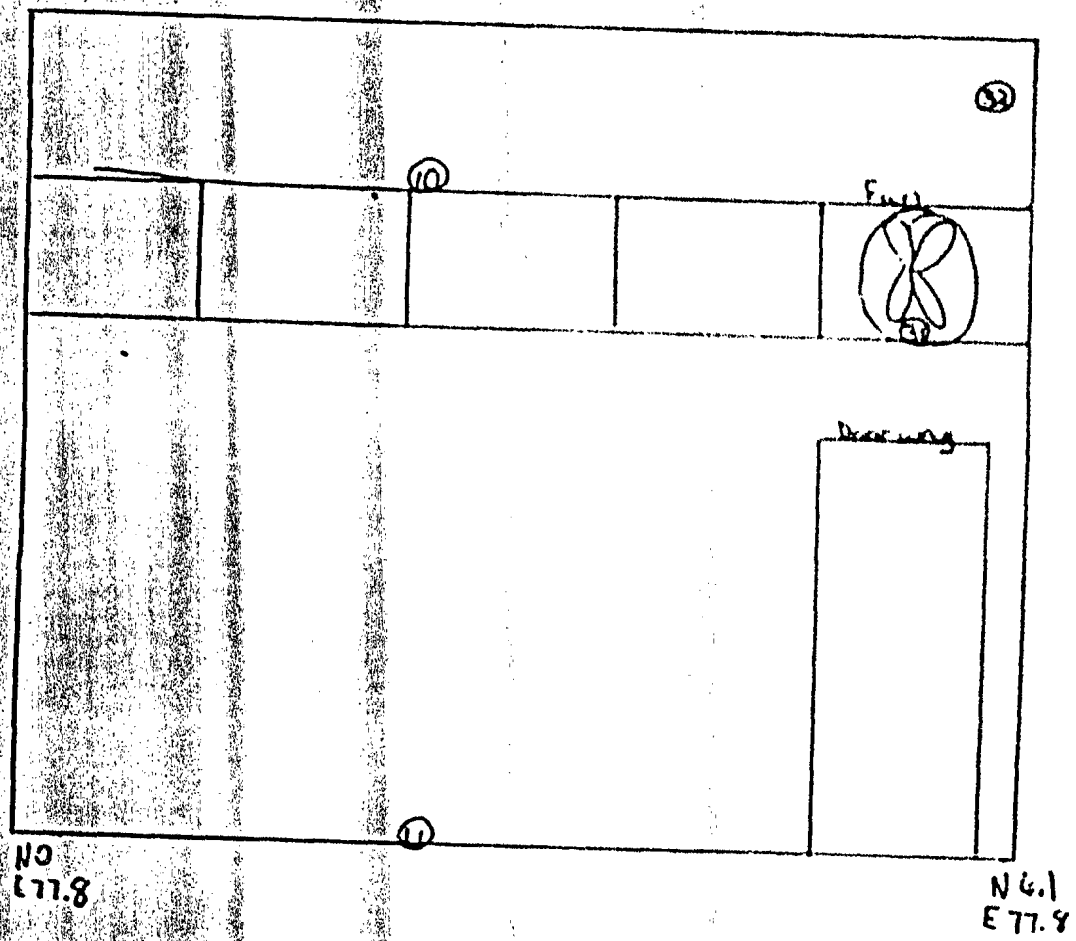
West Wall Room B-17

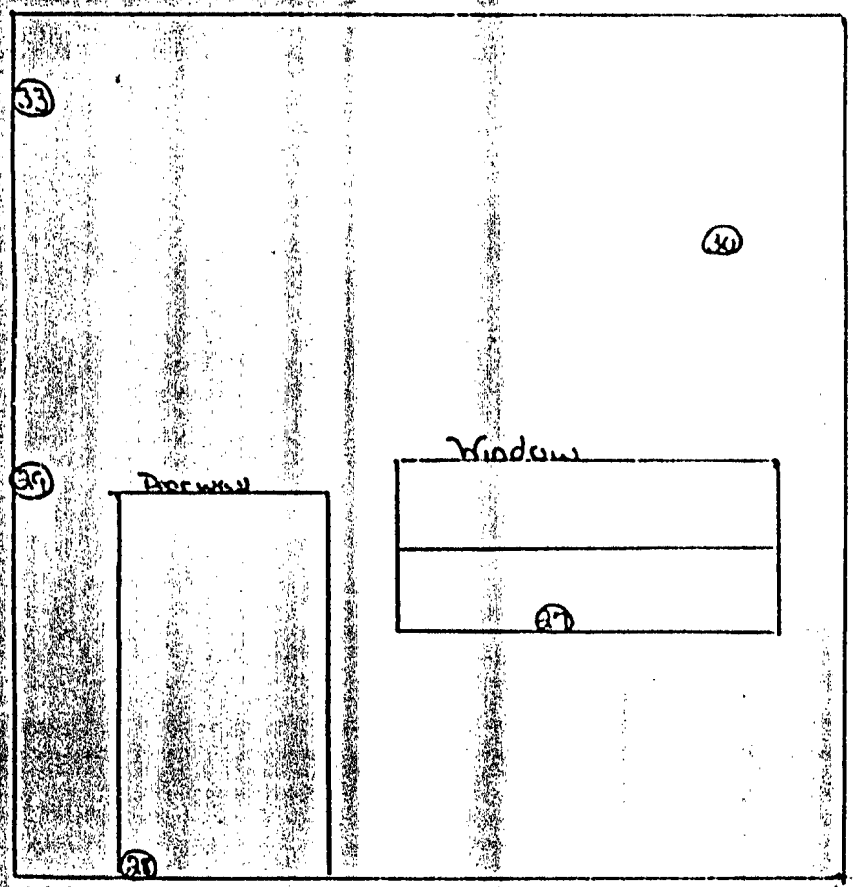


155

1.9950'

North Wall

Room B-17



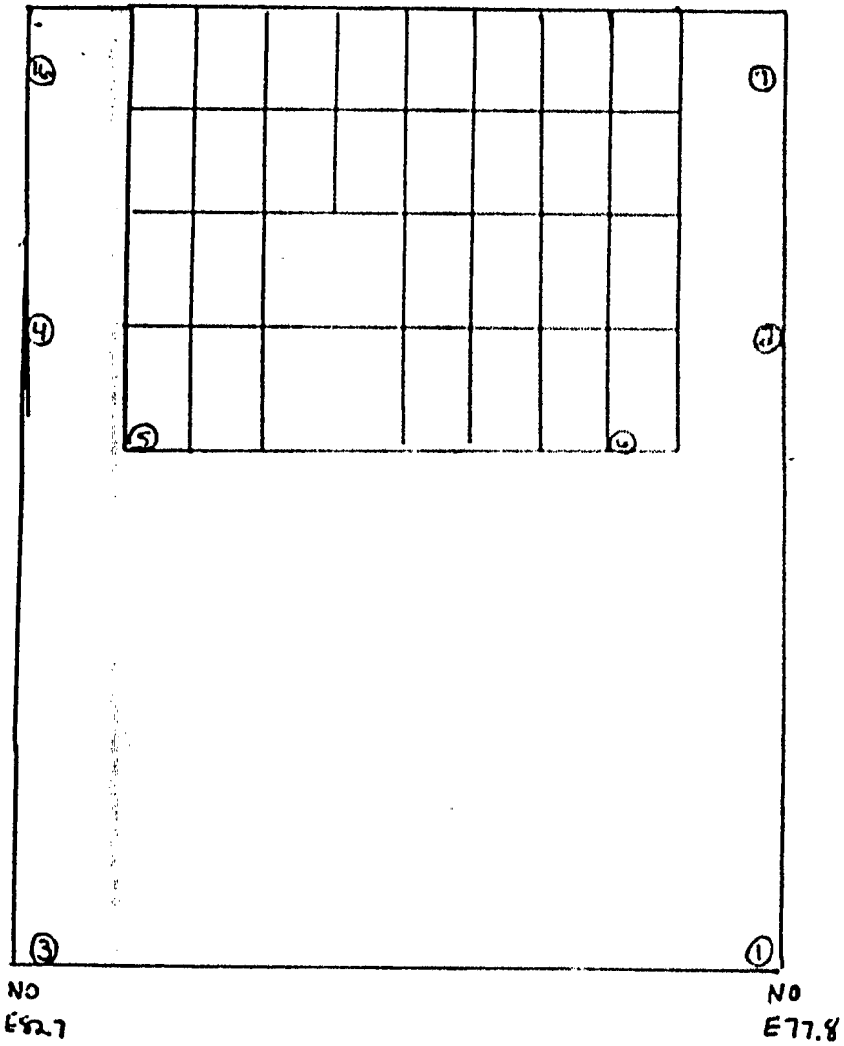
Top View of Light

West Side

East Side

South Wall

Room B-17



U-13623

157

L29577

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ACM Bags Dose Rate Survy Results

ACM-SCAN.XLS	Dose Rate (mR/hr)		
Bag	Beta + Gamm	Gamma	Net Beta
1	0.014	0.016	-0.002
2	0.014	0.024	-0.010
3	0.012	0.012	0.000
4	0.021	0.008	0.013
5	0.014	0.014	0.000
6	0.010	0.014	-0.004
7	0.021	0.013	0.008
8	0.019	0.021	-0.002
9	0.018	0.012	0.006
10	0.014	0.009	0.005
11	0.023	0.022	0.001
12	0.019	0.013	0.006
13	0.021	0.009	0.012
14	0.018	0.014	0.004
15	0.014	0.010	0.004
16	0.012	0.012	0.000

0023

129507

0023

Equipment Scan Data

EQUIP XLS

Location	Equipment	See Note	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable (DPM/100 cm^2) Beta/Gamma
H	Furnace A27-28	LO	168	357		-13
o		LI	8	208	-1	33
r		RO	-19	2464	-1	33
i		RI	-5	1814	-1	33
z		RIR	115	1905		
o		RRO	88	624		
n	Furnace A28-29	LO	101	268		
t		LI	75	149		
a		RO	35	1992	-1	21
l		RI	-19	565		
		RIR	168	1578	-1	-8
S		RRO	48	149		
u	Cement Pad A28-29	T	8	1992	2	-13
r		T	-19	1933	-1	-59
f		L	22	922		
a		R	61	30		
c	Cement Pad A27-28	T	75	1457	-1	-8
e		T	35	1665	-1	-13
s	Furnace A17-18	T	35	476		
		OF	8	208		
		LO	61	-30		
		RO	88	208		
		RRO	168	89		
	Kiln A15	BI	8	1992	2	-33
		TI	-19	1873	2	-21
		LO	222	89		
		RO	-32	149		
		TO	-19	268		
	Furnace Pad A17	T	88	1576	2	8
		T	8	1873	-1	17
	Furnace A12-13	RO	88	1397	-1	-33
		LO	101	1338		
		TO	8	446		
		RO	21	238		
	Furnace A11-12	TO	8	-149		
		LO	75	862		
		RO	8	1338		
		RRO	68	89		
		FL	115	327		
	Round Furn. B-13	T	115	-387		
	Work Table B-14	WT	61	208		
		WT	-32	268		
		WT	48	624		
	Steel Rack A9-10	SR	-32	89		
		SR	21	-178		
		SR	21	149		
	Mezzanine A31	T	35	89		
		T	-5	238		
		T	48	-59		

129507

Equipment Scan Data

EQUIP.XLS Location	Location Equipment	See Note	Direct (DPM/100 cm^2) Alpha	Direct (DPM/100 cm^2) Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable (DPM/100 cm^2) Beta/Gamma
		T	8	416		
		ST	.5	.308		
		STPL	.5	.146		

Note:
- BI - bottom in
- FL - front ledge
- L - left
- LI - left in
- OF - outside front
- R - right
- RI - right in
- RO - right out
- RRI - rear in
- RRO - rear out
- SR - steel rack
- ST - stairs
- STPL - stair platform
- T - top
- TI - top in
- TO - top out
- WT - work table

129507

BH-SCAN.XLS

Borehole Scan Data

Borehole Number	Location	Direct (DPM/100 cm^2) Beta/Gamma	Borehole Number	Location	Direct (DPM/100 cm^2) Beta/Gamma
1E	AREA	880	18	TOP	587
2E	TOP	293	18	BOTTOM	293
2E	BOTTOM	293	18	AREA	880
2E	AREA	880	19	TOP	587
3E	TOP	293	19	BOTTOM	587
3E	BOTTOM	293	19	AREA	1173
3E	AREA	880	20	TOP	587
4E	TOP	587	20	BOTTOM	587
4E	BOTTOM	587	20	AREA	1173
4E	AREA	880	21	TOP	293
5E	TOP	293	21	BOTTOM	293
5E	BOTTOM	880	21	AREA	880
5E	AREA	880	22	TOP	293
6E	TOP	293	22	BOTTOM	587
6E	BOTTOM	587	22	AREA	880
6E	AREA	880	23	TOP	293
1	TOP	880	23	BOTTOM	293
1	BOTTOM	587	23	AREA	880
1	AREA	880	24	TOP	178
3	TOP	1466	24	BOTTOM	-414
3	BOTTOM	880	24	AREA	474
3	AREA	4399	25	TOP	1466
4	TOP	5865	25	BOTTOM	1173
4	BOTTOM	4399	25	AREA	1173
4	AREA	13196	26	TOP	587
5	TOP	293	26	BOTTOM	587
5	BOTTOM	880	26	AREA	880
5	AREA	880	27	TOP	-118
8	TOP	293	27	BOTTOM	-414
8	BOTTOM	293	27	AREA	-414
8	AREA	880	27	BLOCKS	-118
10	TOP	293	28	TOP	-118
10	BOTTOM	293	28	BOTTOM	-414
10	AREA	880	28	BLOCKS	-118
11	TOP	6158	30	TOP	-118
11	BOTTOM	7331	30	BOTTOM	-118
11	AREA	5865	30	AREA	-118
13	TOP	587	30	BLOCKS	-118
13	BOTTOM	587	31	TOP	-118
13	AREA	880	31	BOTTOM	178
14	TOP	4399	31	AREA	-118
14	BOTTOM	3226	31	BLOCKS	-118
14	AREA	3226	32	TOP	880
15	TOP	880	32	BOTTOM	587
15	BOTTOM	587	32	AREA	880
15	AREA	880	35	TOP	-118
16	TOP	587	35	BOTTOM	30
16	BOTTOM	293	36	TOP	474
16	AREA	880	36	BOTTOM	-414
17	TOP	587	36	AREA	474
17	BOTTOM	587	37	TOP	178
17	AREA	880	37	BOTTOM	-118
			37	AREA	178

129507

Elevated Wooden Blocks and Concrete Pieces Scan Data

BLOCKS.XLS

Direct (DPM/100 cm^2)

Type	Beta/Gamma	Type	Beta/Gamma	Type	Beta/Gamma
W	2043	W	1168	C	1460
o	2627	o	1752	o	1460
o	2919	o	1168	n	16058
d	348854	d	1460	c	7882
e	9834	e	2043	r	9547
n	2043	n	1752	e	233083
	876		27733	t	11999
B	876	B	2335	e	261276
l	1168	l	1460		1460
o	10217	o	4963	P	1460
c	1168	c	3795	i	21895
k	2627	k	6258	e	8174
s	1784	s	33572	c	12553
	5839		7590	e	1752
	1752		1752	s	7882
	3503		2043		2335
	1752		3211		1752
	9834		1168		2043
	2919		1168		2043
	2043		3211		1460
	1460		2043		2919
	1168		1752		1752
	1168		8422		2043
	1460		1752		2335
	80280		1168		1168
	7289		2043		6130
	3794		2043		3211
	1460		1752		2043
	1460		3795		1460
	1752		3211		1752
	3795		42330		
	1460		2335		
	2919		1752		
	1168		4671		
	2043		1168		
	1460		2335		
	1752		4087		
	11093		1752		
	1752		2919		
	2919				

	Number of Readings	DPM/100 cm^2 Minimum	Maximum	Average
Wooden Blocks	79	876	348854	9416
Concrete Pieces	30	1168	261276	20912

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Appendix D

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Exterior Gamma Walkover Survey (CPM X 1000 with FIDLER)

South — North

West \ East	-18	-12	-6	0	6	12	18	24	30	36	42
-12	6-9	6-11	6-8	7-12	6-10	7-13	9-11				-12
-6	7-12	7-11	7-12	10-14	9-14	9-13	12-15	11-15	7-14		-6
0	7-11	8-11	9-14						13-17		0
6	9-11	9-12	10-14						12-16		6
12		8-11	10-14						12-17	9-18	12
18		9-12	9-14						12-17	10-14	18
24		10-13	9-12						11-17	10-14	24
30		8-15	10-14						13-17		30
36		10-14	9-12						7-17	11-16	36
42		10-13	11-15						7-17	11-14	42
48		11-15	11-15						10-15	9-15	48
54		11-14	11-16						12-17	12-14	54
60		10-15	11-14		Building 23				12-15	8-15	60
66		10-15	10-15						12-15	12-15	66
72		10-15	12-15						12-16	11-15	72
78		11-15	11-16						12-16	11-17	78
84		7-10	12-15	12-15					11-15	12-17	84
90		8-10	7-9	9-12					8-14	10-17	90
96		7-8	7-9	8-12					10-14	10-15	96
102		8-9	7-10	7-13							102
108		7-9	7-10	9-14					6-11		108
114		8-10	8-12	10-15					9-12	11-15	114
120		8-10	8-12	8-16					9-11	10-13	120
126		9-12	12-16	12-16					6-12	8-14	126
132		12-16	9-11	11-18					11-14	9-13	132
138		10-15		13-19	12-15	10-19	10-15	11-15	9-13		138
144		7-11		10-15	9-12	10-13	10-12	9-12	10-12		144
148	8	-12	-6	0	6	12	18	24	30	42	148

South — North

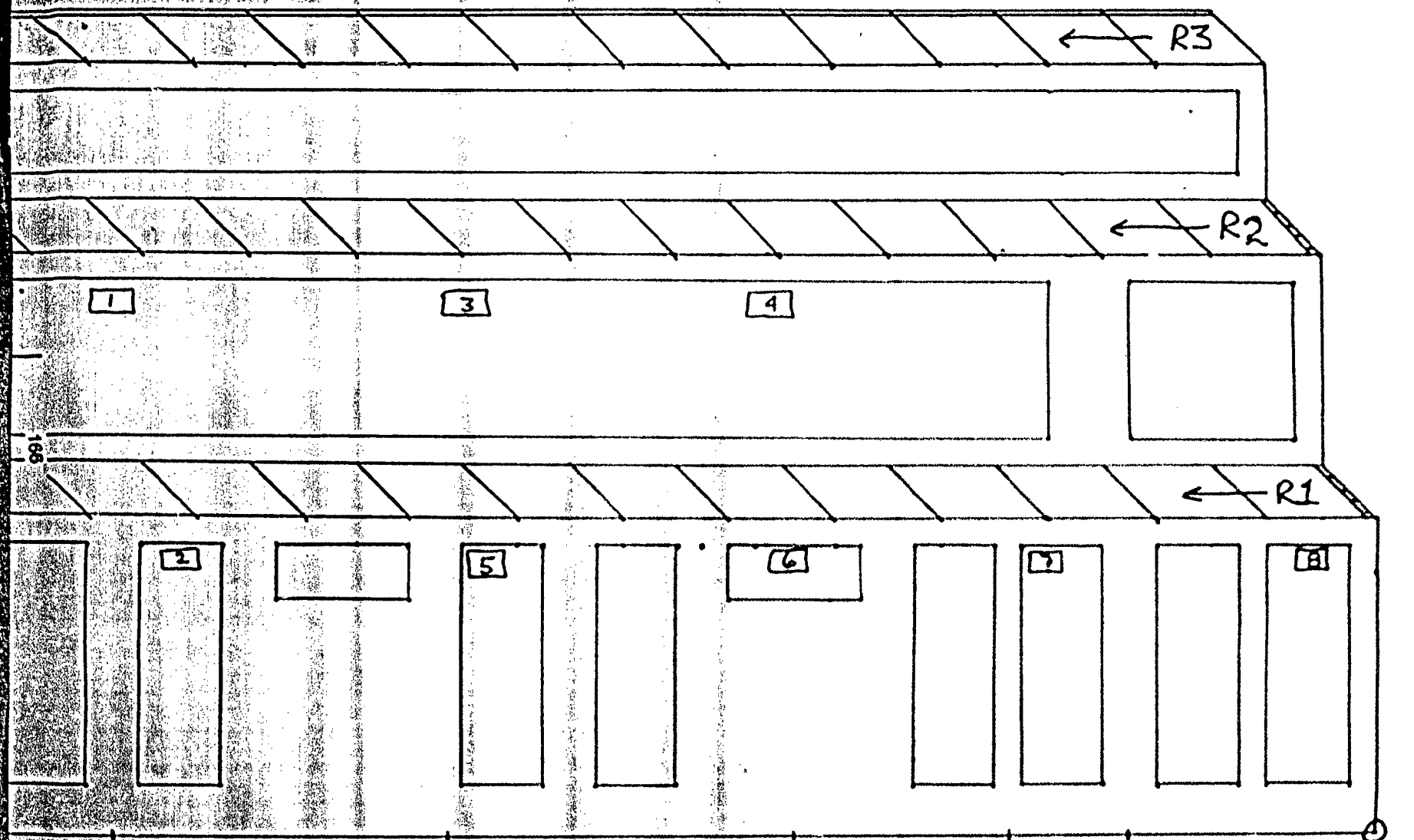
129507

WINDOW.XLS

EXTERIOR WINDOW SURVEY

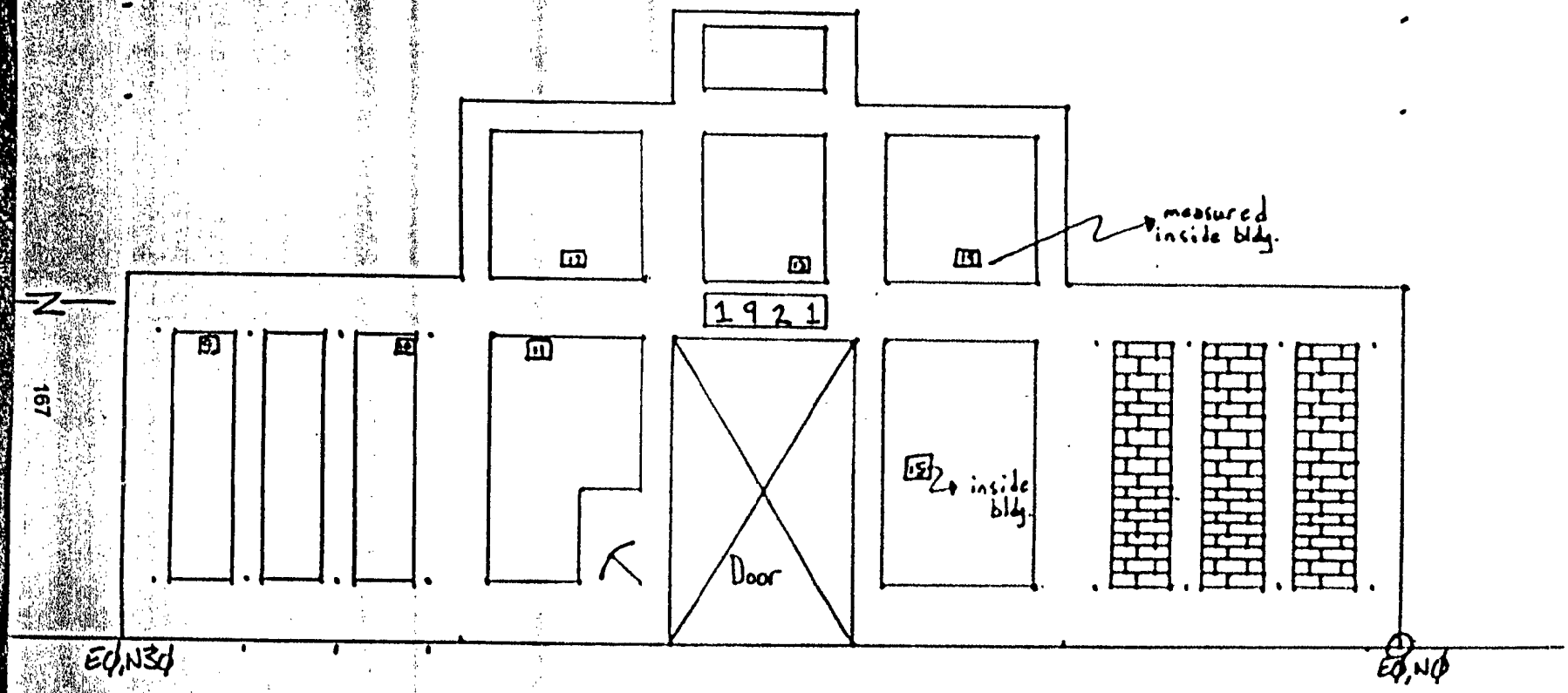
Location Number	Direct (DPM/100 cm^2) Beta/Gamma	Location Number	Direct (DPM/100 cm^2) Beta/Gamma
1	135	18	715
2	522	19	739
3	401	20	256
4	280	21	353
5	764	22	715
6	449	23	981
7	643	24	812
8	619	25	256
9	619	26	522
10	812	27	-82
11	401	28	256
12	570	29	184
13	449	30	377
14	6418	31	570
15	909	32	425
16	884	33	522
17	280		

0023



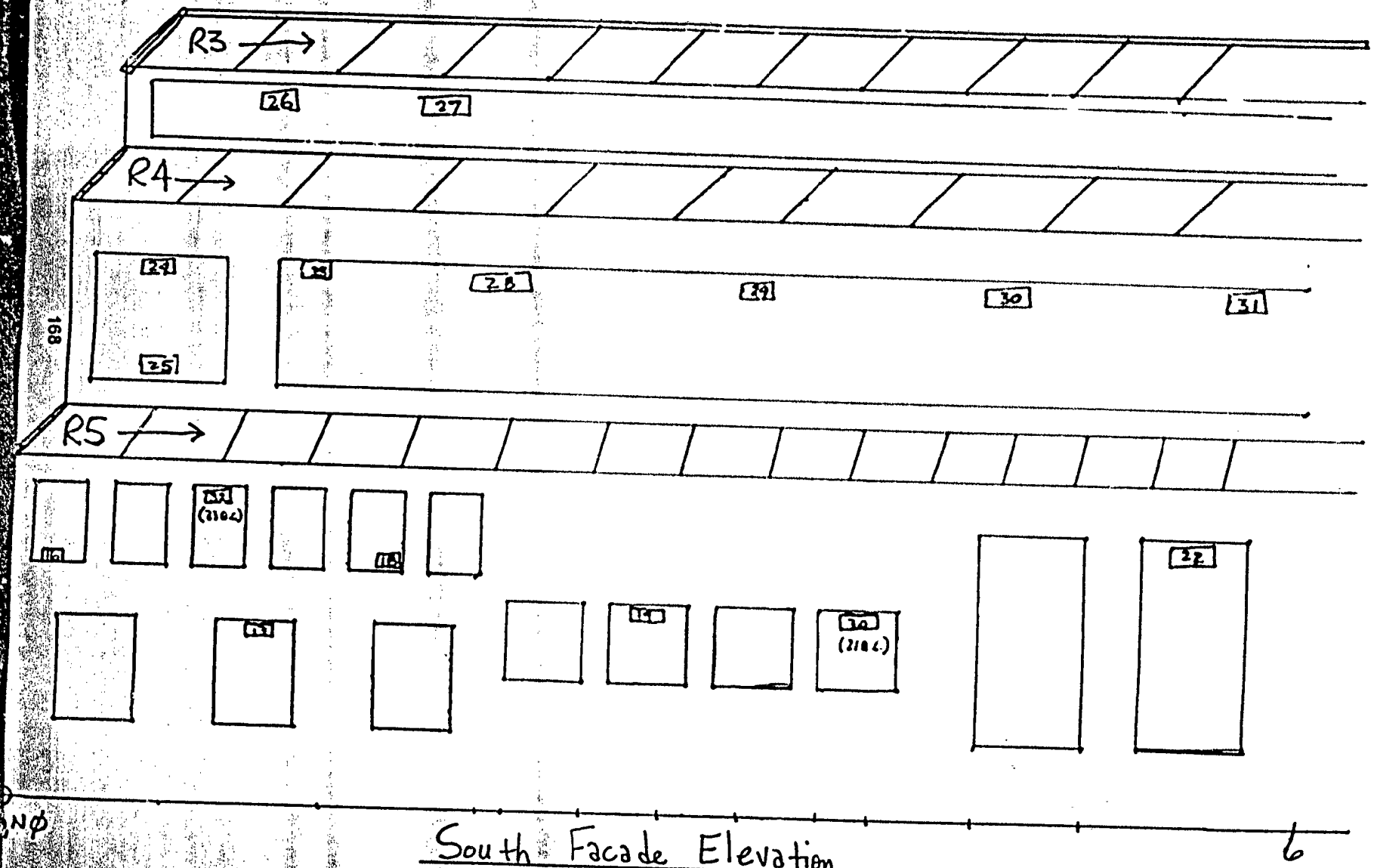
North Facade Elevational
Window Survey Points
Building 23
"2" = Sec 1 point

LRS6E1



West Facade Elevation.
Window Survey Points
"□" indicates survey point
Building 23

139527



South Facade Elevation

Window Survey Points

'#' indicates Survey location

Building 123

0 0 2 4 0

129507

Exterior West Wall Scan Data

EW-WALL.XLS		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
12/29/94	1	173	505				
s	2	120	743				
u	3	13	208				
r	4	133	178				
v	5	27	59				
e	6	-40	565				
y	7	27	803				
	8	160	416				
	9	40	535				
	10	-40	803				
	11	227	8117			2	-33
	12	67	2943			-1	-29
	13	147	1814			-1	-29
	14	334	4014			-1	0
	15	-13	-505				
	16	93	-624				
	17	40	-208				
	18	-93	1308			-1	-4
	19	-67	565				
	20	67	1130			-1	33
	21	-67	1754			-1	-71
	22	-93	1784			-1	-17
	23	-80	1308			-1	-79
	24	-93	803			-1	-13
	25	-40	1011			-1	-17
	26	0	1695			-1	4
	27	-120	1516			-1	-4
	28	120	1754			-1	-4
	29	40	1160				
	30	147	446				
	31	-13	684			-1	-54
	32	-27	1397			-1	-54
	33	40	1338			-1	29
	34	-93	1160			-1	21
	35	827	3241			-1	33
	36	227	1219			-1	-4
	37	173	1100			-1	-29
	38	93	1992			-1	42
	39	93	2884			2	4
	40	173	1100			5	13
	41	27	1933			-1	-21
	42	80	2319			-1	-25
	43	267	1992			-1	-13
	44	747	2349				
	45	307	2289			-1	4
	46	13	803				
	47	13	1070				
	48	67	30				
	49	120	149			-1	-17

129507

0024

Exterior West Wall Scan Data

EW-WALL.XLS		Direct (DPM/100 cm^2)					
	Location	Before Dust Removal		After Dust Removal		Transferrable (DPM/100 cm^2)	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
	50	40	287			-1	-25
	51	307	1695			-1	0
	52	267	2022			-1	-21
	53	307	3051			-1	42
	54	-40	1665			-1	-4
	55	93	1309			-1	0
	56	133	1724			-1	63
	57	40	1933			-1	-8
	58	93	1665			-1	-25
	59	147	3051			2	
	60	107	2081				
12/30/94	1	254	1487			7	8
S	2	214	4103			2	-13
u	3	187	2289			-1	-13
r	4	173	2824			10	-33
v	5	254	2319			-1	-63
e	6	147	1278			2	0
y	7	187	1943			-1	25
	8	93	2081			2	17
	9	147	1397			2	-17
	10	120	1784			7	4
	11	93	1754			-1	33
	12	93	1070				
	13	120	2468			2	-21
	14	107	-178				
	15	93	-149				
	16	120	119				
	17	13	-178				
	18	27	-59				
	19	93	-258				
	20	80	149				
	21	93	1546				
	22	173	2022			2	-4
	23	227	1576			2	17
	24	93	1130				
	25	67	1160				
	26	53	1605			2	-13
	27	93	1160				
	28	40	1397			-1	8
	29	27	951				
12/2/94	1	315	1391			-1	46
S	2	207	1273			2	-50
u	3	234	1687			-1	-13
r	4	153	1332			-1	-29
v	5	207	1480			6	-21
e	6	153	3670			3	13
y	7	153	1273			-1	13

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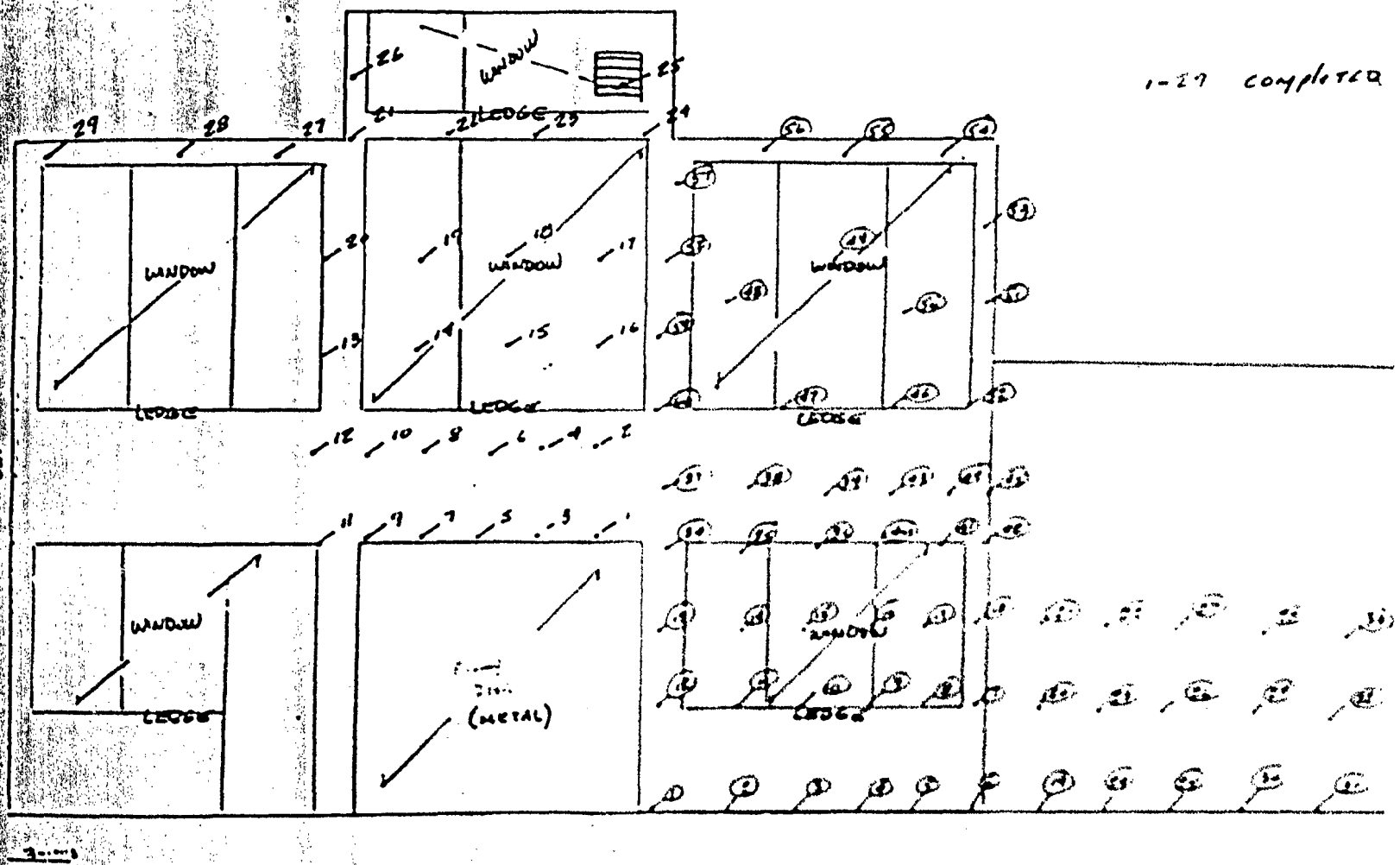
Exterior West Wall Scan Data

EW-WALL.XLS		Direct (DPM/100 cm^2)				Transferrable (DPM/100 cm^2)	
	Location	Before Dust Removal		After Dust Removal			
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
	8	46	1508			3	-54
	9	-8	1421			3	-8
	10	19	1421			-1	-46
	11	100	1125			2	-21
	12	-8	1154			-1	-38
	13	100	868				
	14	19	868				
	15	584	1687			3	-8
	16	46	1273			-1	33
	17	19	1628			3	29

0024

Bldg # 23
EXTERIOR - WEST WALL

1-27 completed 12/30/



Note * windows are covered with corrugated tin
All wall surfaces are brick

* Numbers that are circled done on 12/29/99
Now circled numbers done on 12/30/99

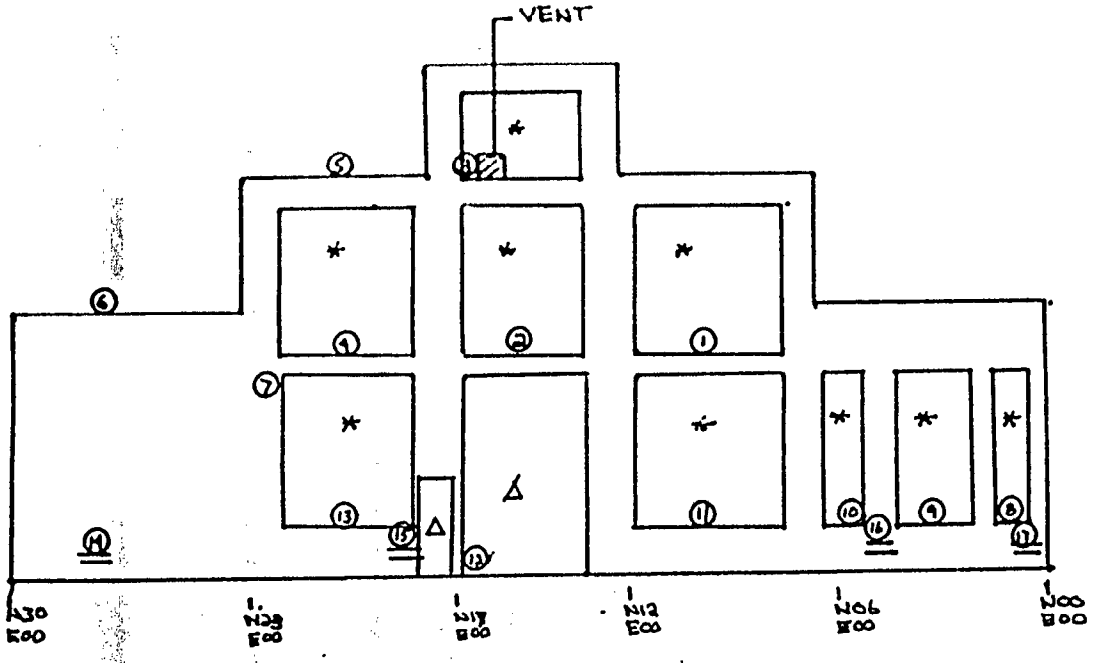
D-19693

139567

172

12/2/94

West Exterior Wall



189527

129507

Exterior North Wall Scan Data

EN-WALL.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Direct (DPM/100 cm^2) Before Dust Removal Beta/Gamma	Direct (DPM/100 cm^2) After Dust Removal Alpha	Direct (DPM/100 cm^2) After Dust Removal Beta/Gamma	Transferable (DPM/100 cm^2) Alpha	Transferable (DPM/100 cm^2) Beta/Gamma
W	1	127	951				-13
a	2	74	951			-1	33
l	3	392	1189			-1	33
l	4	418	1487			-1	33
	5	180	357				
	6	180	1487				
	7	233	535				
	8	365	1130				
	9	551	1903			-1	21
	10	233	297				
	11	339	1189			-1	-8
	12	260	1189				
	13	233	1249			2	-13
	14	207	951			-1	-50
	15	392	476				
	16	207	1249				
	17	339	773			-1	-8
	18	127	178			-1	-13
	19	286	119				
	20	286	1724				
	21	365	59				
	22	101	1189				
	23	74	1546				
	24	286	1011			2	-33
	25	551	714			2	-21
	26	551	59				
	27	74	-119				
	28	339	595				
	29	524	297			2	8
	30	154	476			-1	17
	31	392	1130			-1	-33
	32	365	416			-1	-8
	33	260	2319				
	34	392	-178				
	35	-138	1962			-1	-54
	36	-85	1605			-1	-50
	37	8	1546			-1	-29
	38	-72	2081			-1	-54
	39	-151	1368			3	-42
	40	-5	1695			-1	-33
	41	140	1516			-1	4
	42	8	1903			-1	-25
	43	-164	2081			3	-25
	44	-204	1249			3	-13
	45	-217	1070				
	46	-164	1100				

129507

Exterior North Wall Scan Data

EN-WALL.XLS		Direct (DPM/100 cm^2)				Transferable (DPM/100 cm^2)	
	Location	Before Dust Removal		After Dust Removal			
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
	47	-177	2873			3	-13
	48	-164	2914			-1	0
	49	-217	986				
	50	-217	1100				
	51	-204	1279			-1	33
	52	-217	1070				
	53	-217	1546			-1	-71
	54	-138	954				
	55	-217	1308			-1	38
	56	-58	1899			3	33
	57	-45	3092			6	4
	58	-151	1962			-1	-13
	59	-204	1635			-1	-25
	60	-217	1308			-1	-13
	61	-151	951				
	62	-217	822				
	63	-217	1665			3	-29
	64	-217	1338			6	0
	65	-217	992				
	66	-204	1457			-1	[illegible]
	67	-32	1338			3	-33
	68	-217	832				
	69	-204	1041				
	70	-217	714				
	71	-204	59				
	72	-191	59				
	73	8	1605			-1	-33
	74	-217	1249			-1	4
	75	-217	238				
	76	-124	1576			6	-60
	77	-72	1070				
	78	-164	773				
	79	-217	773				
	80	-217	1100				
	81	-217	1070				
	82	-217	2260			-1	-54
	83	-217	59				
	84	-85	1457			-1	-42
	85	-217	1100				
	86	-124	862				
	87	-151	1308			6	-8
	88	-164	832				
	89	-217	1368			3	-46
	90	-98	922				
	91	-204	714				
	92	-111	1041				

129587

Exterior North Wall Scan Data

EN-WALL.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferable (DPM/100 cm^2) Alpha	Transferable Beta/Gamma
	93	-217	684				
	94		1041				
	95	-138	2380			-1	-64
	96	-217	1835			3	-29
	97	-64	1041				
	98	-24	179				
	99	-11	654				
	100	-91	357				
	101	-51	862				
	102	-64	0				
	103	123	387				
	104	-51	416				
	105	-24	595				
	106	-11	30				
	107	83	624				
	108	-91	773				
	109	69	654				
	110	-104	773				
	111	-77	416				
	112	109	892				
	113	3	535				
	114	-37	238				
C	1	-48	1332			-1	13
o	2	-35	888				
u	3	-48	1361			3	-8
r	4	5	1332			6	-60
t	5	32	629				
y	6	-35	1095				
a	7	-35	769				
r	8	-48	414				
d							

129507

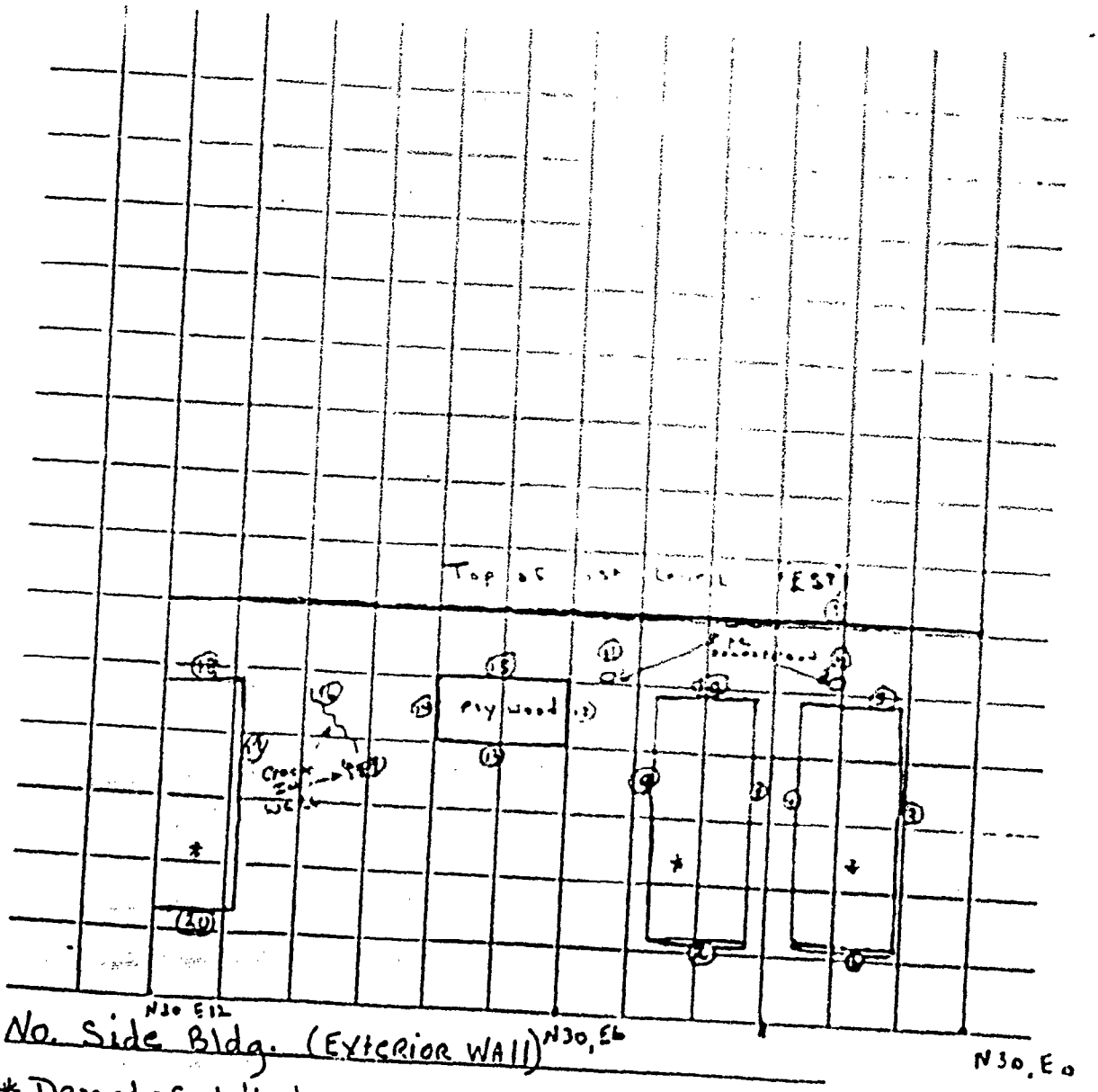
WALKOVER GAMMA SCAN DATA SHEET
(LARGE AREA)

ile: _____ Date _____ Technician _____

caler: _____ Detector _____ S/N _____

onversion: _____ Bldg _____

Coordinates E ... W



ks: No. Side Bldg. (EXTERIOR WALL)

*Denotes Windows

0024

129507

WALKOVER GAMMA SCAN DATA SHEET
(LARGE AREA)

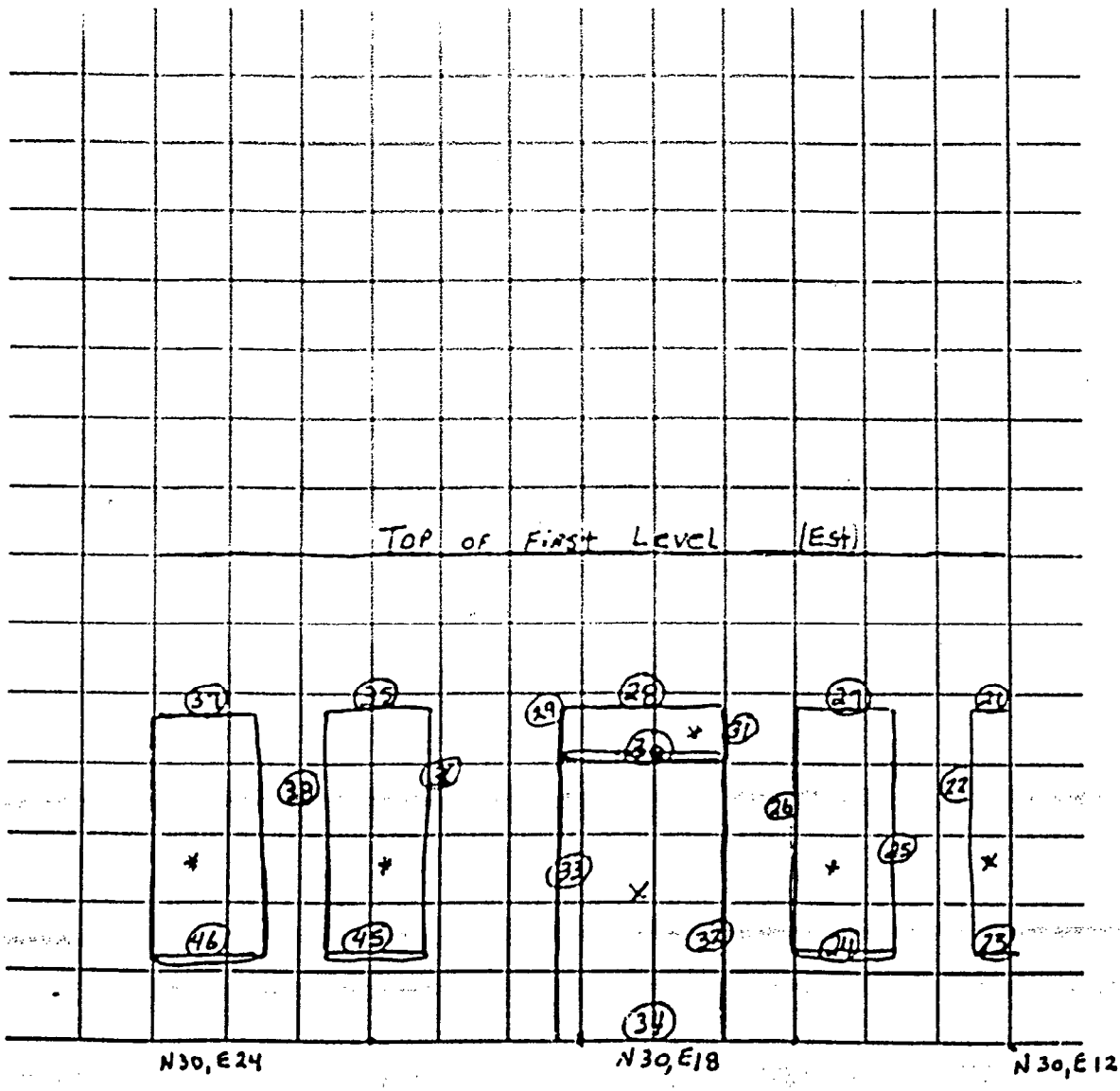
Site: _____ Date _____ Technician _____

Scaler: _____ Detector _____ S/N _____

Conversion: _____ Bkg _____

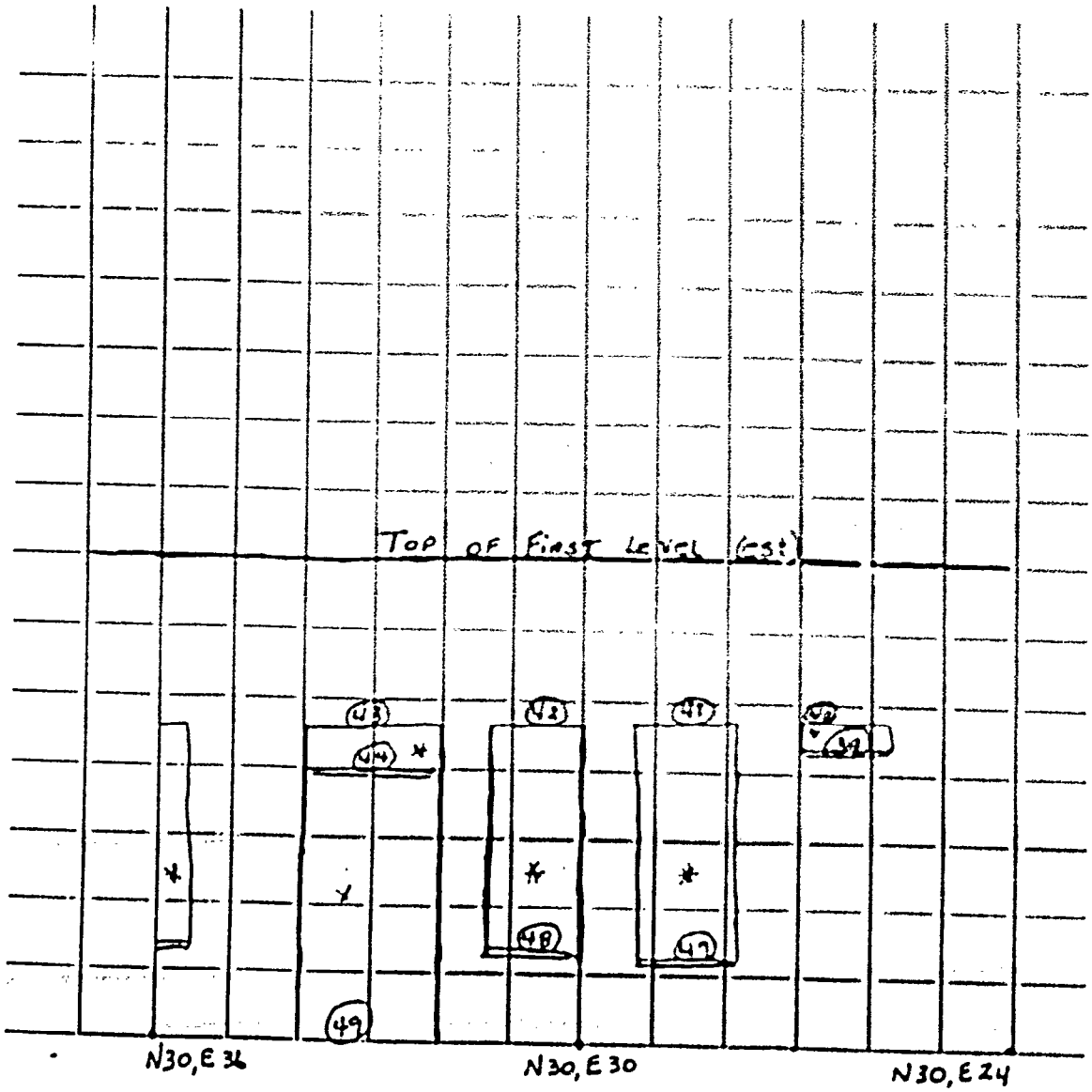
Coordinates E <---> W

Coordinates N <---> S



Remarks: No. Side Bldg. * denotes windows X denotes doors. (Exterior Wall)

139507



No Side of Bldg.
* denotes Windows
x denotes doors
(Exterior WAll)

129507

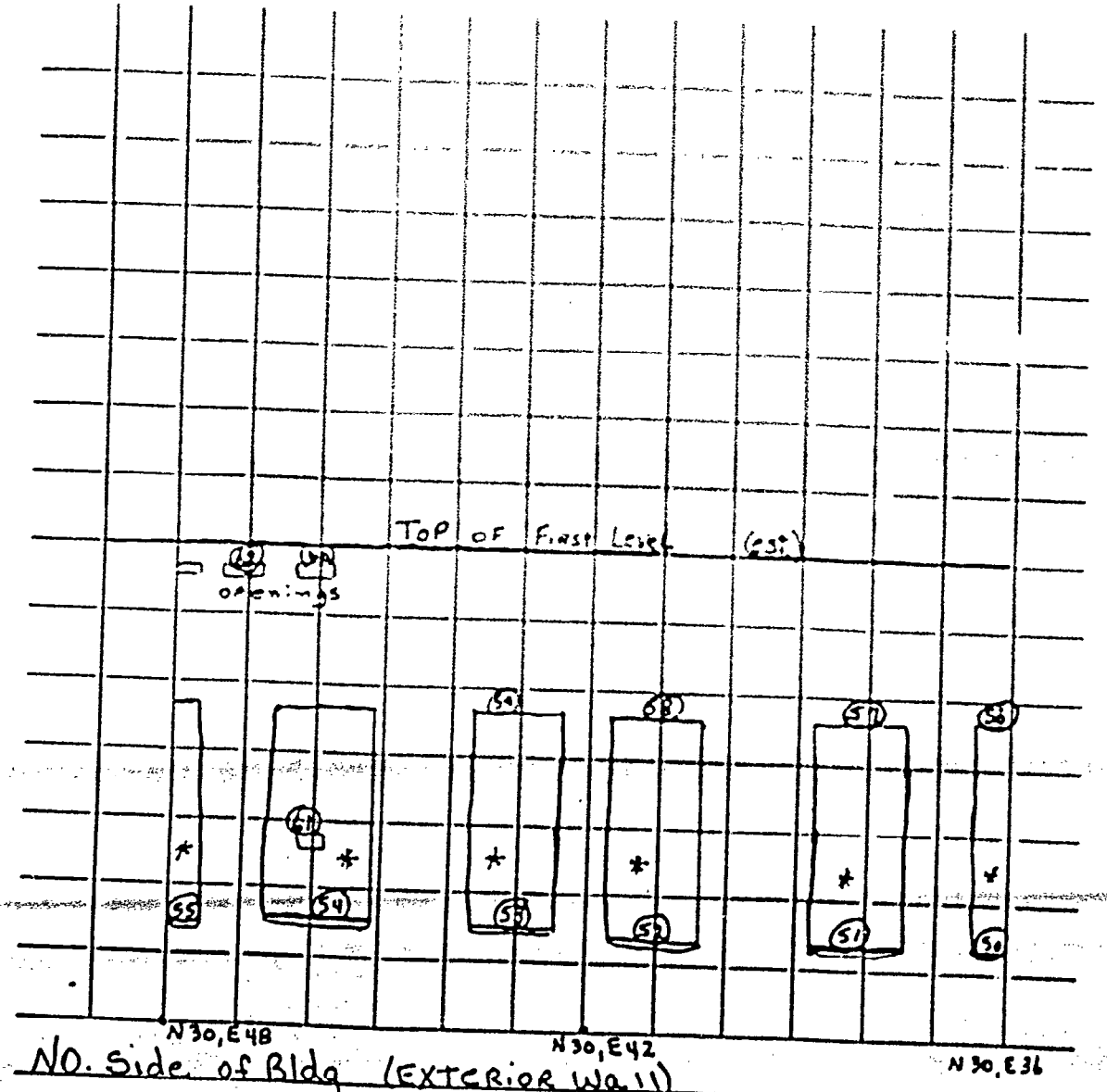
(LARGE AREA)

ile: _____ Date _____ Technician _____

Detector: _____ Detector _____ S/N _____

onversion: _____ Bldg _____

Coordinates E ⟨---⟩ W

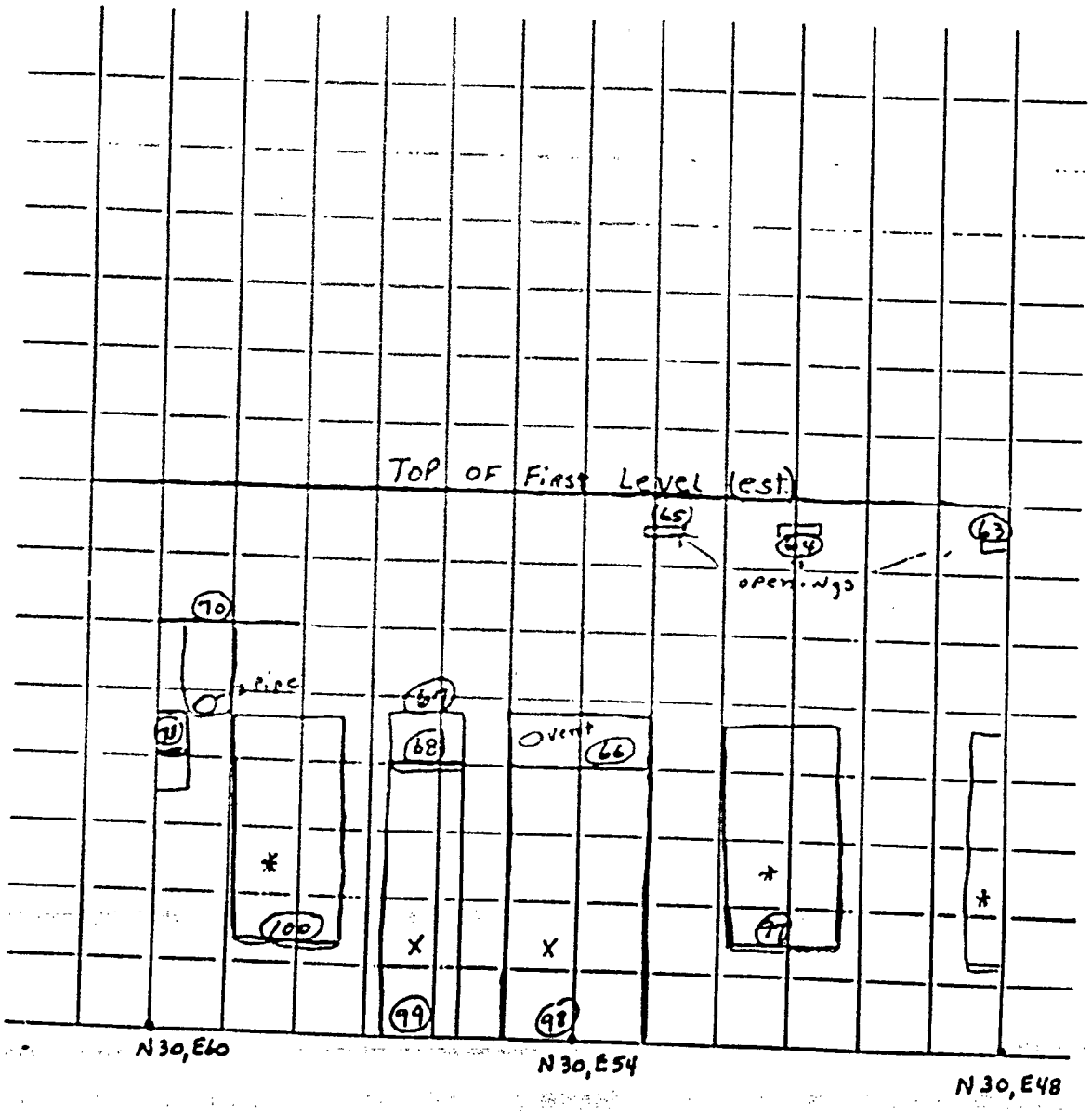


arks: No. Side of Bldg (EXTERIOR WALL)

* denotes Windows

129507

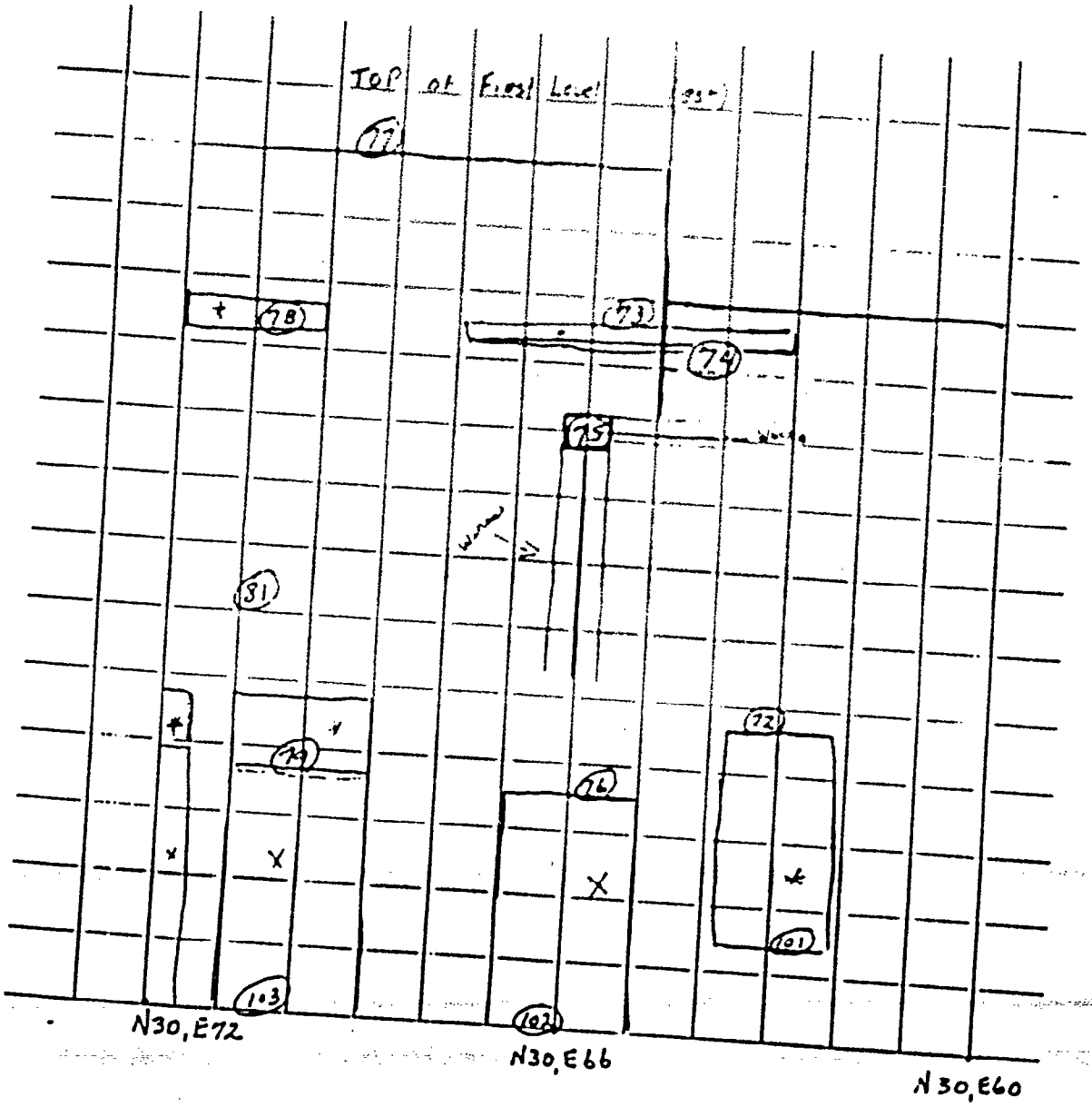
0-19626



NO. Side Bldg. (EXTERIOR WALL)
* denotes windows
X denotes DOORS

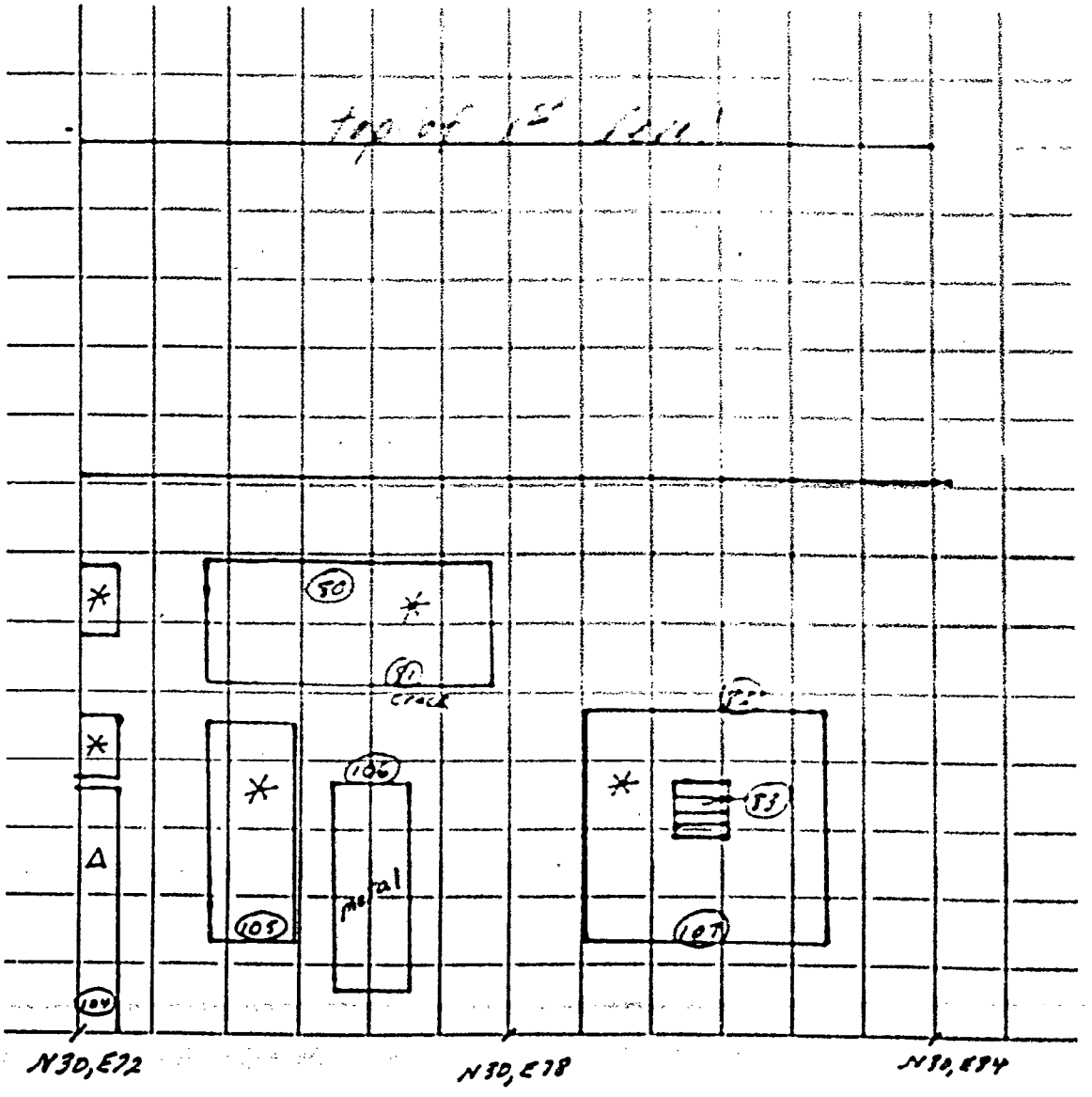
129587

0-5625



No. Side of Bldg (Exterior wall)
* denotes windows
x denotes Doors

129507

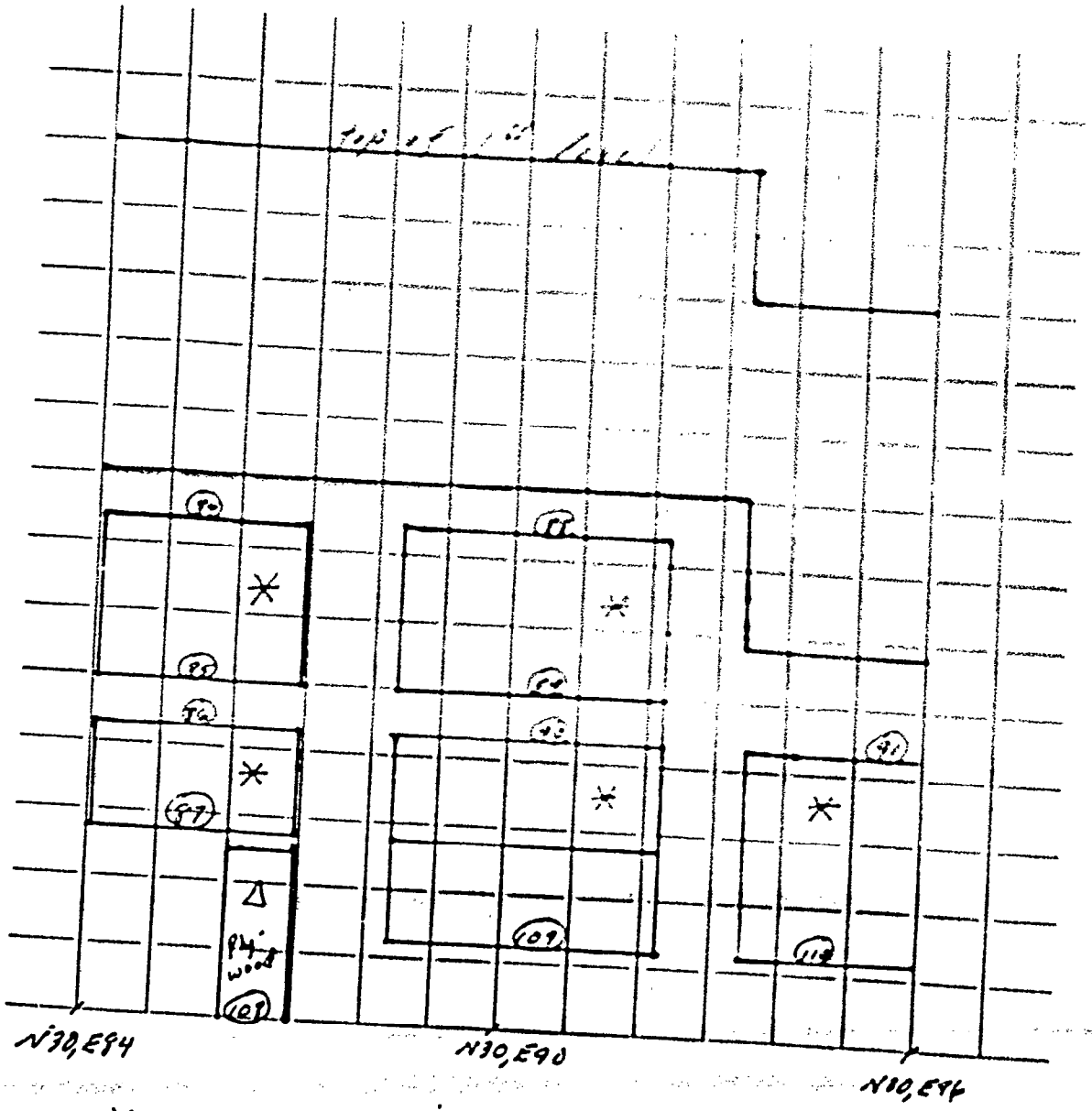


* denotes window

Δ denotes door

North Exterior Wall

129507



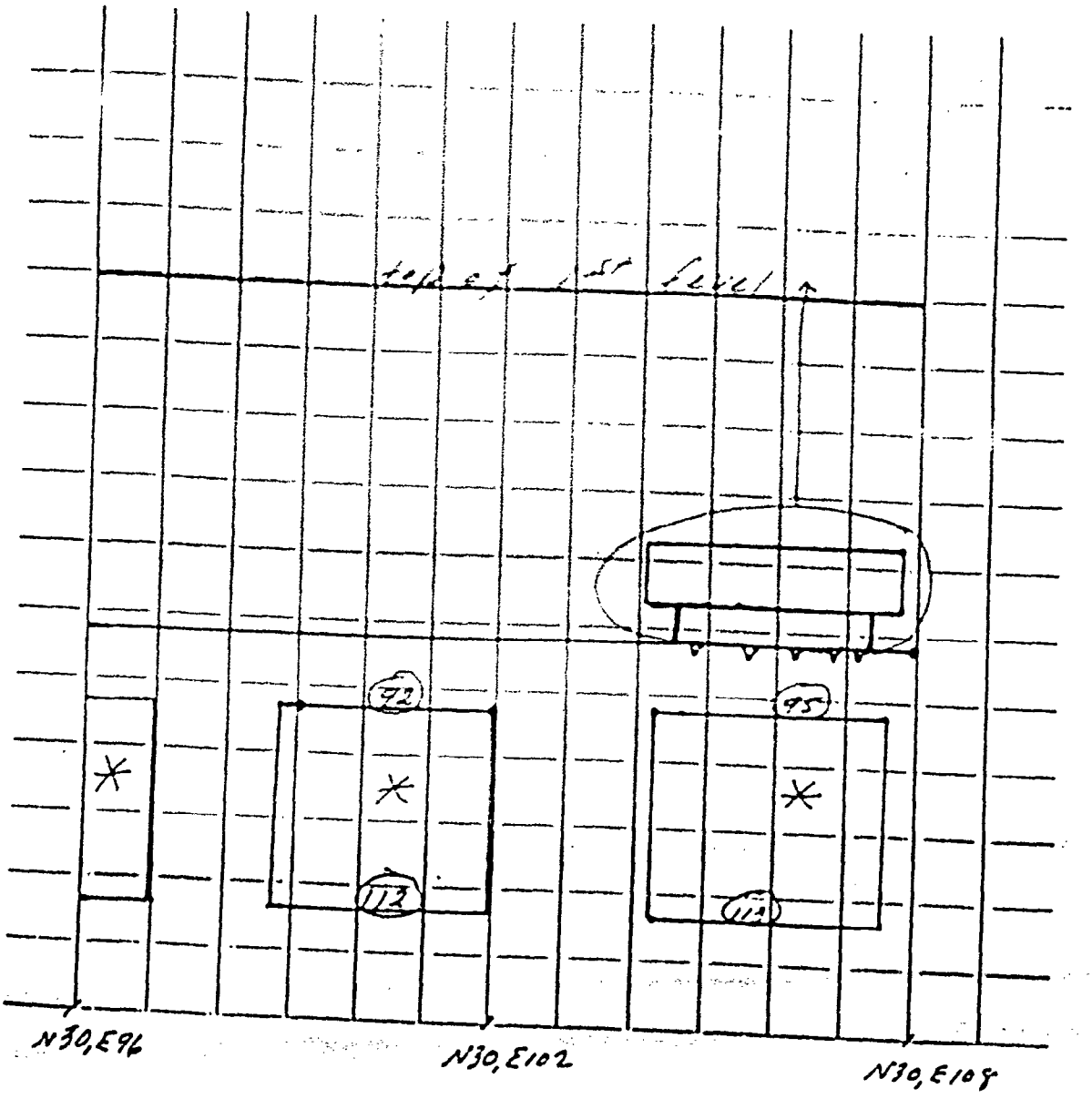
* denotes window

Δ denotes door

North Exterior Wall

129507

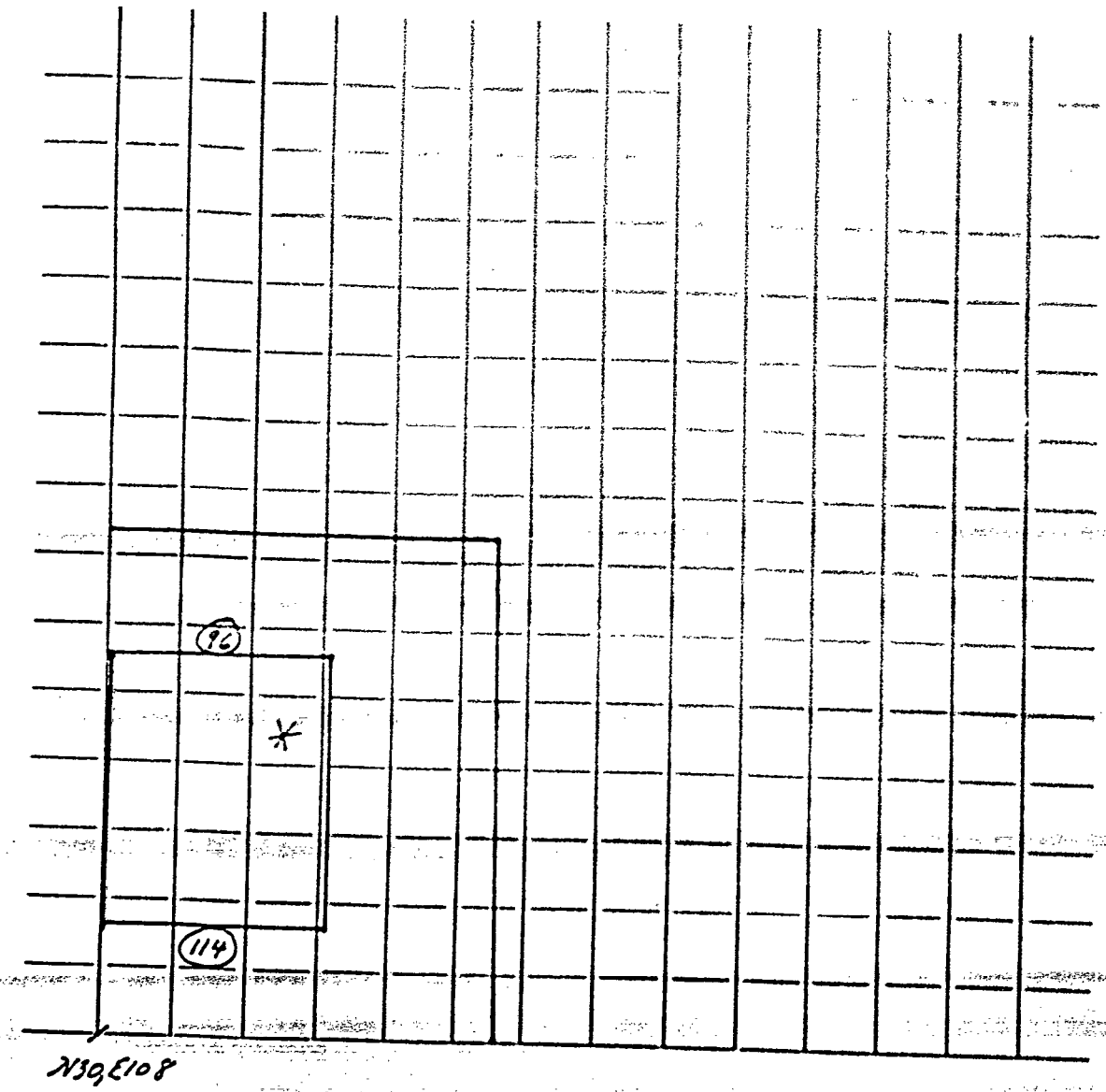
0- 9500



* denotes window

North Exterior Wall

129507



* denotes window

North Exterior Wall

129507

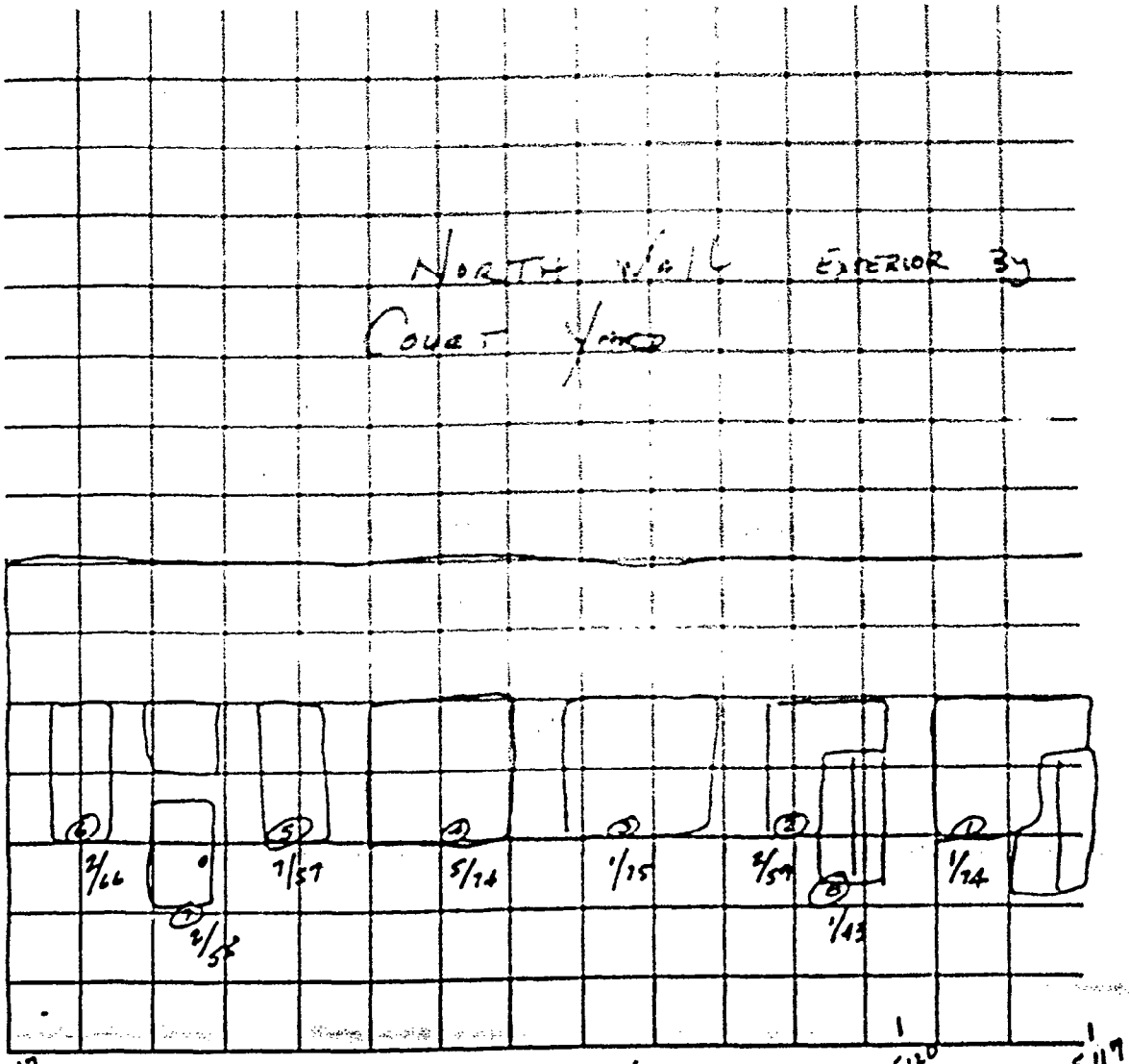
(LARGE AREA)

Date _____ Technician _____

ller: _____ Detector _____ S/N _____

nversion: _____ Bldg _____

Coordinates E <---> W



emarks: NORTH EXTERIOR WALL / COURTYARD

1295017

Exterior South Wall Scan Data

ES-WALL.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
W	1	180	799				
a	2	194	385				
l	3	113	-207				
l	4	180	877			5	-25
	5	73	740				
	6	167	562				
	7	86	858				
	8	234	917				
	9	382	1391			-1	-13
	10	-22	740				
	11	463	10743			5	-8
	12	167	1095				
	13	382	1095				
	14	32	799				
	15	221	2279			2	-42
	16	140	1391			-1	8
	17	355	503				
	18	-54	444				
	19	-67	592				
	20	-13	1332			9	4
	21	13	503				
	22	13	1065				
	23	-40	1125			-1	-50
	24	-94	1480			-1	-21
	25	-67	1006			3	-17
	26	-67	947				
	27	-40	355				
	28	-67	1065			-1	-17
	29	-94	0				
	30	-40	59				
	31	-13	355				
	32	13	1065			-1	-13
	33	-40	947			-1	-29
	34	309	1539			-1	-29
	35	94	178				
	36	13	2190			-1	0
	37	-67	237				
	38	67	0				
	39	-40	1421			-1	-8
	40	390	2782			3	-25
	41	417	355				
	42	175	0				
	43	525	1361			-1	-8
	44	525	-59				
	45	363	1776			-1	-42
	46	13	888				
	47	94	1480			3	-4
	48	-94	533				

129507

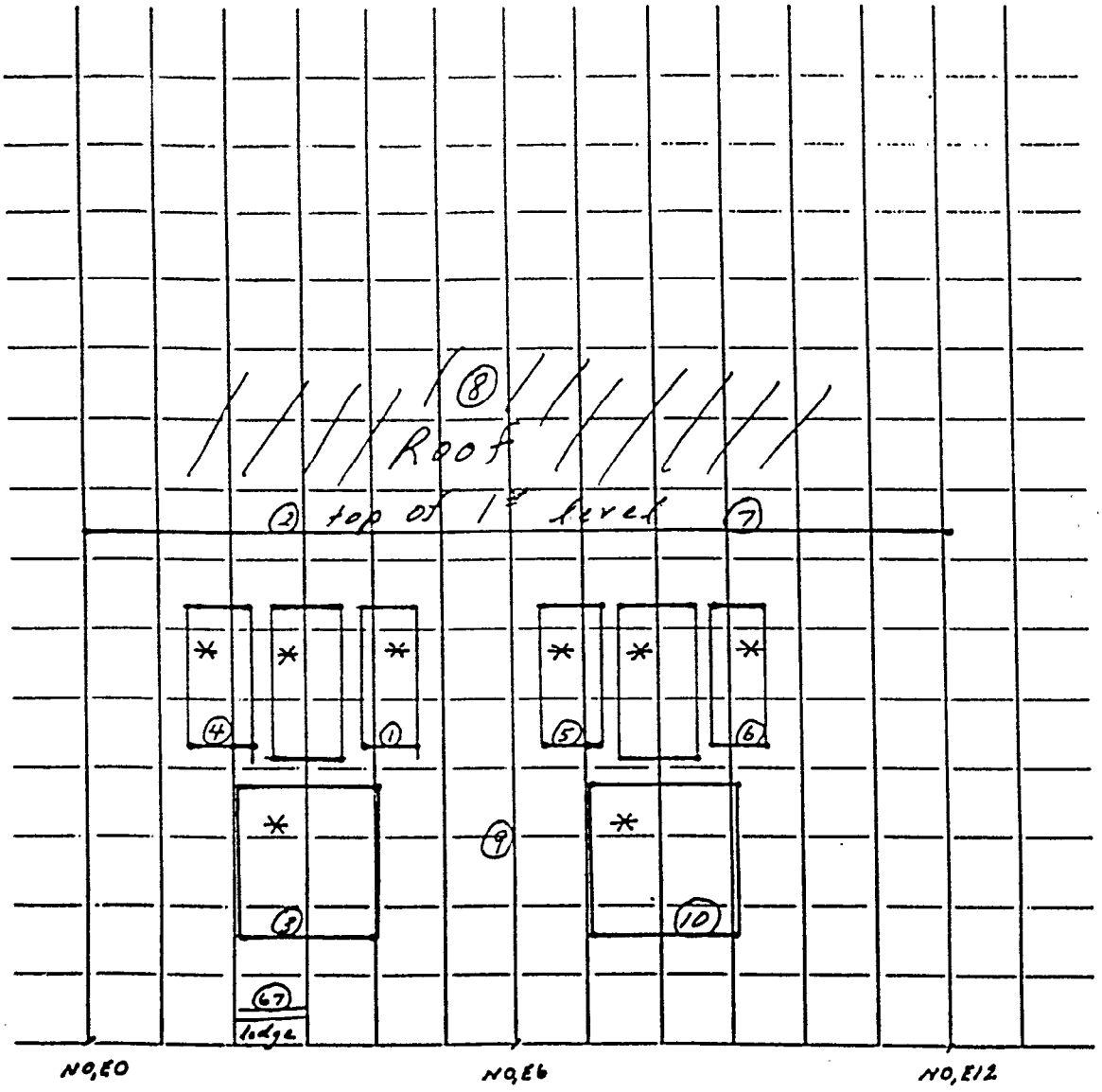
Exterior South Wall Scan Data

ES-WALL.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferable (DPM/100 cm^2) Alpha	Transferable (DPM/100 cm^2) Beta/Gamma
	49	390	1694				
	50	175	-356			-1	-8
	51	40	414				
	52	121	58				
	53	148	178				
	54	94	1361				
	55	282	1065			3	-50
	56	-13	947			-1	13
	57	67	1125				
	58	202	296			-1	4
	59	148	296				
	60	67	710				
	61	-13	-178				
	62	-13	1125				
	63	-67	-474			-1	-46
	64	-13	118				
	65	121	829				
	66	-67	178				
	67	94	1006				
	68	-8	1184			3	0
	69	19	829				
	70	-46	858				
	71	19	1273				
	72	100	1628			3	13
	73	19	799			3	-54
	74	-48	1776				
	75	-8	1361			3	-59
						3	8

129507

0-19623

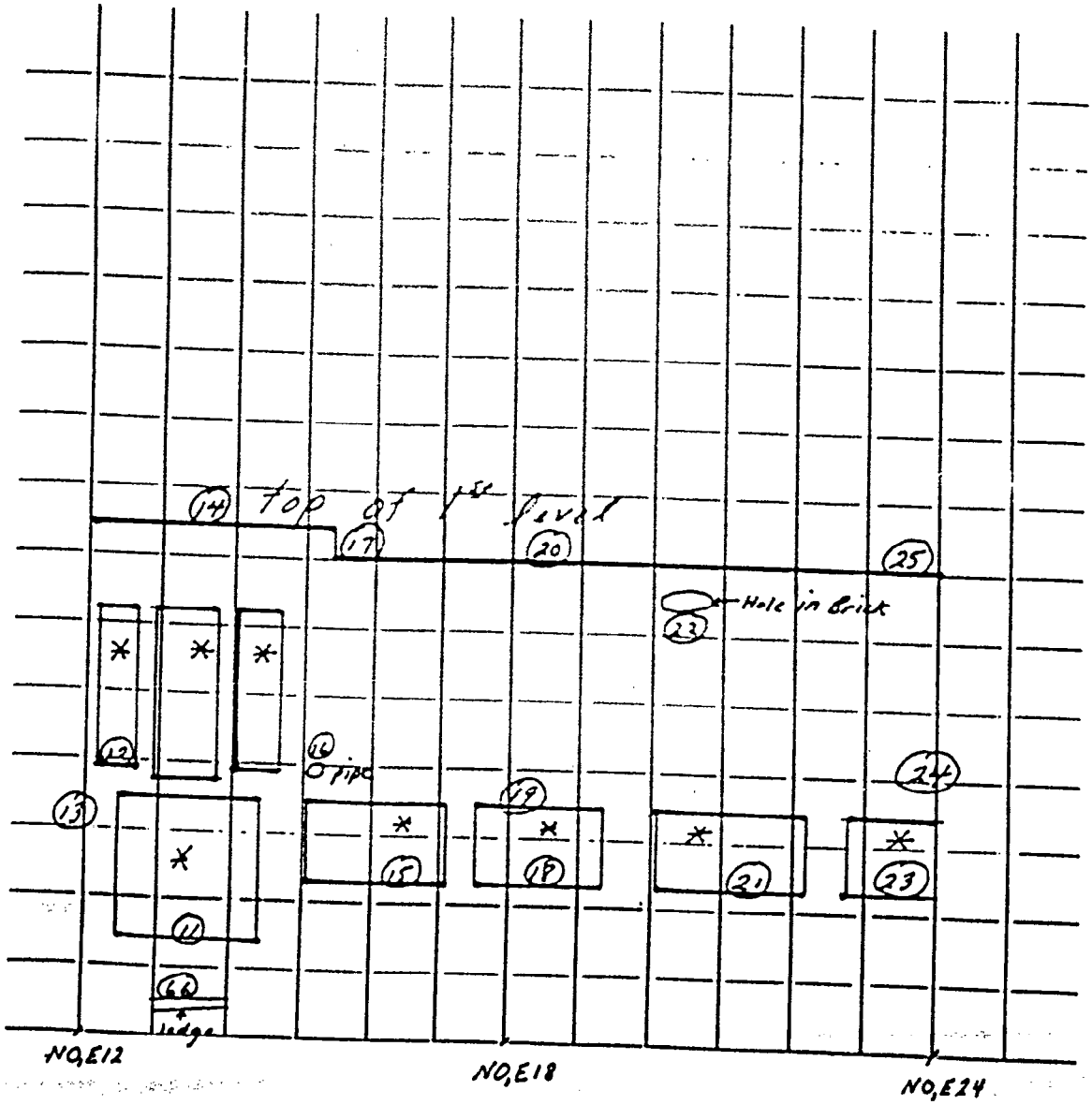


* denotes window

South Exterior Wall

129507

0-15623

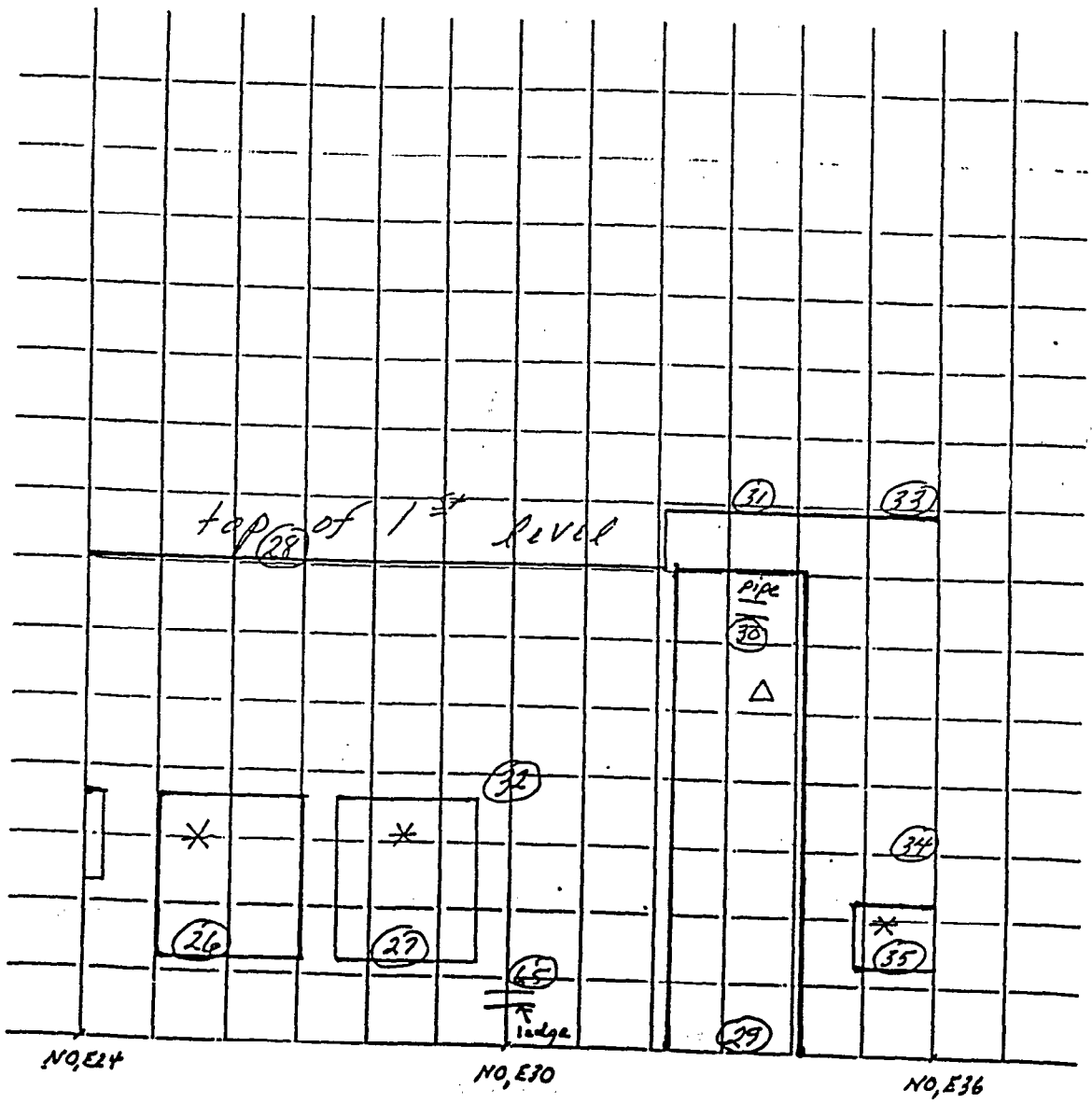


* denotes window

South Exterior Wall

129507

0-19623



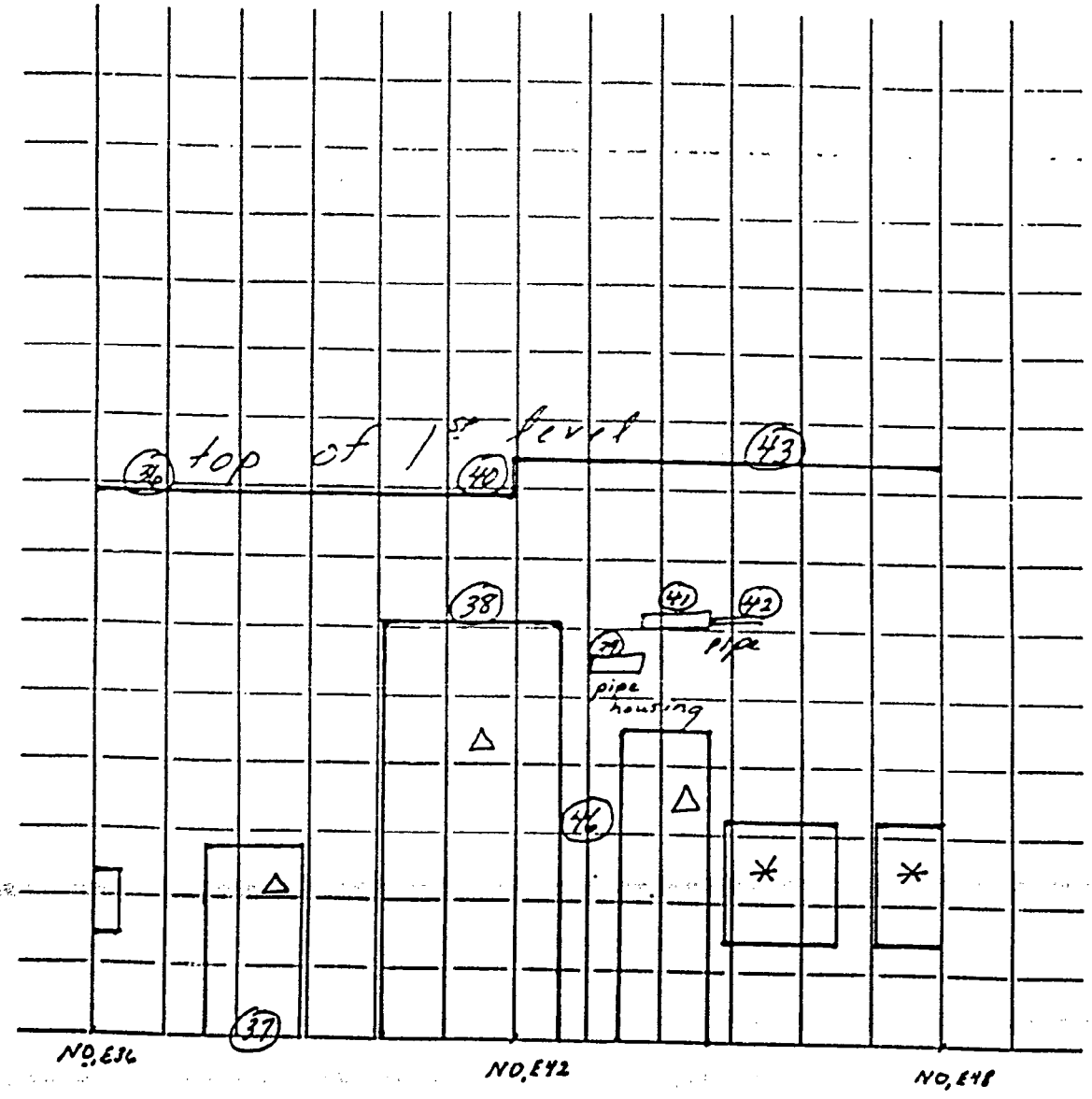
* denotes window

Δ denotes door

South Exterior Wall

129507

D-19623



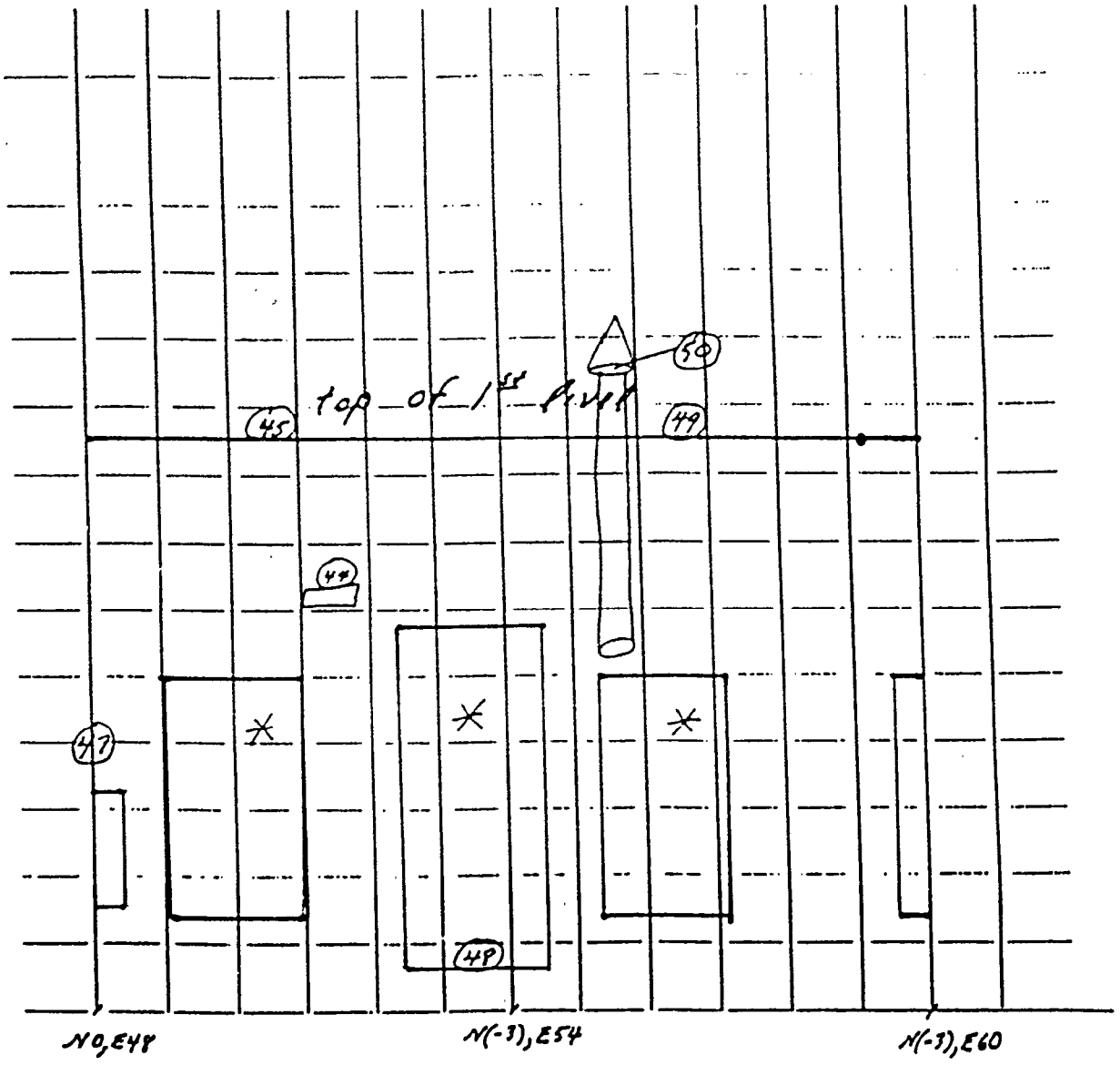
* denotes window

Δ denotes door

South Exterior Wall

129507

D-.3625

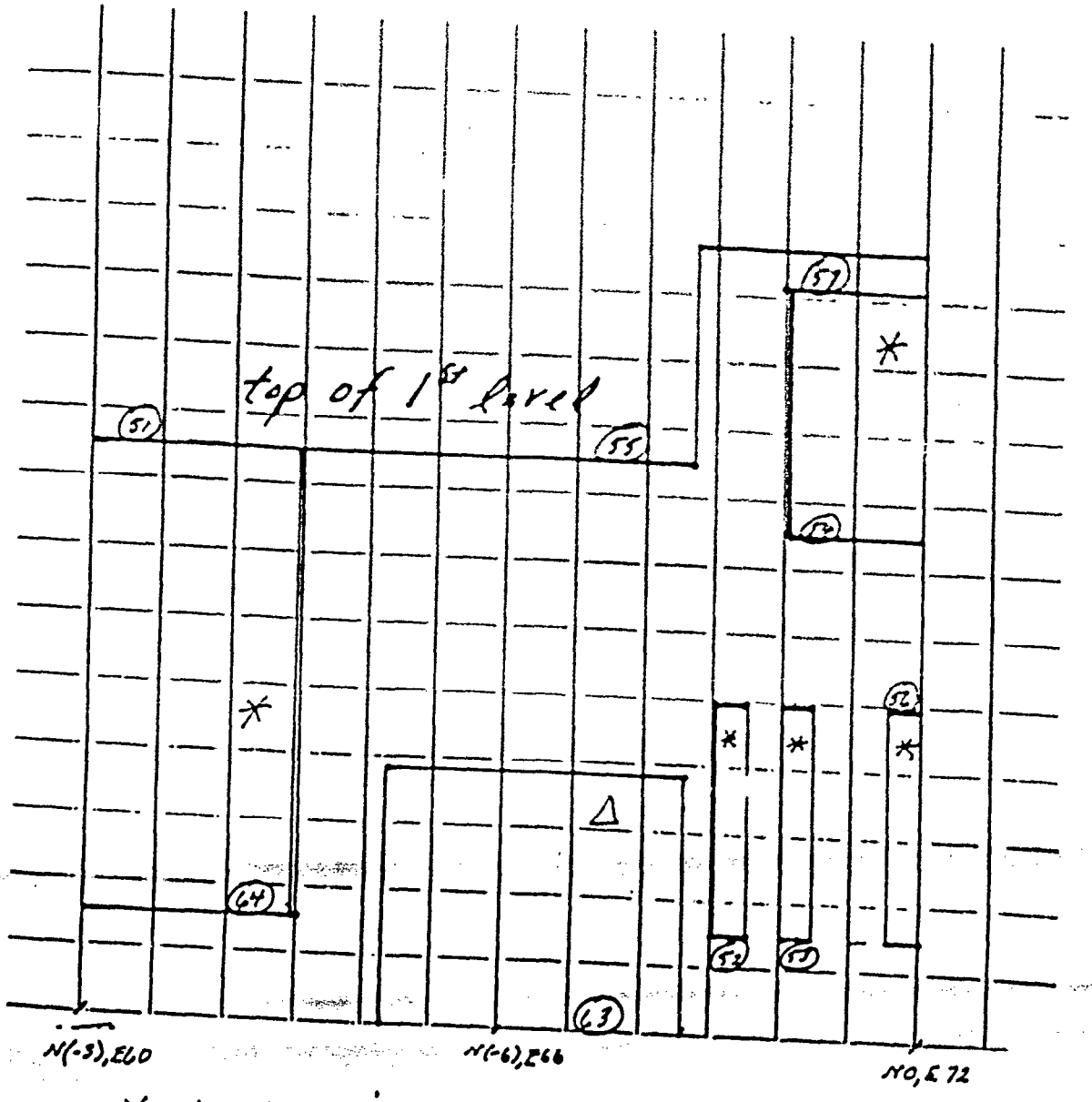


* denotes window

South Exterior Wall

129501

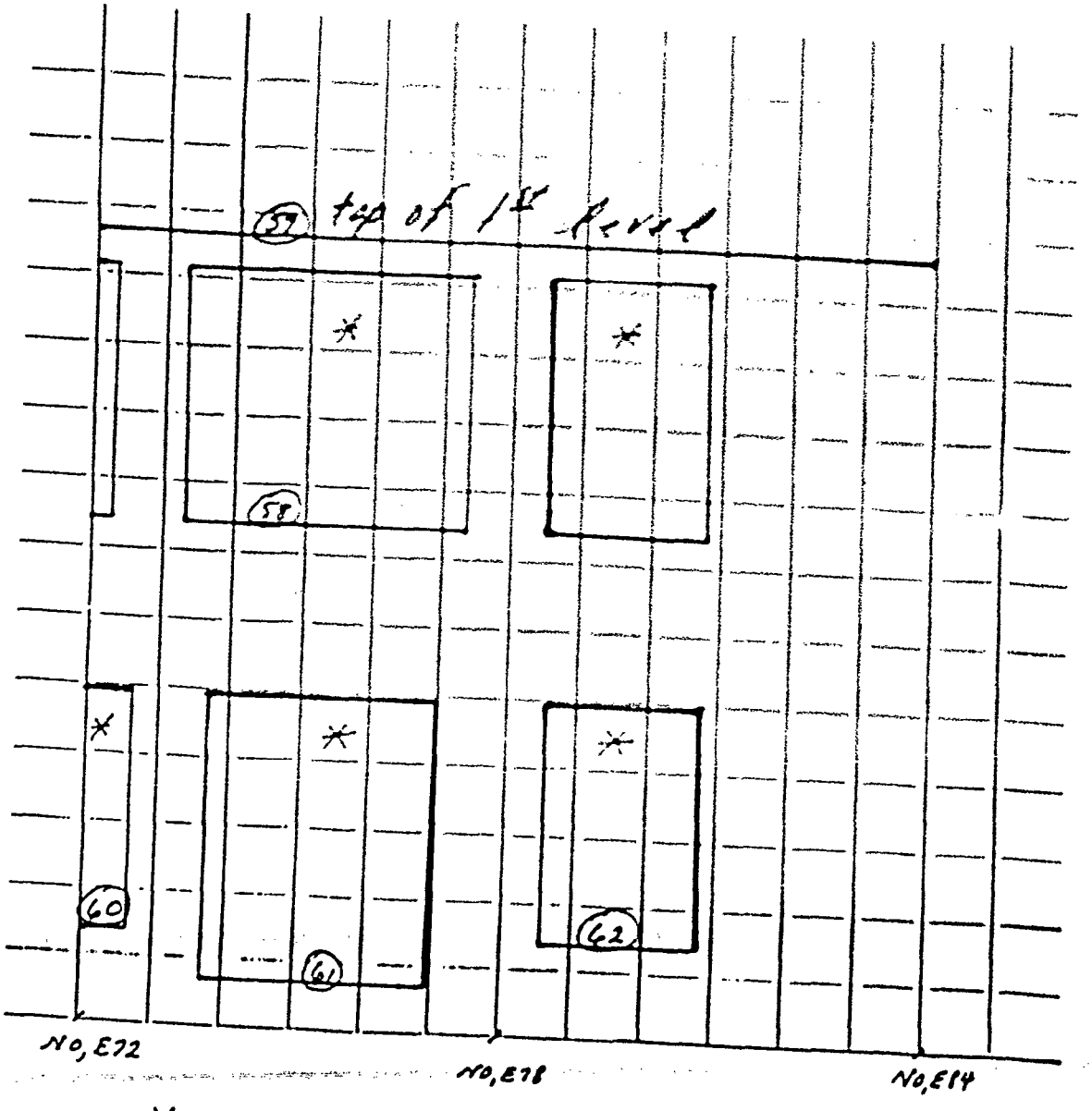
0-.5626



* denotes window

Δ denotes door

South Exterior Wall

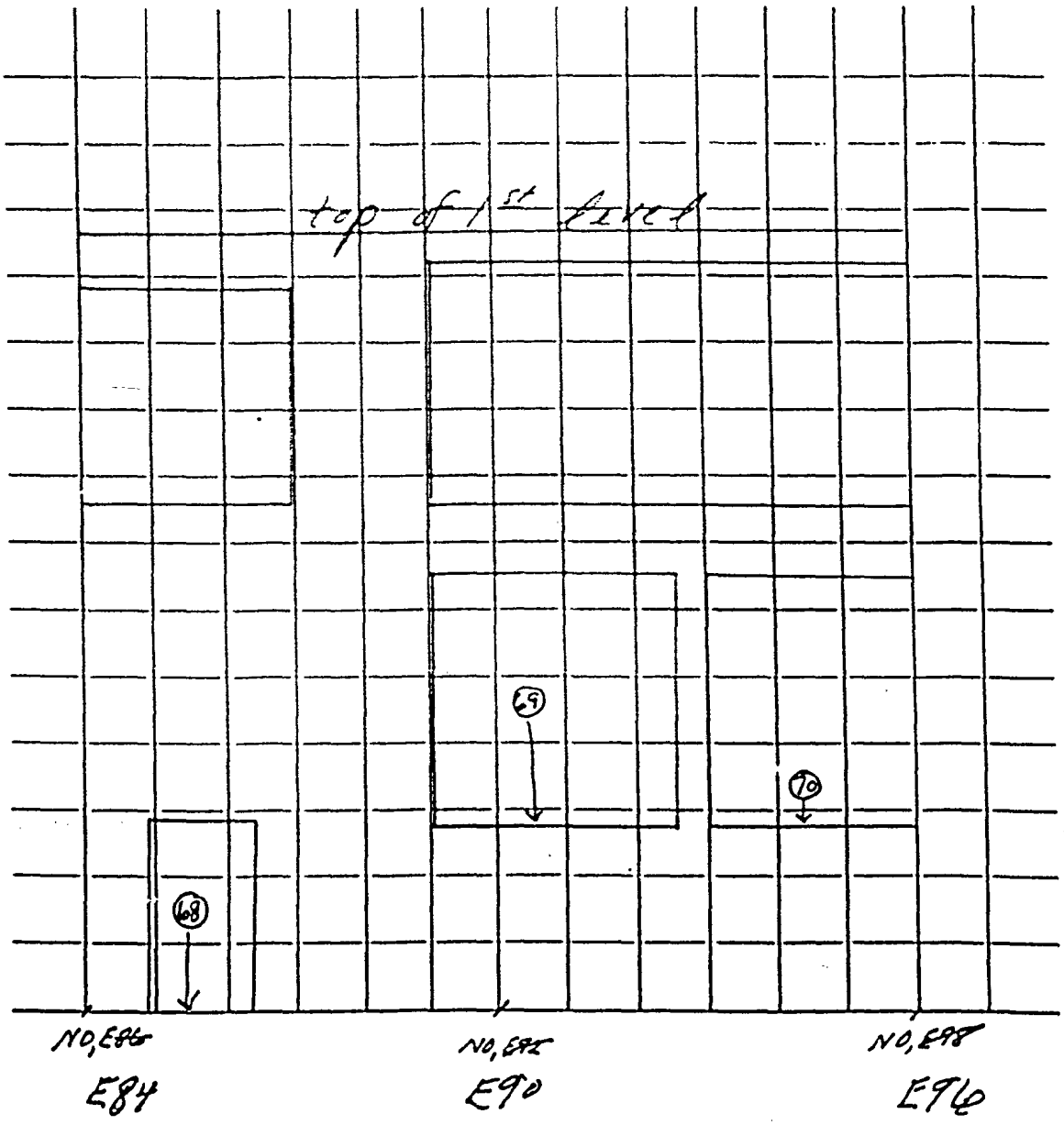


* denotes window

South Exterior Wall

129507

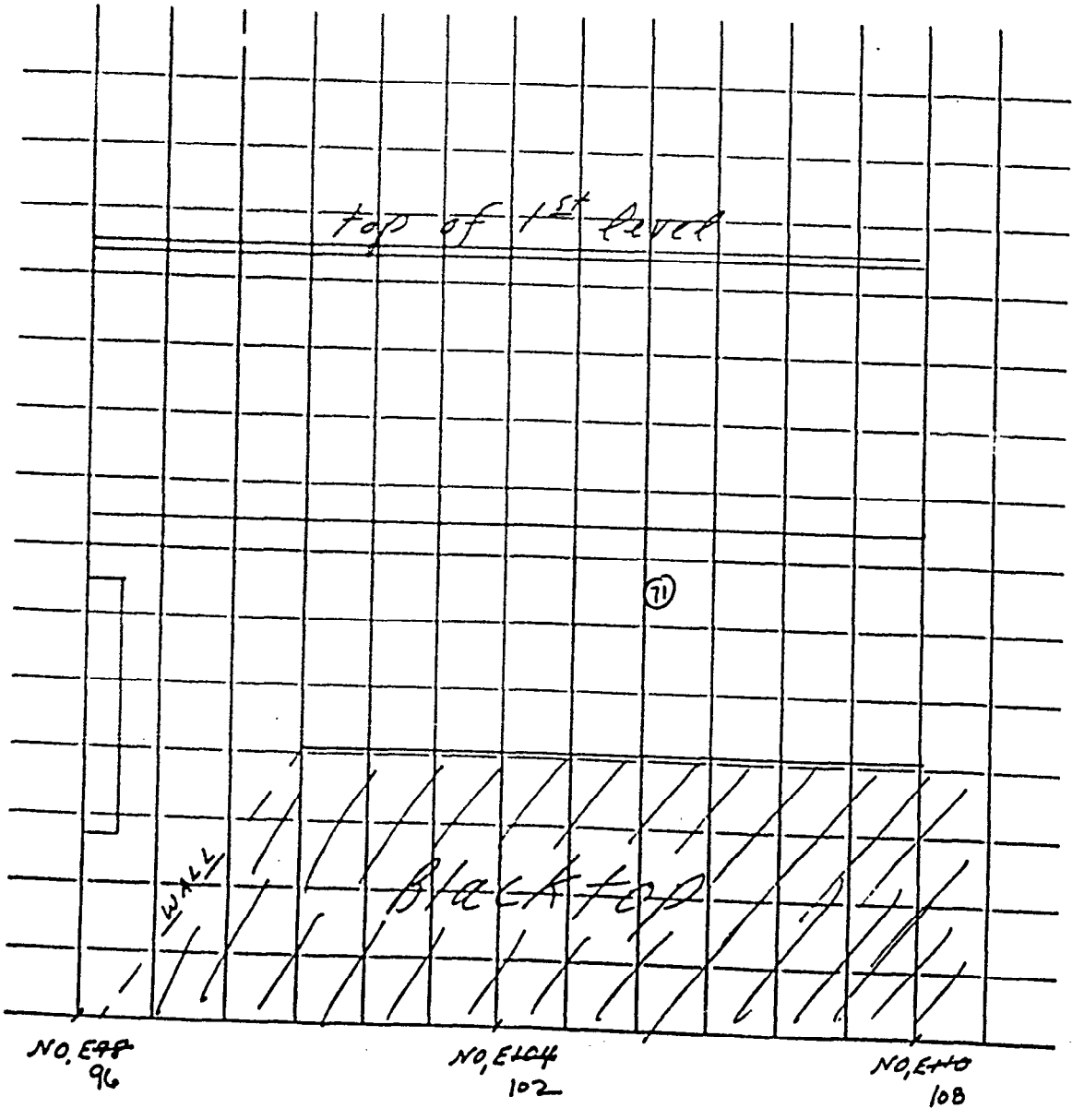
D-19523



South Exterior Wall

129501

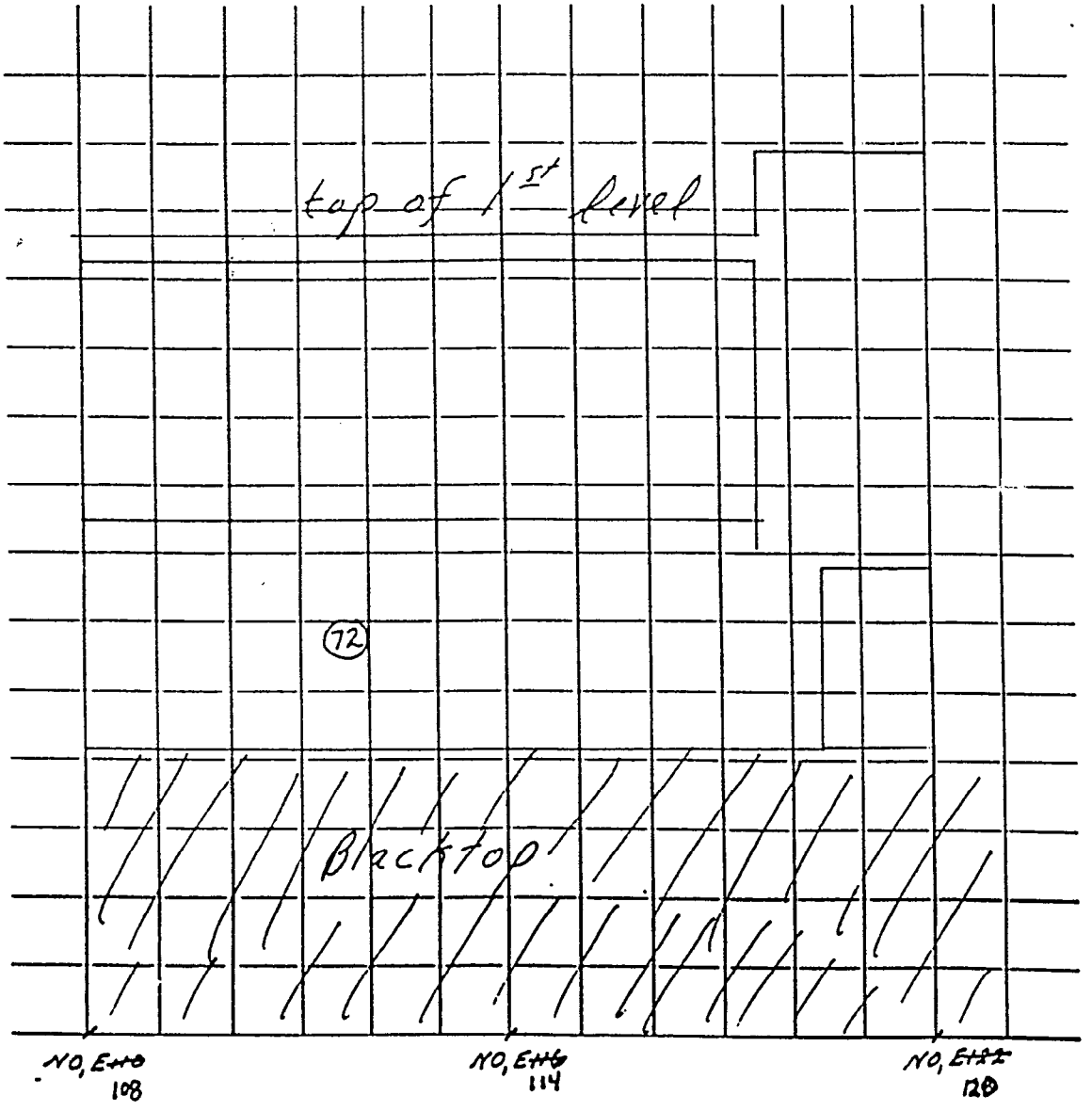
0-19623



South Exterior Wall

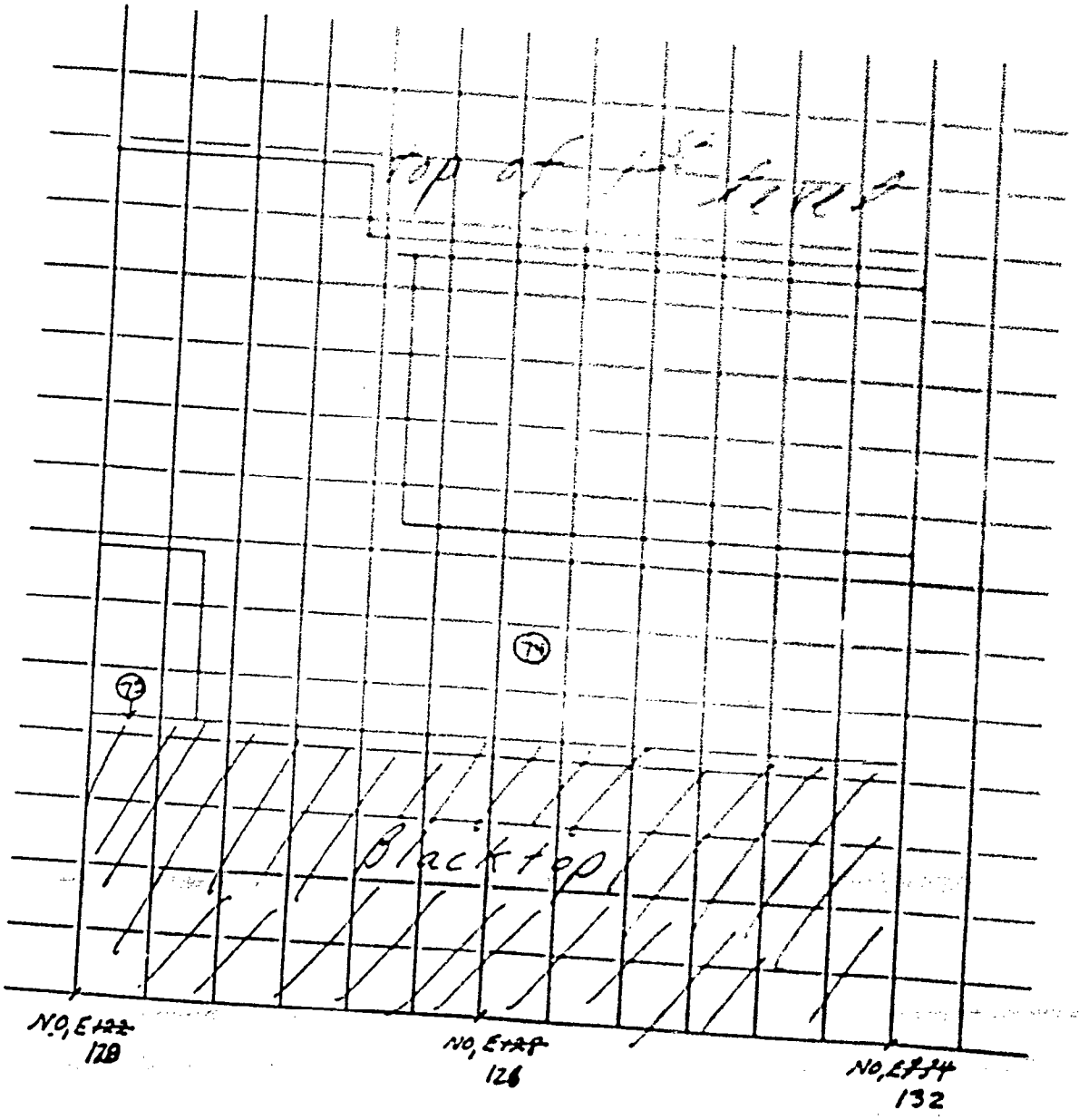
129507

0-19623



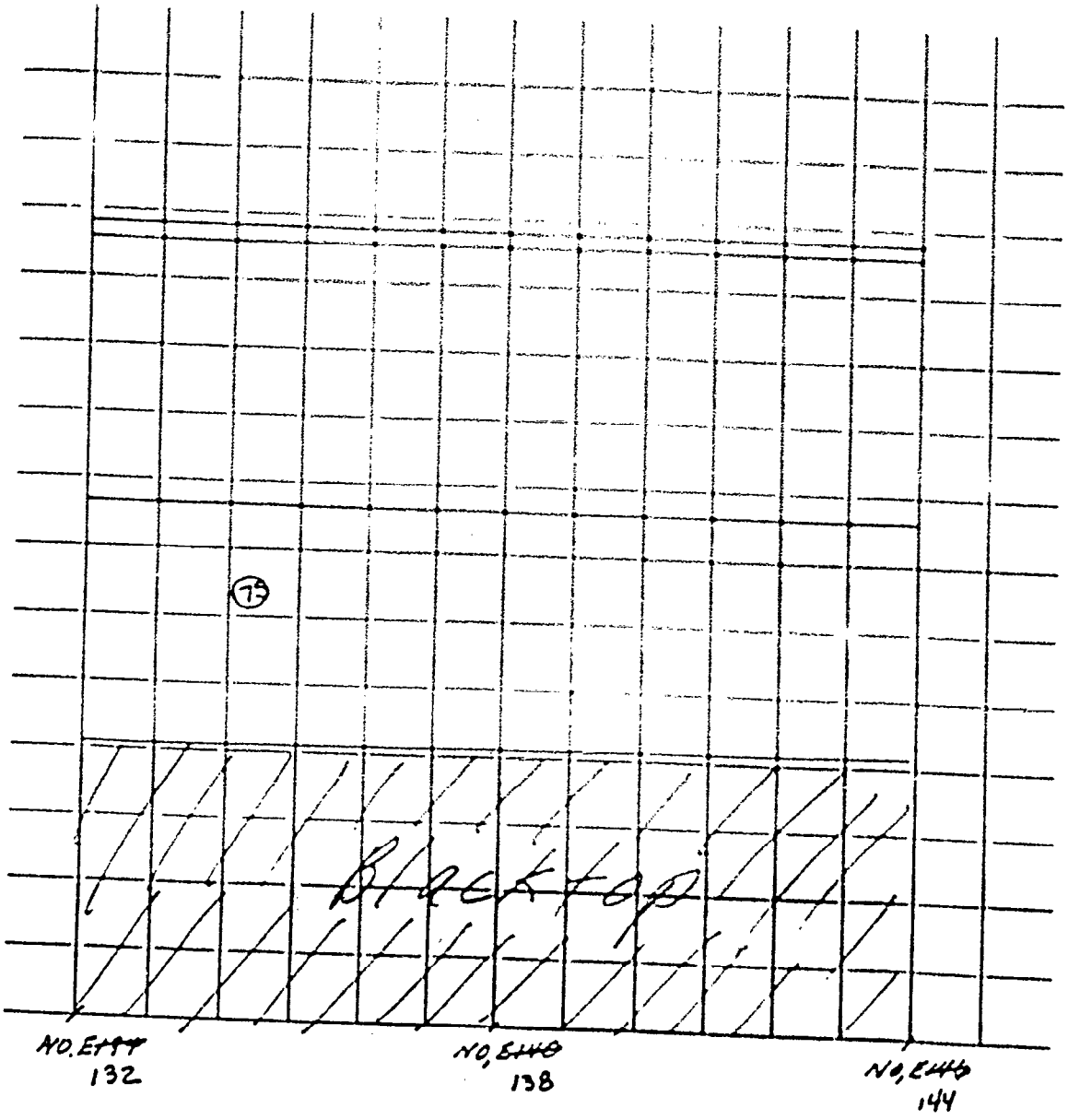
South Exterior Wall

129507



South Exterior Wall

159507



South Exterior Wall

129507

Exterior East Wall Scan Data

EE-WALL.XLS

Location	Location Number	Direct (DPM/100 cm^2) Before Dust Removal Alpha	Before Dust Removal Beta/Gamma	After Dust Removal Alpha	After Dust Removal Beta/Gamma	Transferrable (DPM/100 cm^2) Alpha	Transferrable Beta/Gamma
W	1	-35	869				
a	2	-36	1184				
l	3	-48	1154				
l	4	32	621				

129507

SHEET

(LARGE AREA)

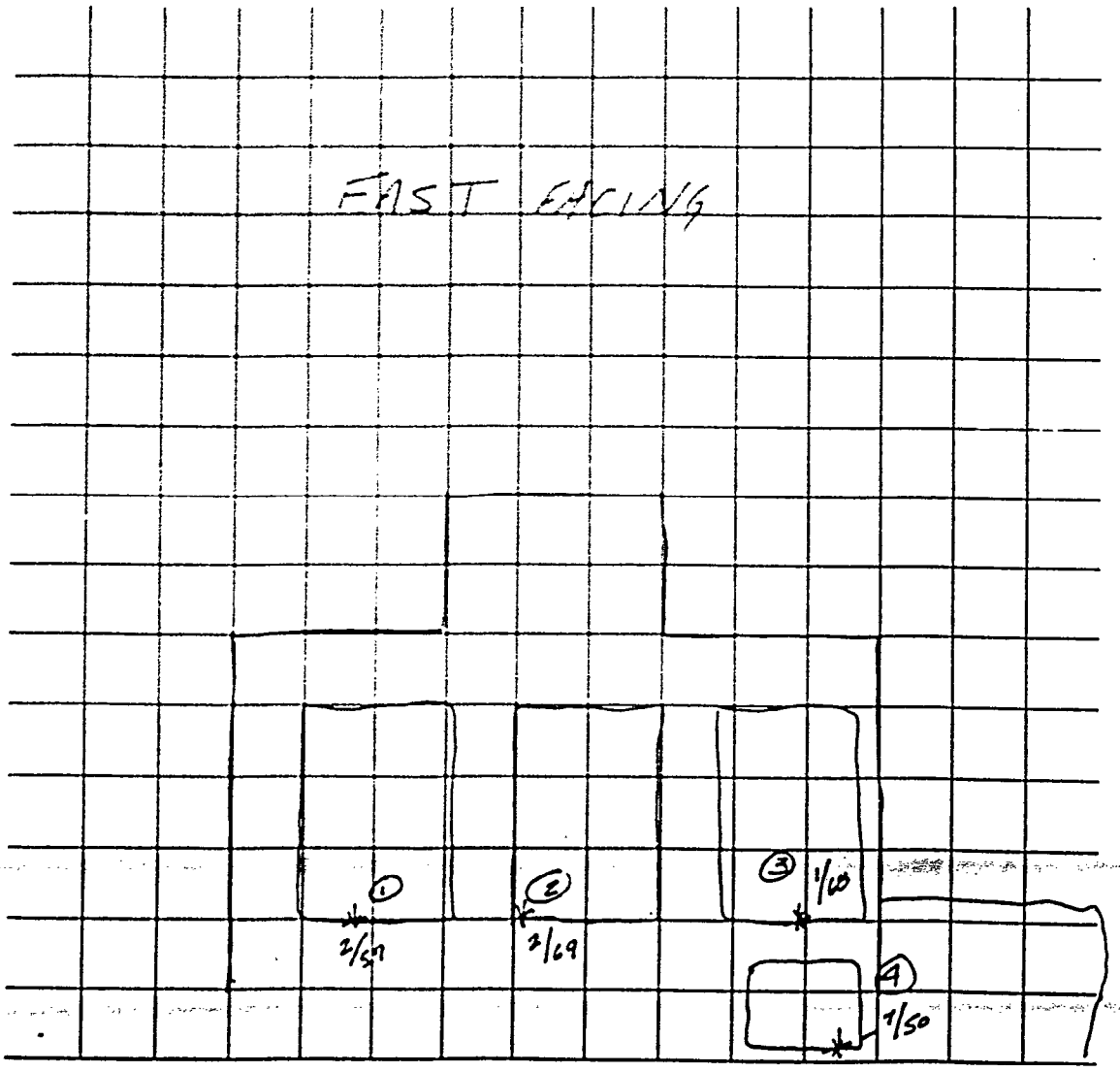
0-3623

: Chapman Valve Date 12-8-94 Technician: J. Davis, M. Awad

ler. _____ Detector _____ S/N: _____

iversion: _____ Bkg _____

Coordinates E <---> W



arks: EXTERIOR WALL

BACK OF BLDG. EAST SIDE BLDG #23

DATA GAP ROOF SAMPLING AND FIDLER SURVEY

ROOFDG.XLS

WEST

R5	R4	R3	R2	R1	
38.2 C	15.3	A	A	10.0	1
55.2 128.2	14.9 89.2	6.4 C	7.3 C	11.0	2
55.4 213.8	13.4 76.7	7.1	7.2	6.6	3
35.8 126.7	9.3	5.9	6.7	4.7	4
21.3	8.2 13.4 C	4.8	6.4 C	4.9	5
9.9 11.1	8.3	4.7	6.2	5.7	6
11.2	7.7 57.7	A	6.9	5.3	7
9.4 50.3	8.0	A	A	5.7	8

NORTH

XX.X = FIDLER readings in CPM x 1,000 taken during data gap effort.
XX.X = U-238 results in pCi/g, taken during data gap effort.
XX.X = U-238 result in pCi/g, taken during characterization.
A = Areas inaccessible due to power lines, etc.
C = Composite U-238 sample (103 pCi/g), taken during characterization.

FIDLER background equal to 7,702 CPM.

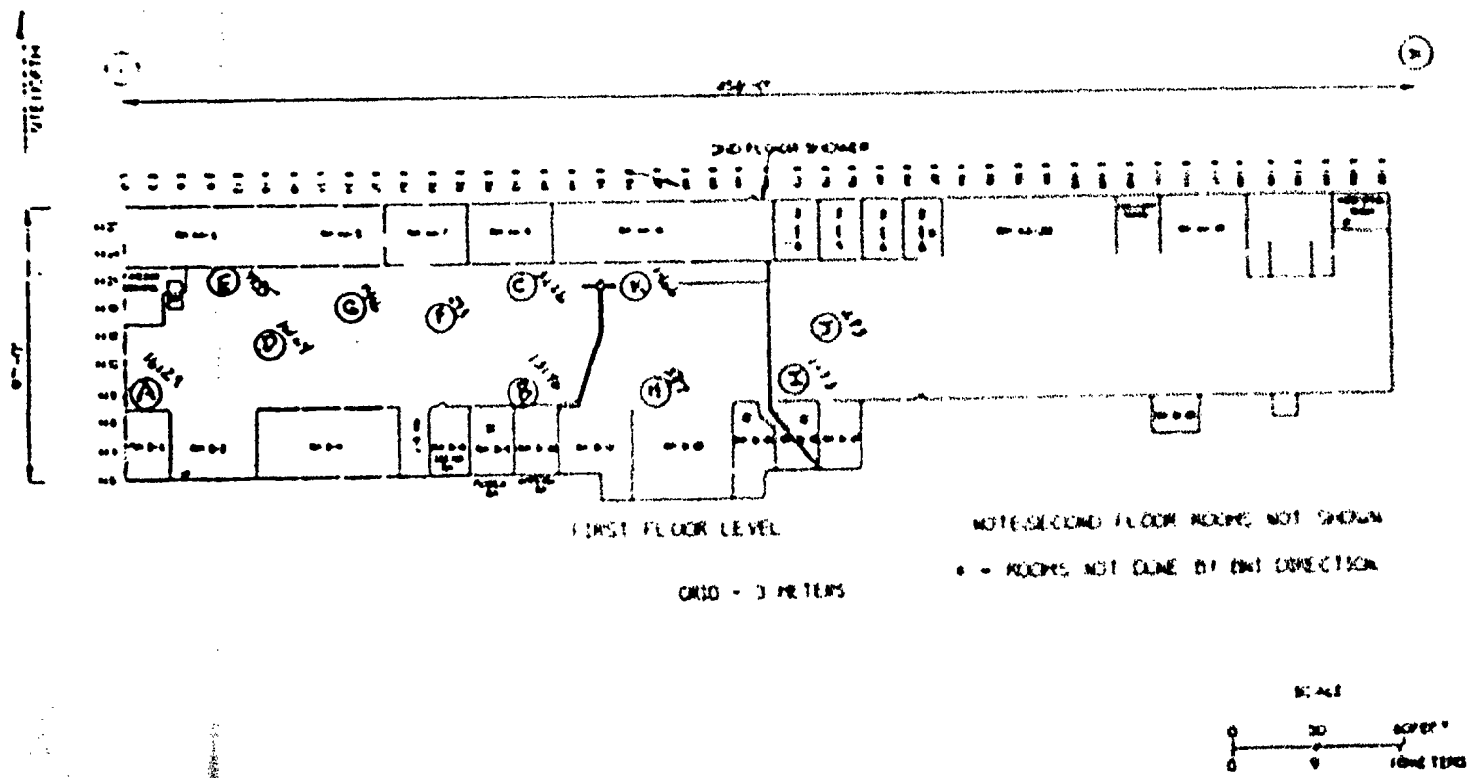
139507

Roof Survey

ROOF.XLS		Direct (DPM/100 cm^2)			
	Location	Before Removal		After Removal	
Location	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma
S	1		5500		33150
u	2		5838		28606
r	3		2487		44745
f	4		2438		31483
a	5		23833		118775
c	6		7879		148722
e	7		2468		140
	8		1565		1882
S	9		8176		624
c	10		11387		1457
a	11		832		
n	12		2230		
S	13		1548		
	14		9680		992
	15		8354		1249
	16		21941		29285
	17		1724		2378
	18		832		89
	19		130		
	20		327		178
	21		30		
C	A		16129		
o	B		13198		
r	C		3226		
e	D		16129		
S	E		4985		
	F		293		
	G		1466		
	H		587		
	I		1173		
	J		293		
	K		1466		

Chapman Valve - Bldg # 23 "Roof Core Locations"

12/16/94
J. Davis
J. Robinson



FILE:
CHAPMAN3

D-19693

208

L9957

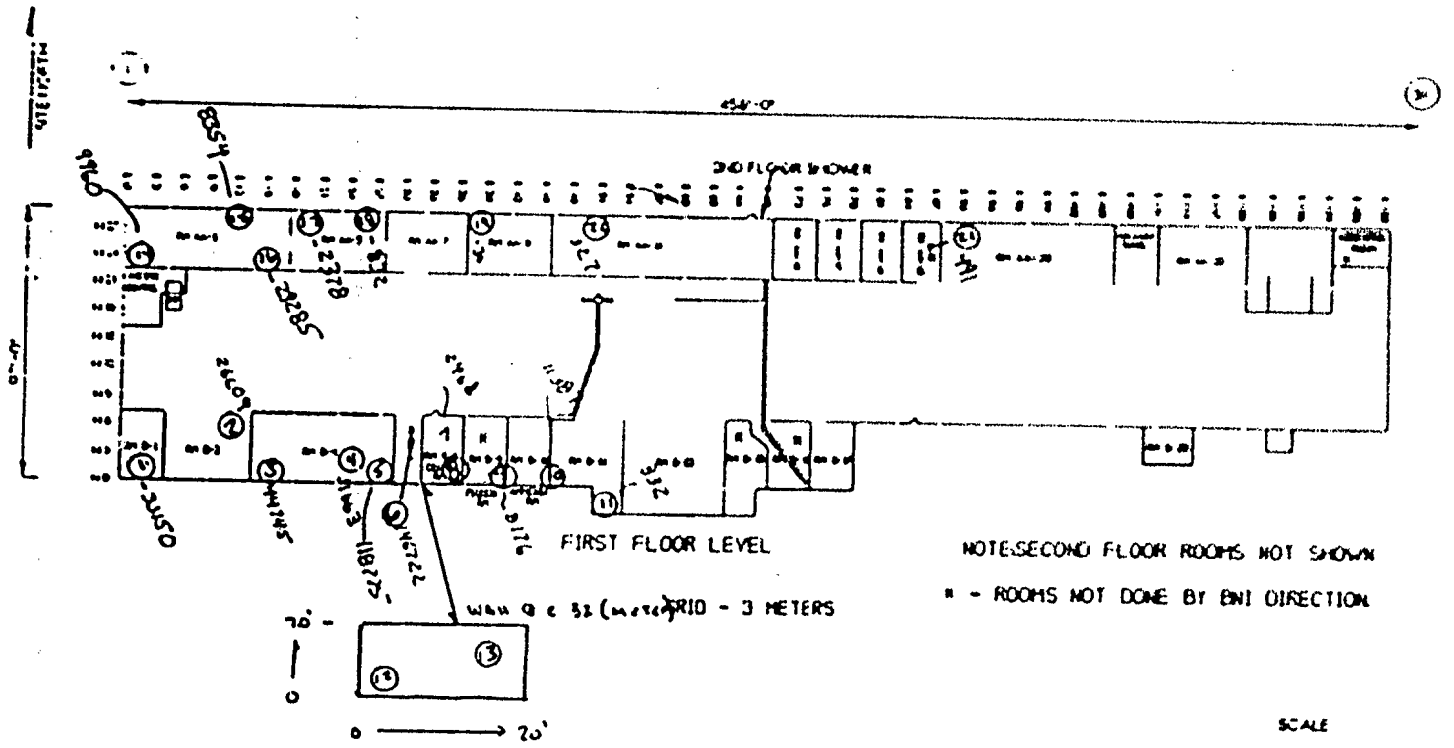
0024

CHAPMAN VALVE BLDG # 23 "ROOF SURVEY"

12-21-94

BY: J. Robinson
J DAVIS

NOTE: LOCATIONS ARE APPROXIMATE



FILE:
CHAPMAN13

D-19693

207

12/9/94

0024

129587

Appendix E

129527

Decontamination Test Data

DECON.XLS		Before Paint Removal		After Paint Removal			
	Location	Direct (DPM/100 cm^2)		Direct (DPM/100 cm^2)		Transferrable (DPM/100 cm^2)	
Test	Number	Alpha	Beta/Gamma	Alpha	Beta/Gamma	Alpha	Beta/Gamma
P	1	149	3315	91	296	2	8
a	2	178	1460	134	1596	2	-25
i	3	120	1805	134	533	2	-33
n	4	164	2190	120	917	2	8
t	5	18	1835	178	296	2	-33
	6	61	2575	178	1213	5	8
	7	324	-148	310	-148	-1	-4
	8	149	1065	266	710	2	25
	9	295	562	207	207	-1	4
	10	178	1332	266	740	-1	-29
	11	193	50015	222	26102	8	8
	12	339	18171	339	6984	24	-21
	1 *	27	3861	204	2238	2	64
	2 *	-2	1754	27	1247	0	11
	3 *	47	135	57	-106		
	4 *	380	14633	1105	7898	17	3
	5 *	253	10574	331	836		
	6 *	283	14174	165	1585	0	11
	7 *	27	570	27	543		
	8 *	37	3953	18	329		
	9 *	8	1102	-2	1054		
	10 *	958	168187	929	4920	11	33

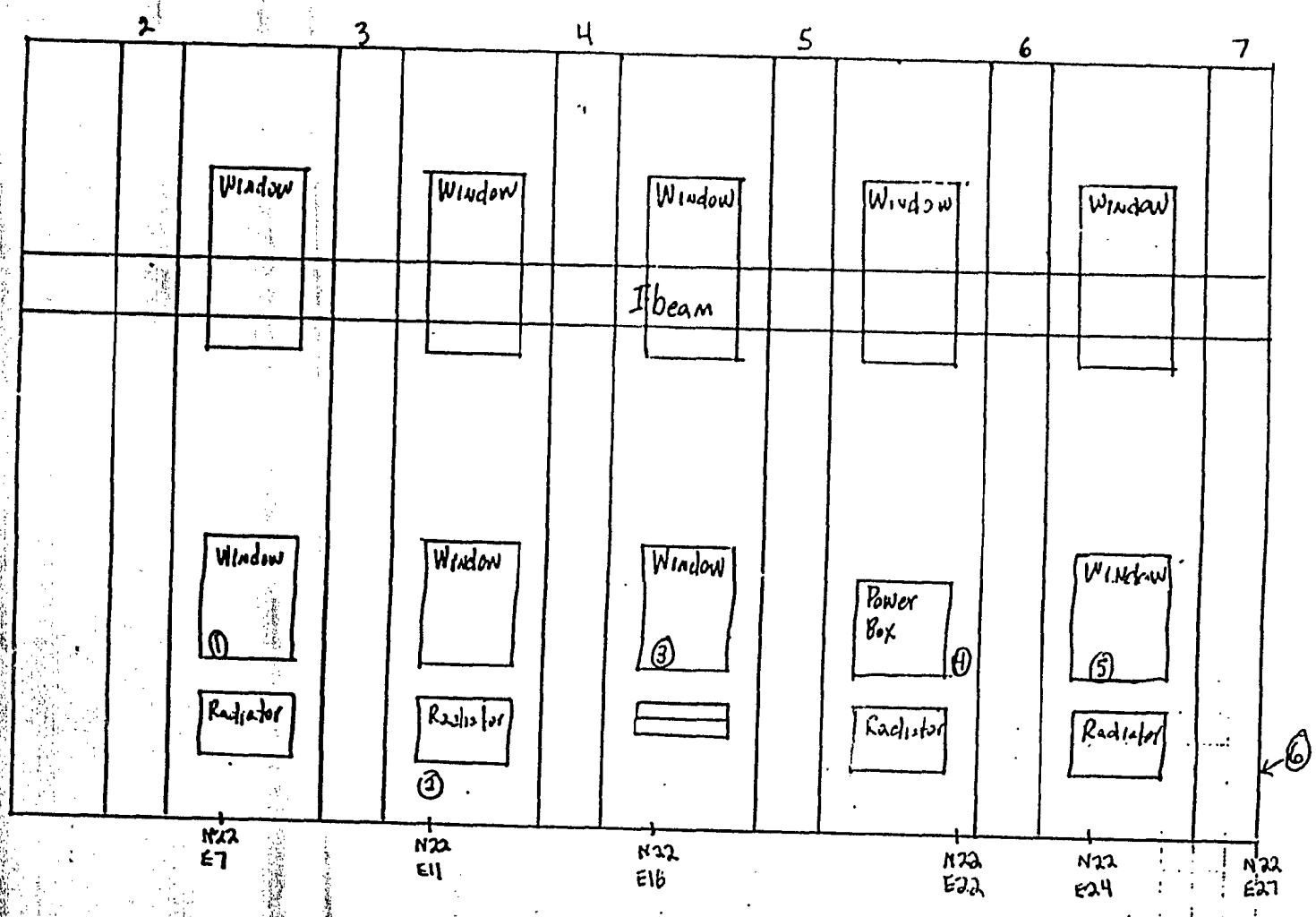
		Before Wire Brushing (DPM/100 cm^2)			After Wire Brushing (DPM/100 cm^2)		
	Truss	Direct	Transferrable		Direct	Transferrable	
Test	Number	Beta/Gamma	Alpha	Beta/Gamma	Beta/Gamma	Alpha	Beta/Gamma
T	2	32377	68	146	1397	5	-13
r	4	6392	10	-4	-59	-1	25
u	6	2408	2	8	59	2	17
s							
s							

* Measurements taken during data gap sampling effort, location coordinates listed below

Location Number	Easting	Northing	Height up Wall (ft)
1	0	19	6
2	0	16	6
3	0	11	4
4	0	9	4
5	0	7	6
6	0	6	6
7	0	5.5	0.5
8	0	2	1
9	0	1	6
10	0	7	N/A

bldg 23 Inwall Interior AA side

PAINT CHIP TEST pg. #1



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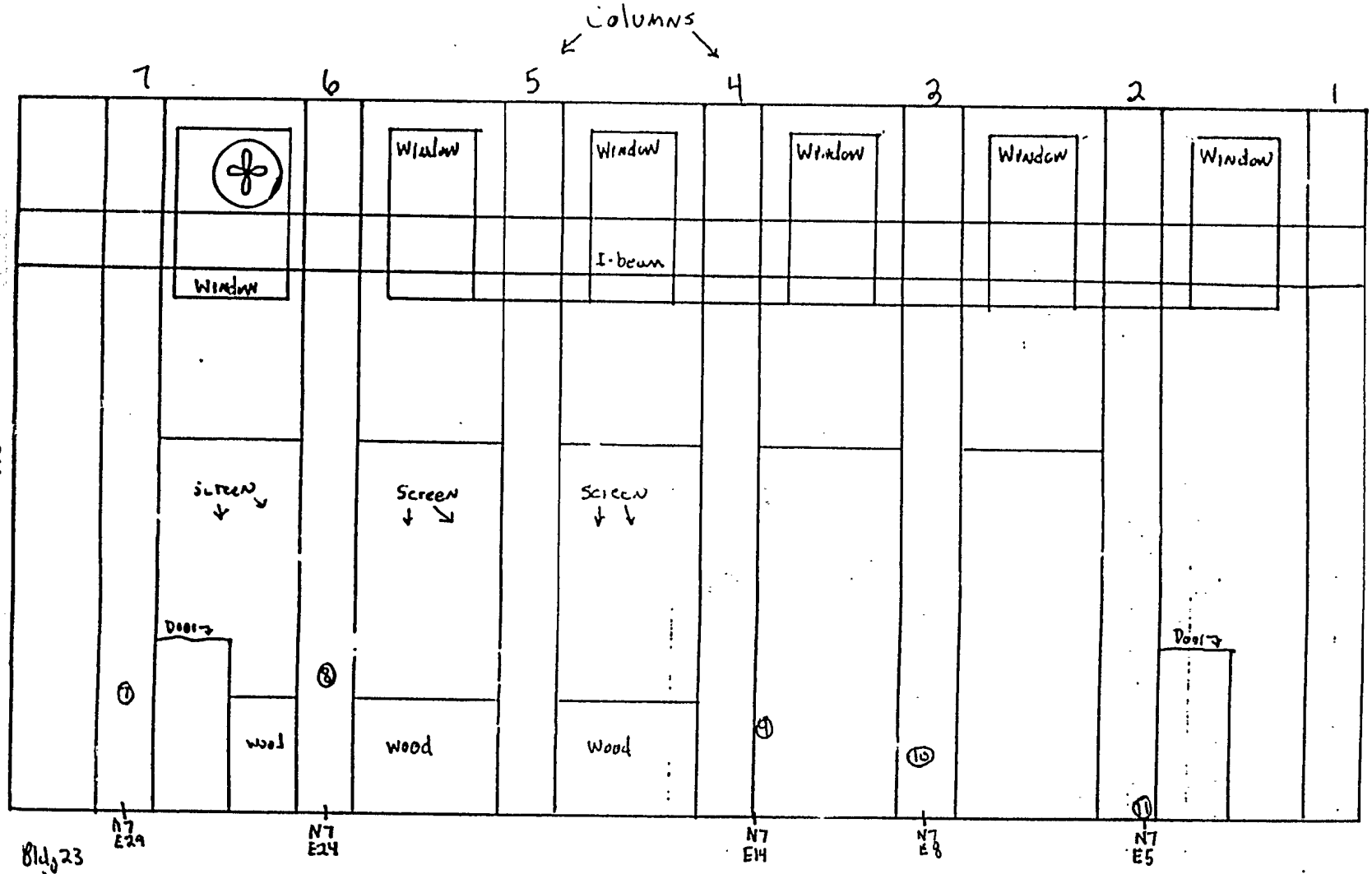
PAINT CHIP TEST pg. #2



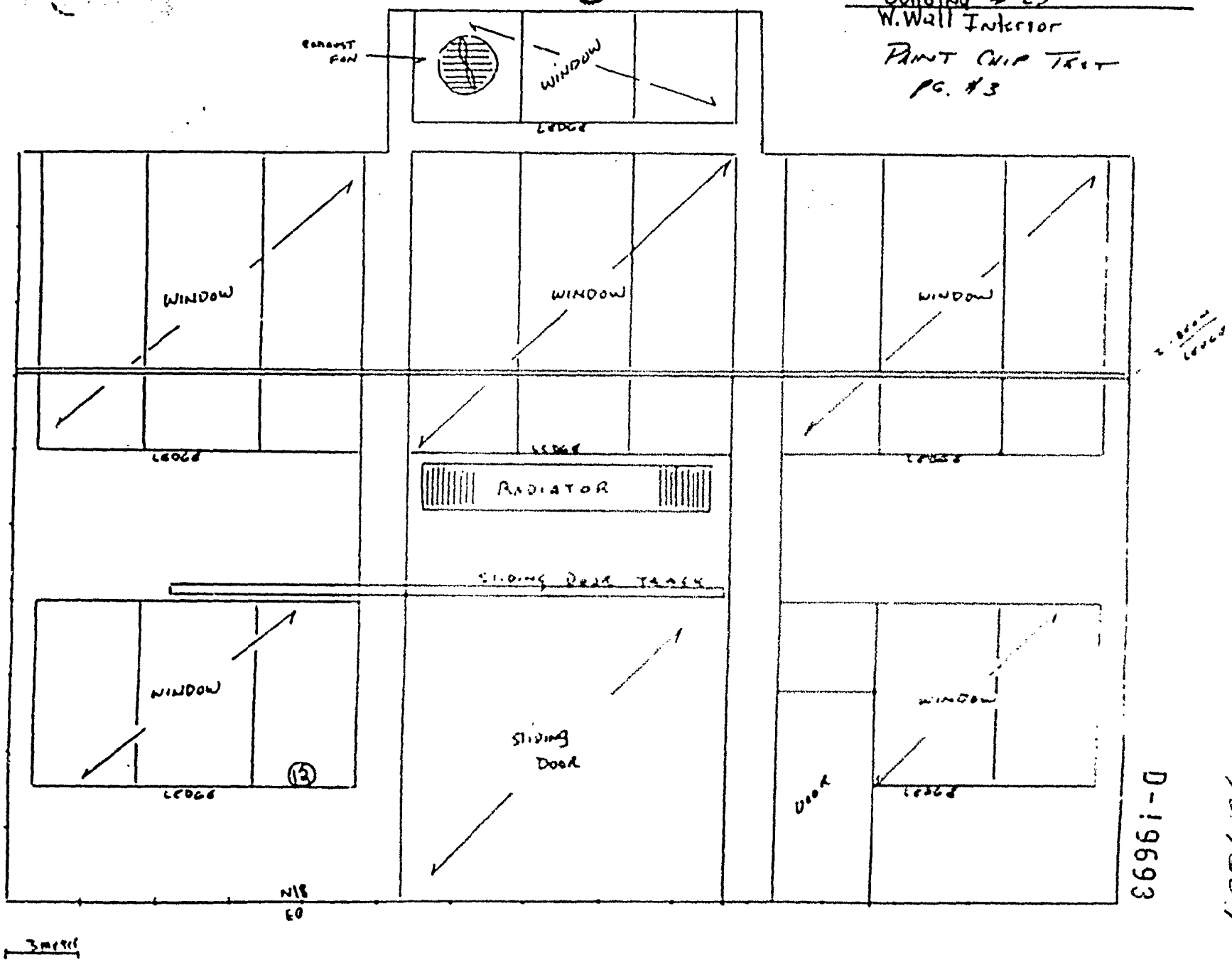
Bldg 23
S. Wall Interior
B Side

211

D-19693

12/95

CHAPMAN VALVE
Building # 23
W. Wall Interior
PAINT CHIP TEST
PG. #3



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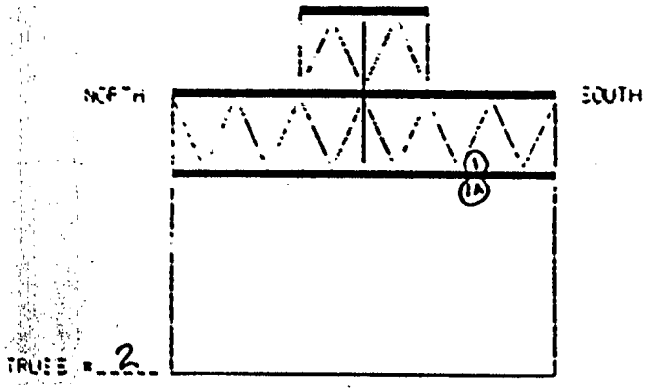
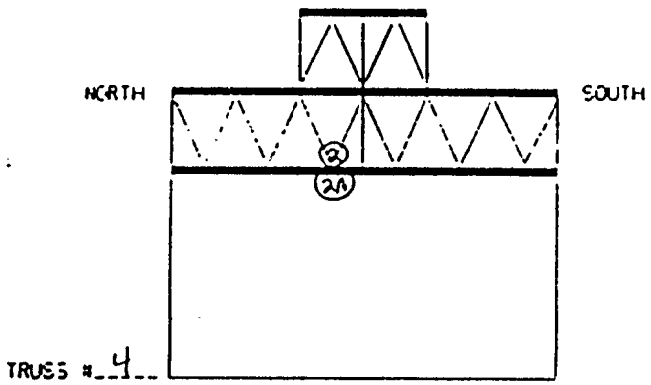
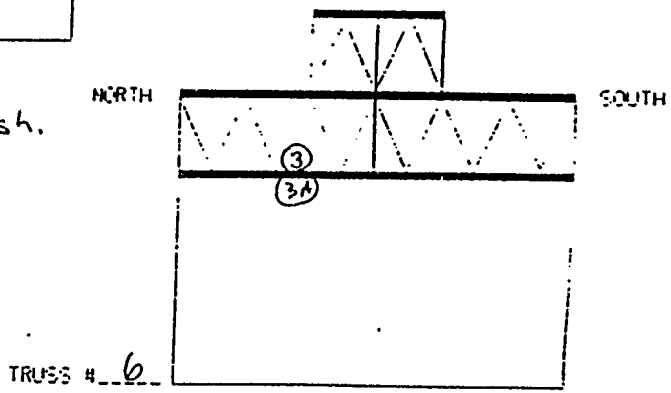
0024

BUILDING #23
NORTH AND SOUTH TRUSS







NOTE * 1, 2, 3 TAKEN PRIOR TO CLEANING
1A, 2A, 3A TAKEN AFTER CLEANING WITH WIRE BRUSH.

189277
0024

098808

ORNL/RASA-92/1

ornl

OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001)

R. D. Foley
M. S. Uziel

MANAGED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

0098808

ORNL/RASA-92/1

HEALTH AND SAFETY RESEARCH DIVISION

Environmental Restoration and Waste Management Non-Defense Programs
(Activity No. EX 20 20 01 0; ADS317O000)

Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001)

R. D. Foley and M. S. Uziel

Date Issued - July 1992

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managed by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U. S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

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ACKNOWLEDGMENTS

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ABSTRACT

At the request of the U.S. Department of Energy (DOE), a team from Oak Ridge National Laboratory conducted a radiological survey at Building 23 (Department No. 40) at the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts. The survey was performed in August 1991. The purpose of the survey was to determine whether the property was contaminated with radioactive residues, principally ^{238}U, as a result of work done for the Atomic Energy Commission (AEC) during the 1940s. The survey included a gamma scan, a beta-gamma scan, and measurement of alpha activity; measurement of direct and removable alpha and beta-gamma levels; and the collection of soil, dust, debris, and smear samples for radionuclide analyses. Survey emphasis was on interior floors, walls, and overhead beams.

Radionuclide analysis of soil, dust, and debris, and analysis of smear samples indicate that residual ^{238}U attributable to former AEC-supported operations is present at this site. Elevated levels of radioactivity were particularly evident on the floors and walls in the western part of the central area of the building (grid blocks A1 through A6). Concentrations of ^{238}U in dust samples collected from overhead beams exceeded DOE guidelines in grid blocks A1 through A14 and remained elevated in grid blocks A15 through A19. Dust on a movable overhead crane in grid block A23 was well above the guideline, probably because the crane had at some time been located further west. Some contamination was evident in grid blocks B1 through B5, but clutter and debris in this area prevented a thorough survey.

Results of the Radiological Survey at the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001)*

INTRODUCTION

The Chapman Valve Manufacturing Company in Indian Orchard, Massachusetts, was one of many companies performing work associated with the development of nuclear energy for defense-related projects during the 1940s. This work, conducted under government contract to the Manhattan Engineer District (MED) and the Atomic Energy Commission (AEC), included the procurement, storage, and processing of uranium oxides, salts, and metals, and the subsequent machining of these products. As a result of these activities, equipment, buildings, and land at some of the sites became radiologically contaminated resulting in low levels of contamination on the properties. At contract termination, sites used by contractors were decontaminated in accordance with the standards and survey methods in use at that time. Since the original assessments, radiological criteria and guidelines for the release of such sites for unrestricted use have become more stringent. In some instances, records documenting decontamination efforts cannot be found, and the final radiological conditions of the site cannot be adequately determined. As a result, the Formerly Utilized Sites Remedial Action Program (FUSRAP) was established in 1974 to identify these formerly used sites and to reevaluate their radiological status.[1]

The radiological survey detailed in this report was performed under the FUSRAP program at the site of the former Chapman Valve Manufacturing Company. Chapman Valve was a principal supplier of regular and special valves and manifolds for the MED and the AEC. Records also indicate that Chapman Valve, under contract with Brookhaven Laboratory, machined uranium metal during the period January through November 1948. In a letter dated January 9, 1948, shipment of approximately 26 tons of extruded rods was directed to a Chapman Valve facility in Indian Orchard, Massachusetts. Machining operations may not have started until May 1948. One AEC memorandum indicates that Chapman Valve may also have conducted rolling operations on the uranium metal. The health and safety program for this work at the Chapman Valve facility was set up by the AEC.[2]

There was a fire in the restricted area of the AEC project, probably in June 1948. The exact nature of the fire and extent of the damage is unknown. The involvement of uranium metal in the fire is suggested by a Chapman Valve letter dated January 19, 1949.[2]

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*The survey was performed by members of the Measurement Applications and Development Group of the Health and Safety Research Division at Oak Ridge National Laboratory under DOE contract DE-AC05-84OR21400.

After the contract work was completed, Chapman Valve had in their possession over 27,000 pounds of metal scrap, oxides, and sweepings. Termination of these operations is indicated in a Chapman Valve letter dated November 8, 1948, which requested termination of AEC film badge services. All radioactive residues and contaminated materials were surveyed by the Brookhaven Medical Group and shipped off-site. The actual shipment date is unknown, but the shipment probably took place in December 1948.[2]

A former Chapman employee recalls that in 1947 Chapman set aside approximately one-third of Department No. 40 at the Chapman site, 203 Hampshire Street, Indian Orchard, Massachusetts, for the machining of uranium rods for Brookhaven Laboratory. The "set aside" portion of the building measured approximately 200 by 60 ft and was separated from the remainder of the building by a floor-to-ceiling wooden partition, which has since been removed. Chapman became part of the Crane Company in 1959. The building that contained the uranium operations has been vacant since Crane discontinued all manufacturing at Indian Orchard early in 1987.[3]

In August 1991, a radiological survey was conducted at the former Chapman Valve Manufacturing Company by members of the Measurement Applications and Development Group of the Oak Ridge National Laboratory (ORNL) at the request of the U.S. Department of Energy (DOE). The survey and sampling at this site covered selected areas outdoors and accessible areas inside Building 23 (formerly Department No. 40, Fig. 1, p. 10). Survey emphasis was on interior floors, walls, and overhead beams.

SCOPE OF THE SURVEY

The radiological survey included (1) a surface gamma scan of the floor and walls in all accessible areas inside the building and a gamma scan of the ground surface in selected areas outdoors; (2) measurement of surface and 1-m gamma exposure rates at the center of the north and south sections of main-bay survey blocks; (3) a beta-gamma scan of dose rates in accessible areas of the floors and walls inside the building; (4) measurement of alpha activity levels at selected locations; (5) collection and radionuclide analysis of 26 dust and debris samples from overhead beams, 1 from a high-wall window sill, 1 from floor debris, and 2 from a concrete floor anchor; (6) measurement of direct and removable alpha and beta-gamma levels at 31 locations inside the building; (7) collection and radionuclide analysis of 2 soil samples outside the building. It was judged unsafe to conduct a survey of the deteriorating roof.

SURVEY METHODS

A comprehensive description of the survey methods and instrumentation used in this survey is given in *Procedures Manual for the ORNL Radiological Survey Activities (RASA) Program*, ORNL/TM-8600 (April 1987).[4]

To facilitate reporting of results, the east-west support columns and overhead beams in Building 23 were numbered 1 to 31 (Fig. 2, p. 11), and the north-south columns were

labeled AA, A, B, and C, dividing the building into a series of survey blocks. The northwest corner of each survey block was designated as the grid block identifier.

Using a NaI scintillation probe connected to a Victoreen ratemeter, surface gamma levels were recorded for accessible areas of the floor and walls inside the building and in selected areas outdoors. The detector was held approximately 2 in. above the floor/ground surface, and measurements were recorded and then converted to μR/h. Using a Geiger-Mueller pancake detector, beta-gamma levels were recorded and then converted to mrad/h. Alpha levels were measured at selected locations with a Bicron ratemeter connected to an ORNL ZnS scintillation probe and were then converted to dpm/100 cm^2.

Dust and debris samples were collected inside the building from areas where residuals may have been deposited; these samples were analyzed for ^{226}Ra, ^{232}Th, and ^{238}U. Smears were obtained from selected surfaces to establish transferable alpha and beta-gamma activity levels. Soil samples collected outdoors to depths of 15 cm were analyzed for ^{226}Ra, ^{232}Th, ^{238}U.

SURVEY RESULTS

DOE guidelines are summarized in Table 1 (p. 30). Typical background radiation levels for Massachusetts are presented in Table 2 (p. 32). These data are provided for comparison with survey results presented in this section. All direct measurement results presented in this report are gross readings; background radiation levels have not been subtracted. Similarly, background concentrations have not been subtracted from radionuclide concentrations measured in soil, dust, and debris samples. Removable radioactivity levels (smears) are reported as net disintegrations per minute (dpm) with background subtracted.

Current photographs of the site are shown in Figs. 3 and 4 (p. 12).

GAMMA EXPOSURE RATE MEASUREMENTS

Results of gamma exposure rate measurements at Building 23 are shown in Figs. 5 and 6 (pp. 13 and 14). Gamma exposure rates at 1 m above the floor in the center of the north and the center of the south sections of grid blocks A1 through A30 (Fig. 5, p. 13) ranged from 5 to 11 μR/h; surface exposure rates at the same points ranged from 5 to 13 μR/h. A gamma scan (Fig. 6, p. 14) of accessible floor and wall surfaces showed that, generally, highest gamma levels were associated with bricks, concrete, and other such naturally radioactive materials used in paving and building construction. Gamma levels in the east end of the building were near typical background levels for Massachusetts (Table 2, p. 32); however, some gamma exposure rates in the west end of the building were definitely elevated: 30 μR/h on the north side of the west entrance, 32 μR/h in grid block A1, and 26 μR/h in grid block A5.

BETA-GAMMA DOSE RATE MEASUREMENTS

Ranges of beta-gamma dose rates measured in grid blocks at Building 23 are shown in Fig. 7 (p. 15). Highest beta-gamma levels were 4 and 2.9 mrad/h found in grid blocks A1 and A2, respectively. Beta-gamma dose rates exceeded background levels of 0.02 to 0.08 mrad/h in grid blocks A1 through A6, A8, A9, A11, A13, B1, B2, and B5.

DIRECT AND REMOVABLE ALPHA AND BETA-GAMMA RADIOACTIVITY LEVELS

Direct alpha and beta-gamma activity levels were measured at 30 locations in grid blocks A1 through A17. Results are given in Table 3 (p. 33). Directly measured alpha activity ranged from below the minimum detectable activity (MDA*) to 2900 dpm/100 cm^2 with 15 locations showing measurable alpha activity. One measurement exceeded 1000 dpm/100 cm^2. Directly measured beta-gamma activity ranged from 0.02 to 4 mrad/h with 15 locations showing measurable beta-gamma activity. Four measurements exceeded the DOE guideline of 1.0 mrad/h in a 100-cm^2 area.

After recording the direct measurements, the locations were smeared to determine if removable activity was present. Results from smear analyses (Table 3, p. 33) indicate no removable beta-gamma activity above the instrument-specific MDA*; removable alpha activity was above the instrument-specific MDA in 4 of the 30 smear samples.

SOIL AND INDOOR DUST AND DEBRIS SAMPLES

Radionuclide analysis was performed on 2 outdoor soil samples and 32 indoor dust and debris samples. Samples locations are shown on Fig. 8 (p. 16), and analytical results are listed in Table 4 (p. 36). Soil concentrations of ^{226}Ra (0.3 and 1.0 pCi/g), ^{232}Th (0.4 and 1.4 pCi/g), and ^{238}U (<1.4 and 1.9 pCi/g) were similar to typical background concentrations for Massachusetts (Table 2, p. 32).

Concentrations of ^{226}Ra, ^{232}Th, and ^{238}U in dust and debris samples ranged from 0.16 to 0.75 pCi/g, 0.11 to 0.8 pCi/g, and 6.0 to 36,000 pCi/g, respectively. (Elevated concentrations of uranium sometimes mask the true concentration of ^{226}Ra; in these cases, concentrations of ^{226}Ra are reported as less than (<) the minimum concentration that can be detected by the analytical procedure in the presence of elevated uranium.) Highest concentrations of ^{238}U (36,000 pCi/g) were found in dust deposited inside a concrete floor anchor located underneath the wooden flooring of grid block A5. Uranium-238 concentrations of 240 and 120 pCi/g were found in floor debris samples (M11 and M31); chips from the brick facing of column B2 (M32) contained 900 pCi/g ^{238}U. Uranium-238 concentrations in dust samples from overhead beams were converted from pCi/g to dpm/100 cm^2 in order to compare these values with DOE guideline values for removable

*The instrument-specific minimum detectable activities (MDAs) for directly measured and removable alpha radiation levels are 25 and 10 dpm/100 cm^2, respectively. For directly measured and removable beta-gamma radiation levels the MDAs are 0.01 mrad/h and 200 dpm/100 cm^2, respectively.

surface contamination (Table 1, p. 30). Table 5 (p. 38) shows that 17 of 23 dust samples from overhead beams contained ^{238}U at concentrations ≥100% of the DOE guideline of 1000 dpm/100 cm^2 for removable surface contamination (Table 1, p. 30).

Two dust samples, M10 and M31, were selected for uranium isotopic analysis to determine if the uranium content was normal or enriched. Normally, uranium contains 0.711% ^{235}U. Uranium in which the 235 isotope concentration has been artificially increased above the normal level is called enriched uranium. Sample M10 contained the normal percentage of ^{235}U; sample M31 had been slightly enriched to 2.16% ^{235}U.

Radiation measurements prior to and after collection of dust and debris samples are given in Table 6 (p. 39). Frequently, alpha radiation levels, and in some cases beta-gamma radiation levels, remained elevated after sample collection, showing that scraping away the dust or debris did not completely eliminate all contamination.

DISTRIBUTION OF CONTAMINATION

Radiological contamination in Building 23 was concentrated in the westernmost 15 grid areas, primarily in grid blocks A1 through A15. This central area of the building was relatively free of debris and clutter, allowing good access to the area. Only a few spots of contamination were found on the floors and walls, but overhead beams and horizontal surfaces were consistently elevated. Some contamination was evident in the westernmost B grid blocks, but clutter and debris in this area prevented a thorough survey. Anomalies identified in this area (Fig. 7, p. 15) included a radiator on the south wall of grid block B1 with beta-gamma dose rates of 0.1 to 0.4 mrad/h; a spot on the floor at column C2 with beta-gamma dose rates of 0.2 mrad/h; elevated beta-gamma dose rates in the northwest corner of grid block B2; and a circular area on the floor near the southeast corner of grid block B5 with beta-gamma dose rates of 0.3 mrad/h. No anomalies were identified in grid blocks AA1 through AA10. Highlights of the survey of the central part of the building are presented in Figs. 9 through 19 (pp. 17 through 27).

Grid Block A1 and Building Exterior

On the west end of the building, contamination was identified on the building exterior surface, underneath a former window approximately 4 ft south of the main entrance and 1 ft above the ground. At this location (Fig. 20, p. 28), beta-gamma dose rates measured 0.7 mrad/h and maximum alpha activity reached 2900 dpm/100 cm^2. Smear 17 from this area demonstrated that the contamination was not removable (Table 3, p. 33).

Other details of the survey of grid block A1 are diagramed in Fig. 9 (p. 17). (Solid lines represent the floor area; extended dashed lines represent the walls.) North of the concrete ramp immediately inside the main entrance, an elevated area (Fig. 21, p. 28) contained gamma exposure rates of 32 μR/h and beta-gamma dose rates up to 1.3 mrad/h. Debris sample M31 collected at this location was slightly enriched (2.16% ^{235}U) and contained 120 pCi/g ^{238}U (Table 4, p. 36). Beta-gamma dose rates of 1.5 mrad/h were found on the wall and on a ledge just north of the main entrance (Fig. 22, p. 29). The west wall south of

the main entrance had beta-gamma dose rates up to 4 mrad/h and alpha levels of 650 dpm/100 cm^2. Smear 11 (Table 3, p. 33) from this area showed that 36 dpm/100 cm^2 of the alpha activity was removable. Beta-gamma levels near column B2 reached 2.9 mrad/h. Beta-gamma dose rates in grid block A1 consistently exceeded the DOE guideline of 0.2 mrad/h averaged over not more than 1 m^2 and 1.0 mrad/h in any 100-cm^2 area (Table 1, p. 30). Dust samples from overhead beams (M3–M6, Table 5, p. 38) contained 170 to 340 pCi/g (2000 to 5300 dpm/100 cm^2) ^{238}U, which is 2 to 5 times the DOE guideline for residual removable surface contamination (Table 1, p. 30).

Grid Block A2

Beta-gamma dose rates in grid block A2 (Fig. 10, p. 18) reached 2.9 mrad/h on the brick facing of column B2 (Fig. 23, p. 29); alpha levels were 300 dpm/100 cm^2. Radiological analysis of chips of the brick facing material (M32, Table 4, p. 36) demonstrated the presence of 900 pCi/g ^{238}U. Smear 13 (Table 3, p. 33) from this area showed that 32 dpm/100 cm^2 of the alpha activity was removable. Dust from a high-wall window sill located between columns B2 and B3 contained 1300 pCi/g (M30, Table 4, p. 36). Dust from an overhead beam in grid block A2 also contained 1300 pCi/g ^{238}U (1400 dpm/100 cm^2, M29, Table 5, p. 38), which was 1.4 times the DOE guideline for removable alpha activity (Tables 1 and 5, pp. 30 and 38).

Other areas of elevated beta-gamma radiation were observed around column B3 (>0.2 mrad/h), column A2 (1.4 and 1.9 mrad/h), and column A3 (>0.2 mrad/h). Spots on the floor measured 1.2 mrad/h (beta-gamma) and 28 dpm/100 cm^2 (alpha).

Grid Block A3

In grid block A3 (Fig. 11, p. 19), columns B3 and B4 showed beta-gamma dose rates of 0.4 and 0.1 mrad/h, respectively. Beta-gamma dose rates on the radiator and behind the radiator on the north wall reached 0.8 mrad/h; alpha levels reached 120 dpm/100 cm^2. Smears 14, 15, and A3 from these areas (columns B3, B4, and the north wall) did not contain removable radioactivity (Table 3, p. 33) above the instrument-specific MDA. Beta-gamma dose rates of 0.2 mrad/h were measured on top of an electrical box and in an elevated area near the base of column A4; dose rates of 0.6 mrad/h were measured on the wall below the radiator near the floor. Dust samples from overhead beams (M7–M9, Table 5, p. 38) demonstrated the presence of ^{238}U at concentrations of 48 to 130 pCi/g (1500 to 4000 dpm/100 cm^2), which is 150% to 400% of the DOE guideline for residual removable surface contamination (Table 1, p. 30).

Grid Block A4

On the north wall of grid block A4 (Fig. 12, p. 20), beta-gamma dose rates on the brick behind the radiator reached 0.1 mrad/h; alpha radiation levels measured 110 dpm/100 cm^2. Beta-gamma dose rates of 0.2 mrad/h were measured at several places on column A5 and at the floor and column join of column B5. Smear A5 (Table 3, p. 33) from column A5 did not exhibit removable radioactivity above the MDA.

Grid Block A5

In grid block A5 (Fig. 13, p. 21), a spot on the floor near column A5 showed gamma exposure rates of 14 to 26 μR/h, alpha activity levels of 2900 dpm/100 cm^2, and beta-gamma dose rates up to 0.2 mrad/h. Beta-gamma levels on the radiator on the north wall ranged from 0.2 to 0.5 mrad/h; three spots near column A6 measured 0.4 mrad/h, and a contaminated region at column A6 ranged from 0.05 to 0.2 mrad/h. Smear A5 from the north wall and smear 16 from column B5 (Table 3, p. 33) did not contain removable radioactivity above the MDA.

Samples M1 and M2 mark the location of a concrete floor anchor that was probably used to anchor machinery. Dust collected from this location (M1, Table 4, p. 36) contained 36,000 pCi/g ^{238}U. Dust from two overhead beams (M10, M14, Table 5, p. 38) contained 25 and 35 pCi/g (1000 and 1200 dpm/100 cm^2) ^{238}U, which is 100% and 120% of the DOE guideline for residual removable surface contamination (Table 1, p. 30).

Grid Block A6

Grid block A6 (Fig. 14, p. 22) showed elevated beta-gamma levels on the north wall. Contaminated spots on column A6 and around the radiator ranged from 0.2 to 0.6 mrad/h. The brick behind the radiator ranged from 0.02 to 0.1 mrad/h. Smear A6 (Table 3, p. 33) from the wall behind the radiator contained 90 dpm/100 cm^2 removable alpha activity. Firebrick in the center of the floor exhibited beta-gamma levels of 0.02 to 0.1 mrad/h. Dust samples M12 and M13 (Table 5, p. 38) from overhead beams contained 330 and 60 pCi/g (9000 and 2600 dpm/100 cm^2) ^{238}U, which is 9 and 2.6 times the DOE guideline for residual removable surface contamination, respectively.

Grid Block A7 and A8

Elevated gamma exposure rates of 12 and 16 μR/h in grid block A7 and A8 (Figs. 15 and 16, pp. 23 and 24) are typical of naturally occurring radioactive substances present in bricks, concrete, and other such materials used in building construction. Beta-gamma dose rates in these grid blocks generally ranged from 0.02 to 0.08 mrad/h on the walls and floor. Alpha activity measured less than the MDA (<25 dpm/100 cm^2) on low walls. Smears A7 and A8 contained no removable radioactivity above MDA. Dust sample M15 (Table 5, p. 38) from an overhead beam contained 350 pCi/g (12,000 dpm/100 cm^2) ^{238}U, which is 1200% of the DOE guideline for residual removable surface contamination (Table 1, p. 30).

Grid Block A9

In grid block A9 (Fig. 17, p. 25), an area of contamination on the floor near column B10 showed beta-gamma dose rates of 0.02 to 0.5 mrad/h and alpha levels of 43 dpm/100 cm^2. Composite floor debris collected at this location (M11, Table 4, p. 36) contained 240 pCi/g ^{238}U. Two additional spots of slightly elevated alpha activity were noted on the floor (57 and 77 dpm/100 cm^2). Beta-gamma activity on the floor ranged from 0.02 to 0.08 mrad/h, and gamma levels reached 16 μR/h in an area where firebrick was present. Dust samples (M16, M17, and M18, Table 5, p. 38) collected from overhead beams

contained 16, 57, and 65 pCi/g (900, 1300, and 2800 dpm/100 cm^2) ^{238}U, which is 90%, 130%, and 280%, respectively, of the DOE guideline for residual removable contamination (Table 1, p. 30).

Grid Block A11

An area with spots of elevated surface beta-gamma activity ranging from 0.1 to 0.8 mrad/h was identified in grid block A11 (Fig. 18, p. 26) on the west side of the furnace (Fig. 4, p. 12). Similar spots on the east side of the furnace ranged from 0.1 to 0.2 mrad/h. Overhead beams in the block showed beta-gamma activity of 0.1 mrad/h, and an electrical conduit on the north wall measured 0.3 mrad/h.

Grid Block A13

Surface gamma exposure rates, beta-gamma dose rates, and alpha activity levels for the floors and lower walls in grid block A13 (Fig. 19, p. 27) were near typical background levels for Building 23. Dust sample M19 (Table 5, p. 38) from an overhead beam contained 50 pCi/g (1400 dpm/100 cm^2) ^{238}U, which is 1.4 times the DOE guideline residual removable surface contamination (Table 1, p. 30).

SIGNIFICANCE OF FINDINGS

Radionuclide analysis of soil, dust, and debris, and analysis of smear samples collected at Building 23 from the former Chapman Valve facility, Indian Orchard, Massachusetts, indicate that residual ^{238}U attributable to former AEC-supported operations is present at this site. Elevated levels of radioactivity were particularly evident on the floors and walls in the western part of the central area of the building (grid blocks A1 through A6), but contamination on overhead beams was not limited to this area. Concentrations of ^{238}U in dust samples collected from overhead beams exceeded DOE guidelines in grid blocks A1 through A14 and remained elevated but below guidelines in grid blocks A15 through A19. Dust on a movable overhead crane in grid block A23 was well above the guideline, probably because the crane had at some time been located further west. A floor debris sample collected just north of the main entrance in grid block A1 contained uranium that had been slightly enriched (from 0.711% to 2.16% ^{235}U). Some contamination was evident in grid blocks B1 through B5, but clutter and debris in this area prevented a thorough survey. No anomalies were identified in grid blocks AA1 through AA10.

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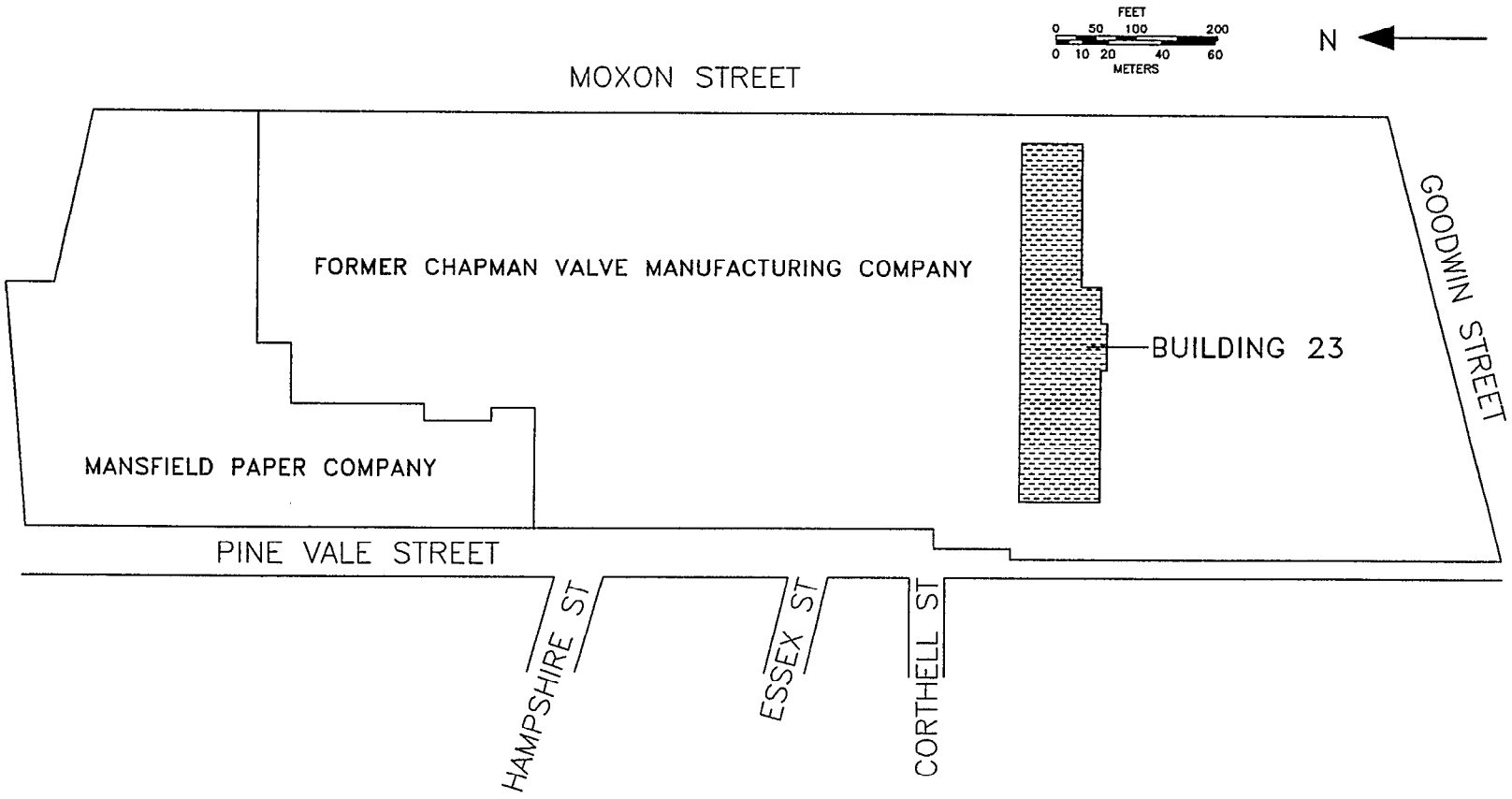

Fig. 1. Location of Building 23 (Department No. 40) at the site of the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts.

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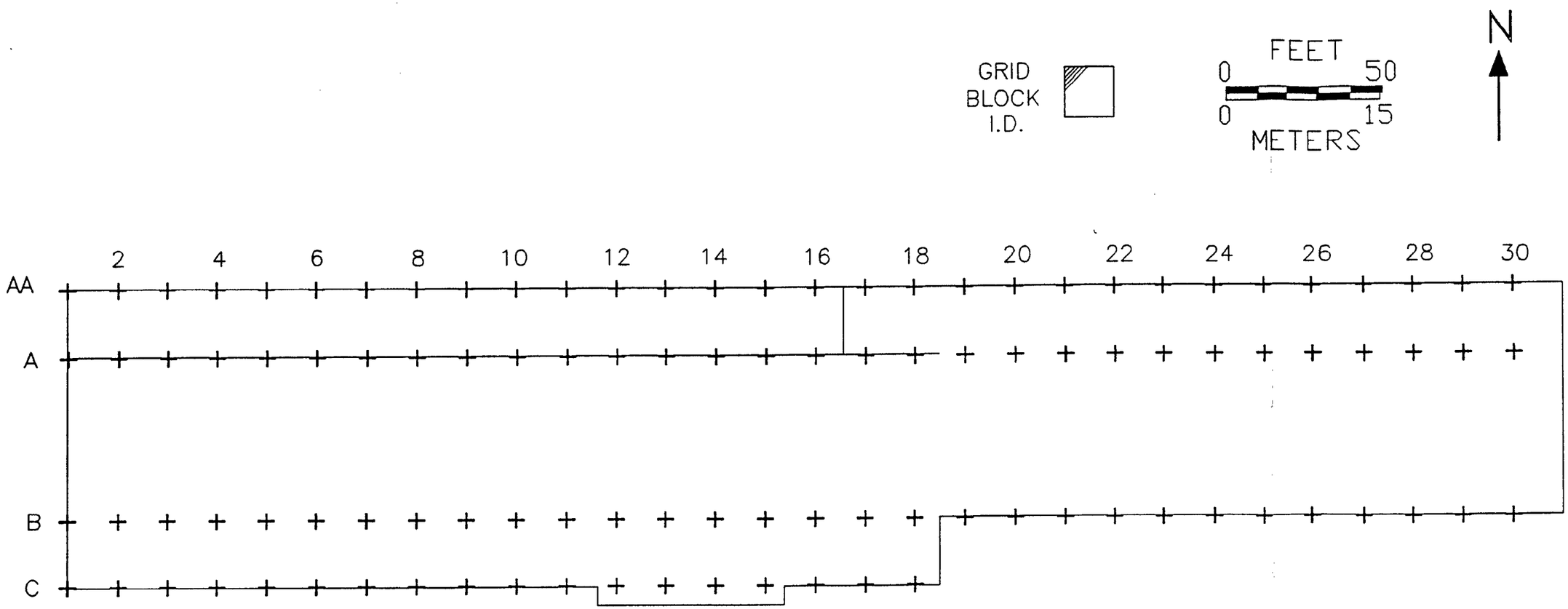

Fig. 2. Building 23 at the site of the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts. To facilitate reporting of results, the east-west support columns and overhead beams were numbered 1 to 31 and the north-south columns were labeled AA, A, B, and C.

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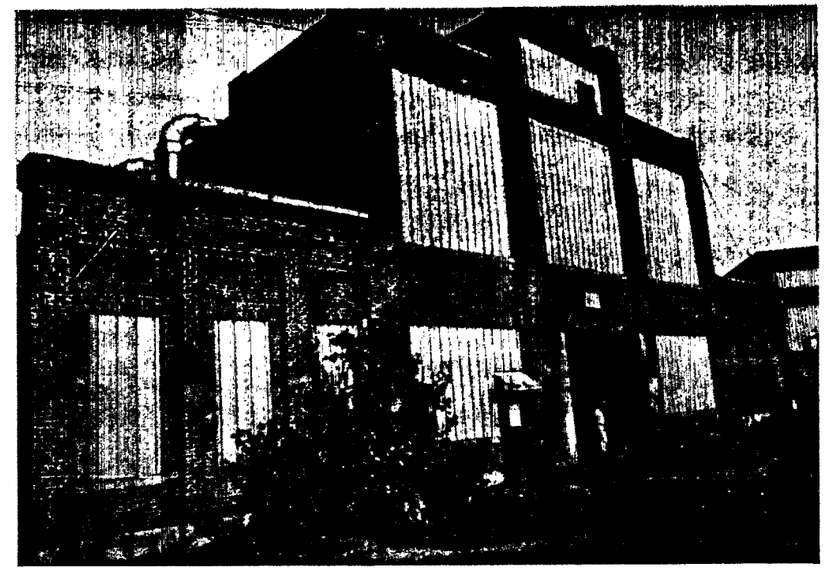

Fig. 3. View looking east at the main entrance to Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

ORNL-PHOTO 1141-92

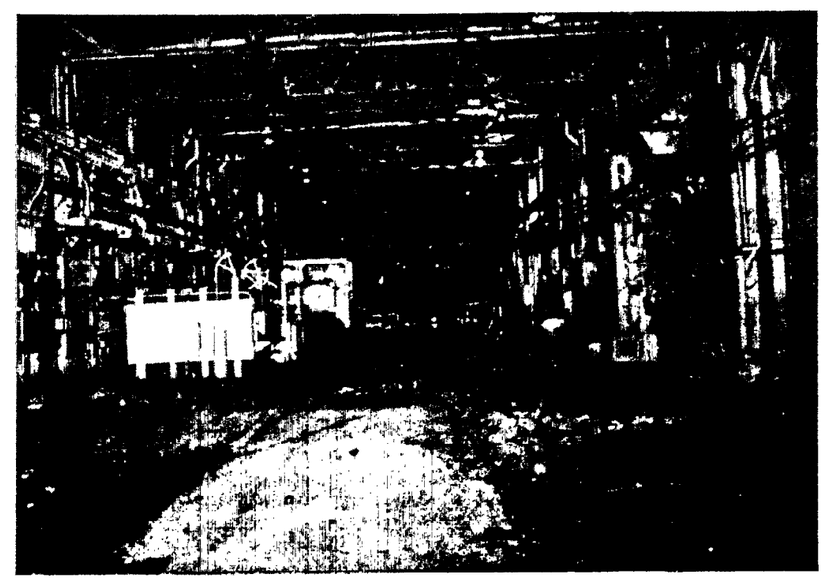

Fig. 4. View inside the main entrance looking east at grid blocks A1 through A30 in the center of Building 23 (Indian Orchard, Massachusetts). Columns labeled B are on the right; those labeled A are on the left. Dust samples were collected from overhead beams visible at the top of the photo.

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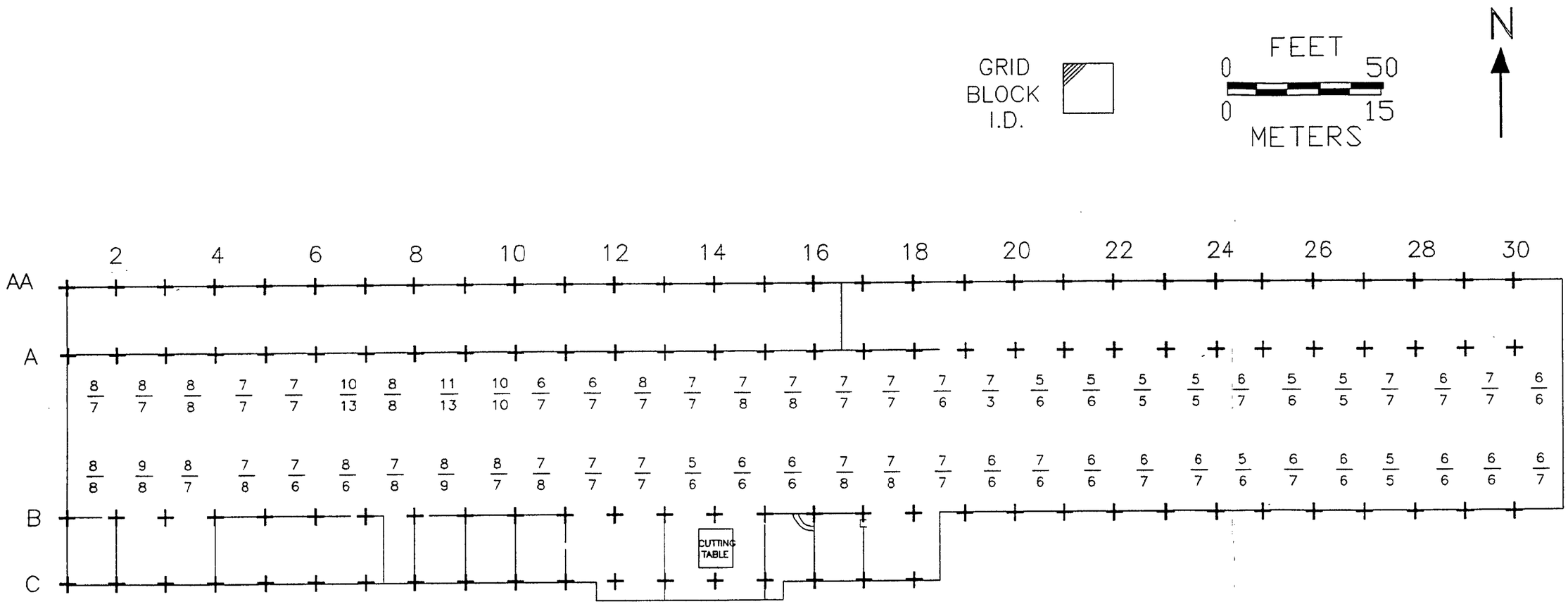

Fig. 5. Surface and 1-m gamma exposure rates (μR/h) taken at the center of the north and south sections of grid blocks A1 through A30, Building 23, Indian Orchard, Massachusetts. Lower number = surface; upper number = 1 m.

ORNL-DWG 92-6730

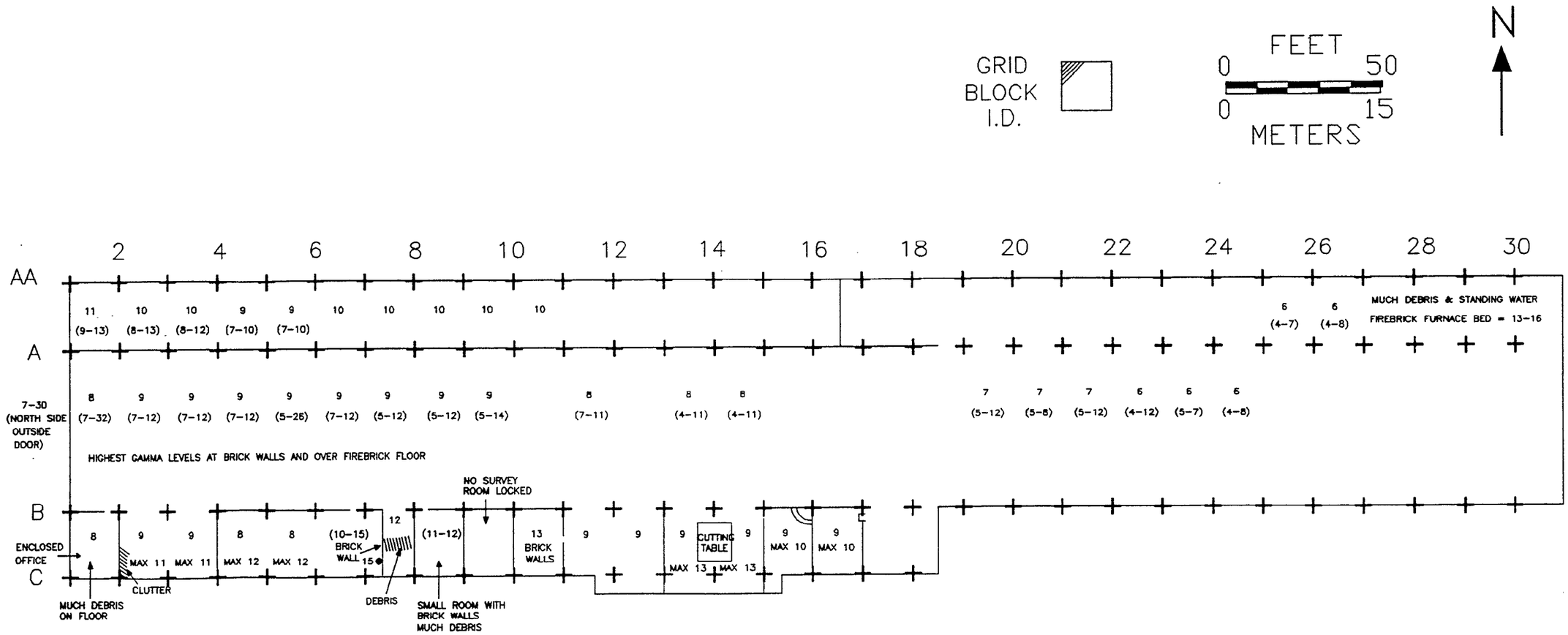

Fig. 6. Average surface gamma exposure rates (μR/h) measured in grid blocks at Building 23, Indian Orchard, Massachusetts. Ranges are given in parentheses.

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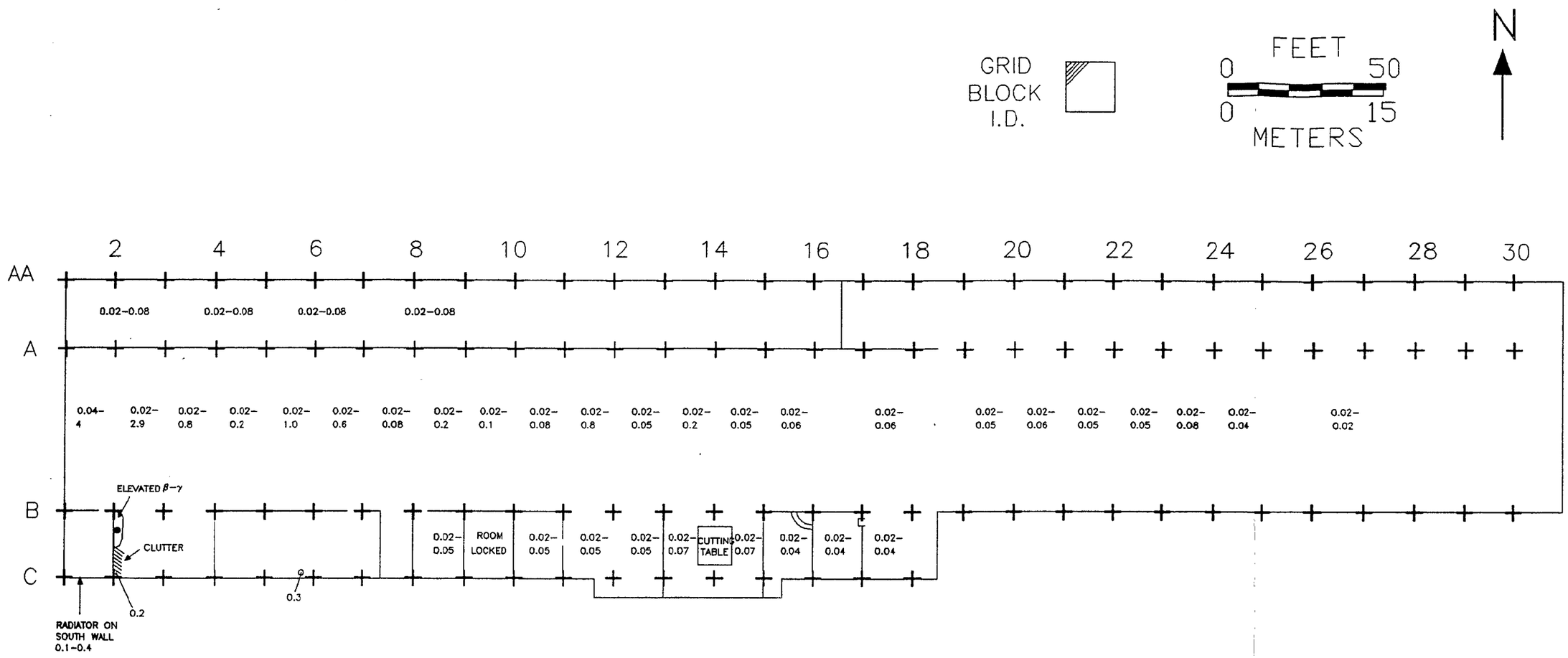

Fig. 7. Ranges of beta-gamma dose rates (mrad/h) measured in grid blocks at Building 23, Indian Orchard, Massachusetts.

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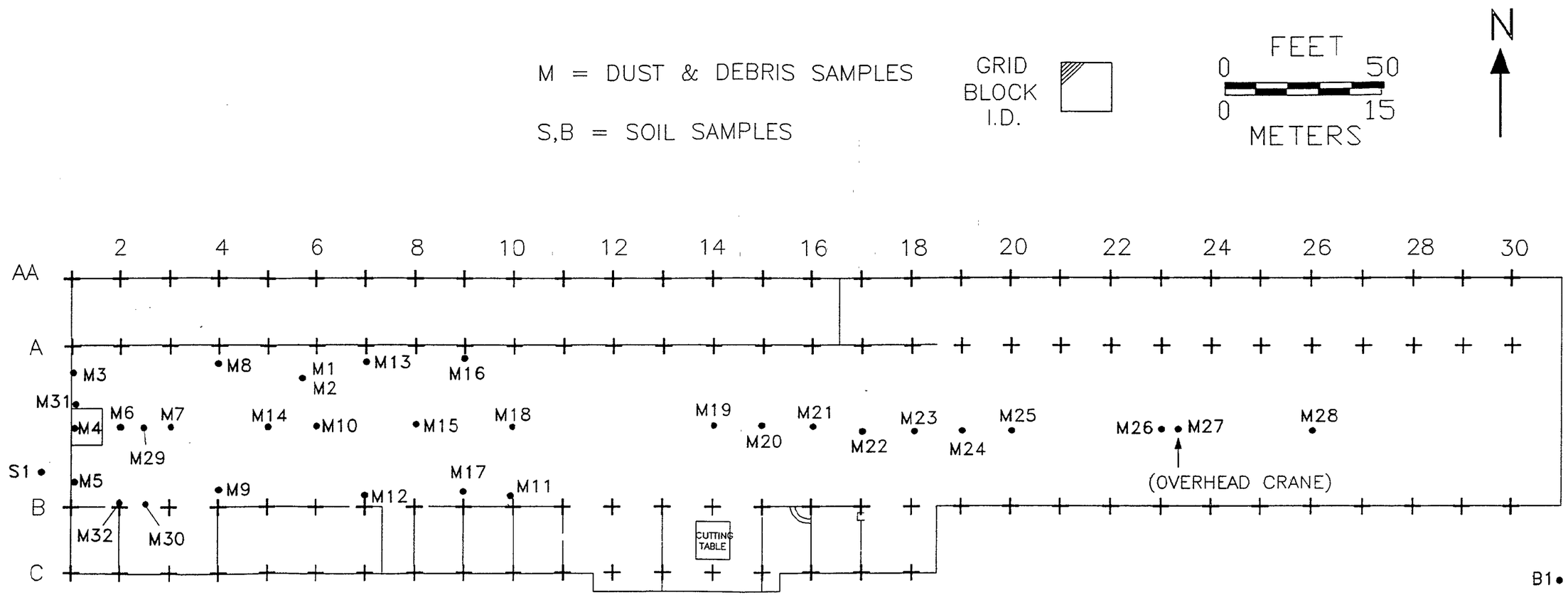

Fig. 8. Locations of soil, dust, and debris samples collected at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

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ORNL-DWG 92-6733

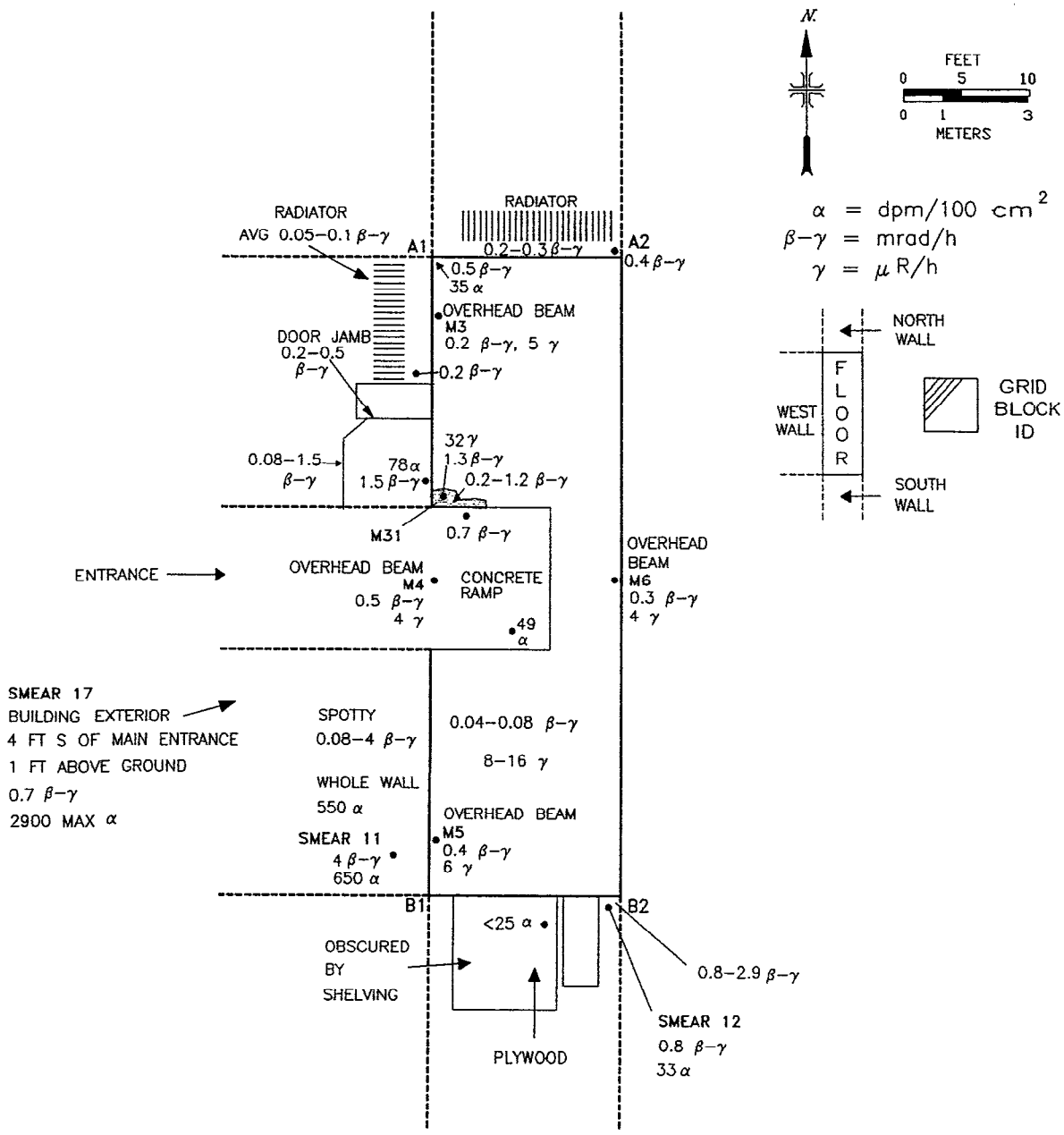

Fig. 9. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm²) in grid block A1.

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ORNL-DWG 92-6734

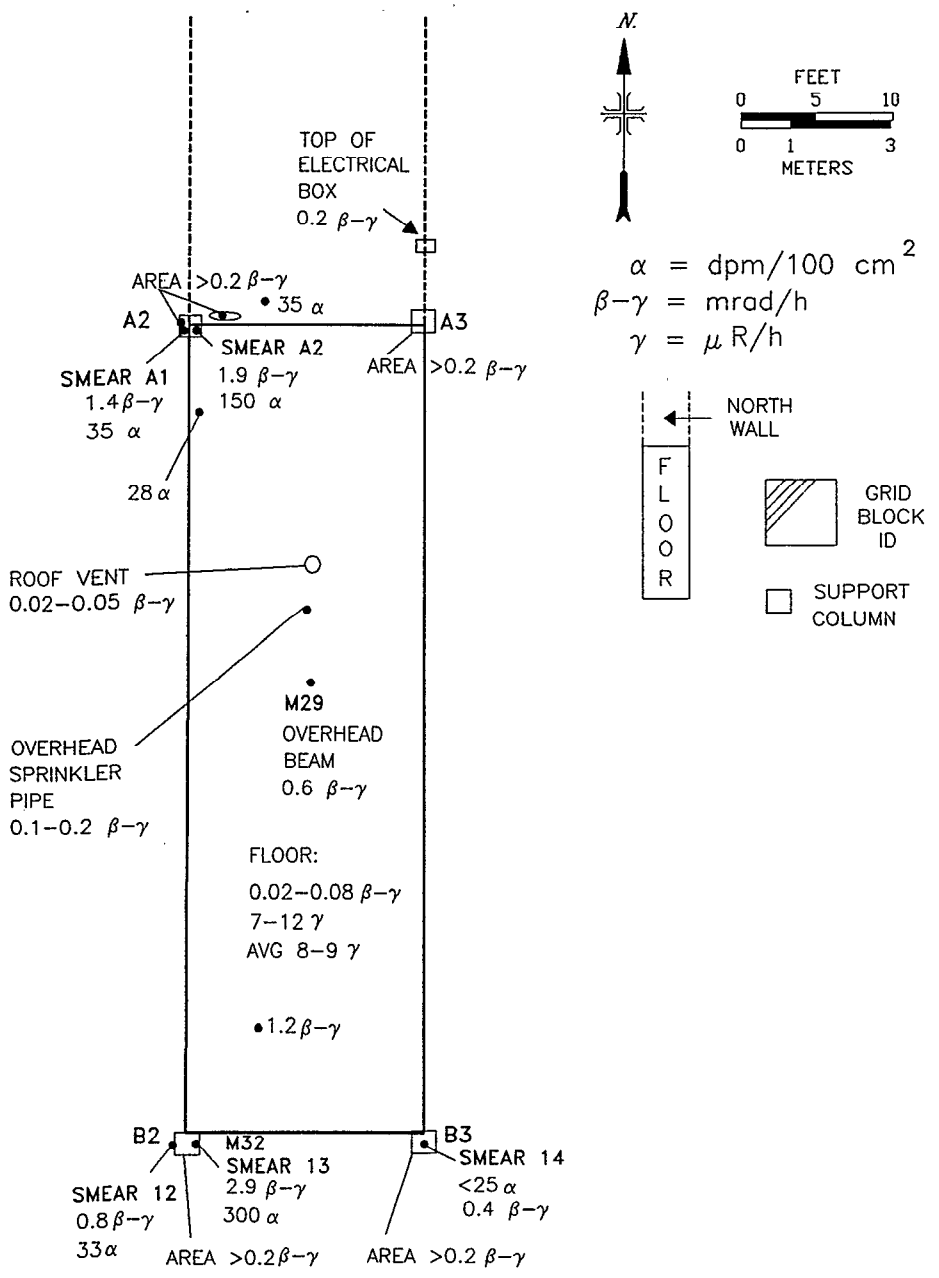

Fig. 10. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm²) in grid block A2.

ORNL-DWG 92-6735

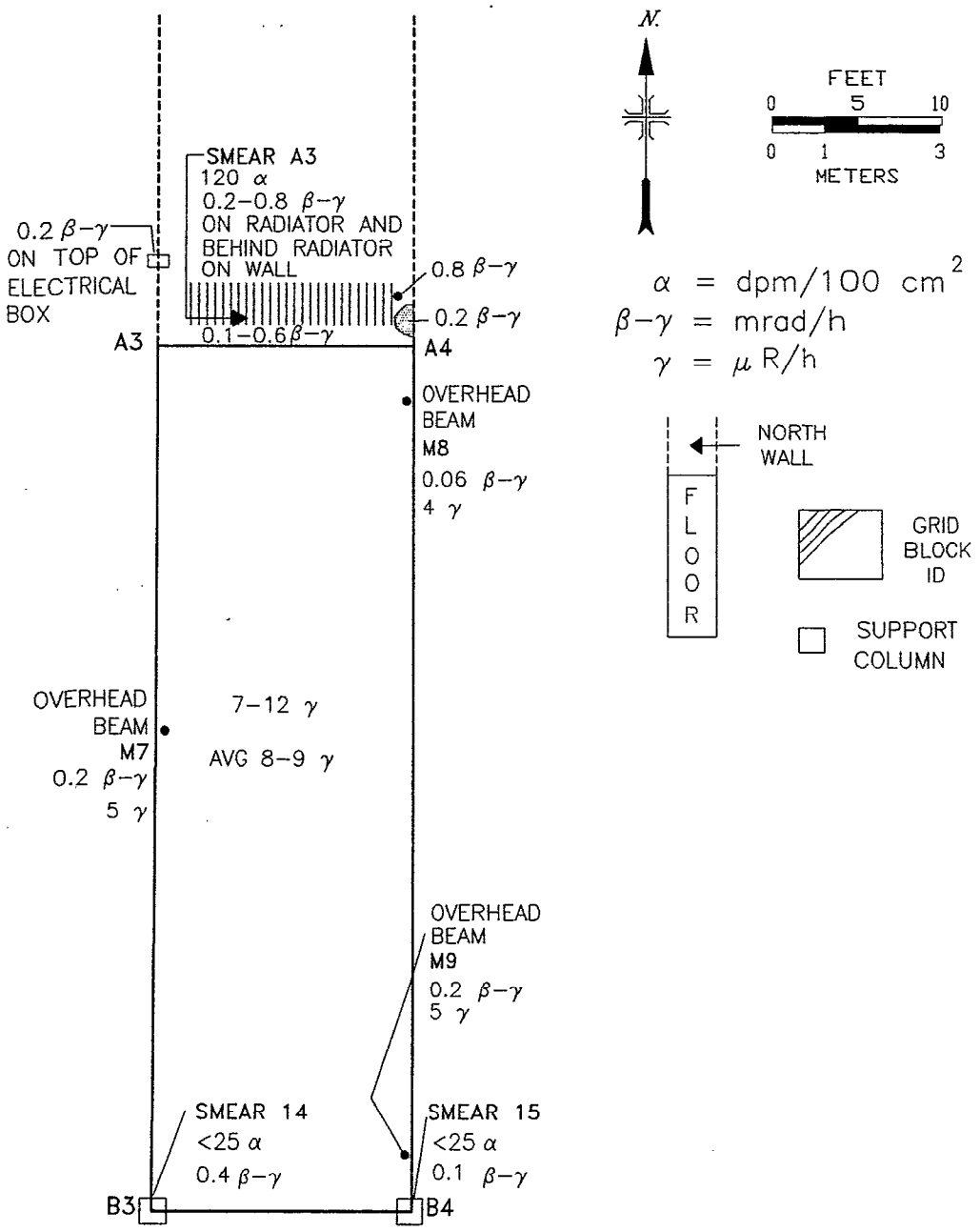

Fig. 11. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A3.

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ORNL-DWG 92-6736

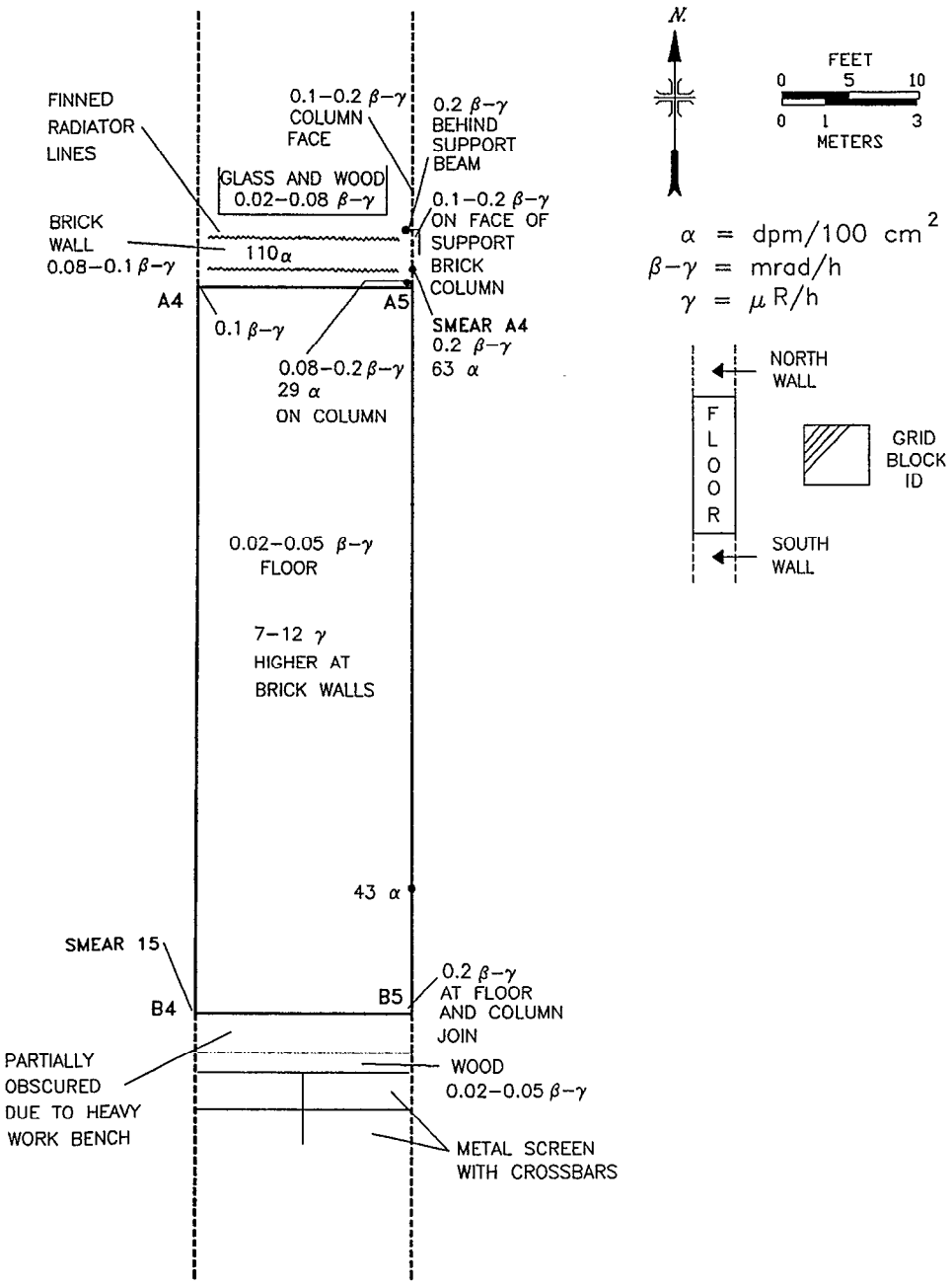

Fig. 12. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A4.

ORNL-DWG 92-6737

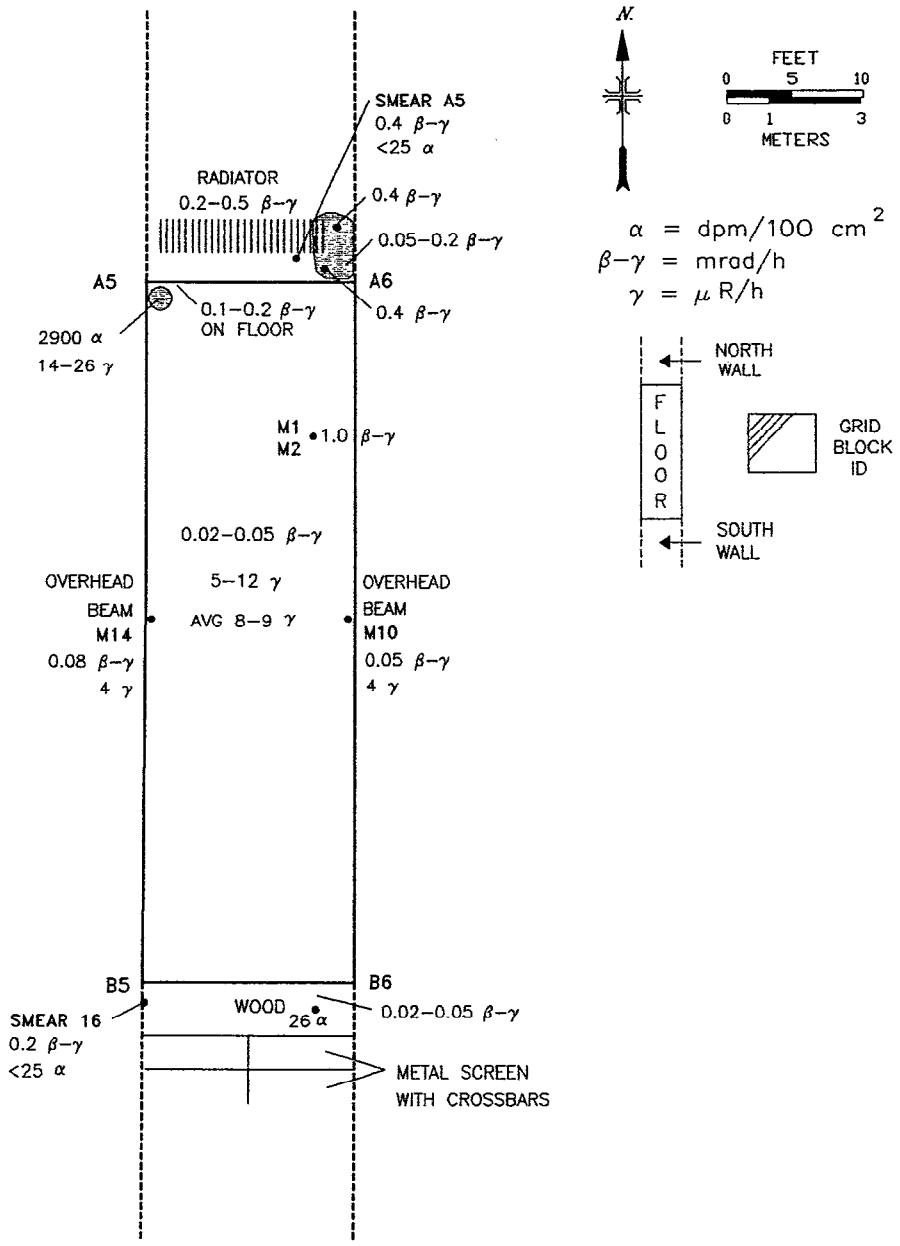

Fig. 13. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm²) in grid block A5.

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ORNL-DWG 92-6738

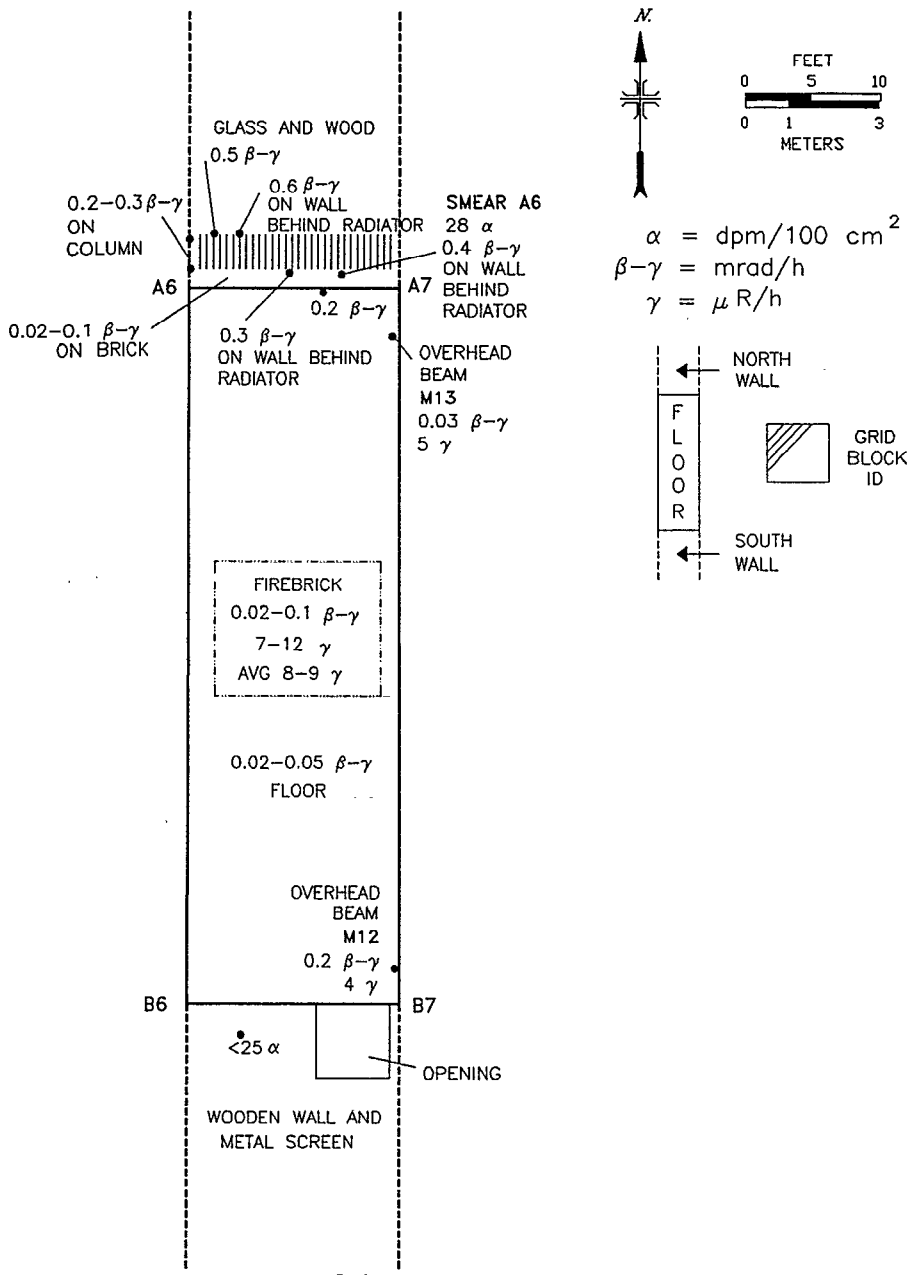

Fig. 14. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A6.

ORNL-DWG 92-6739

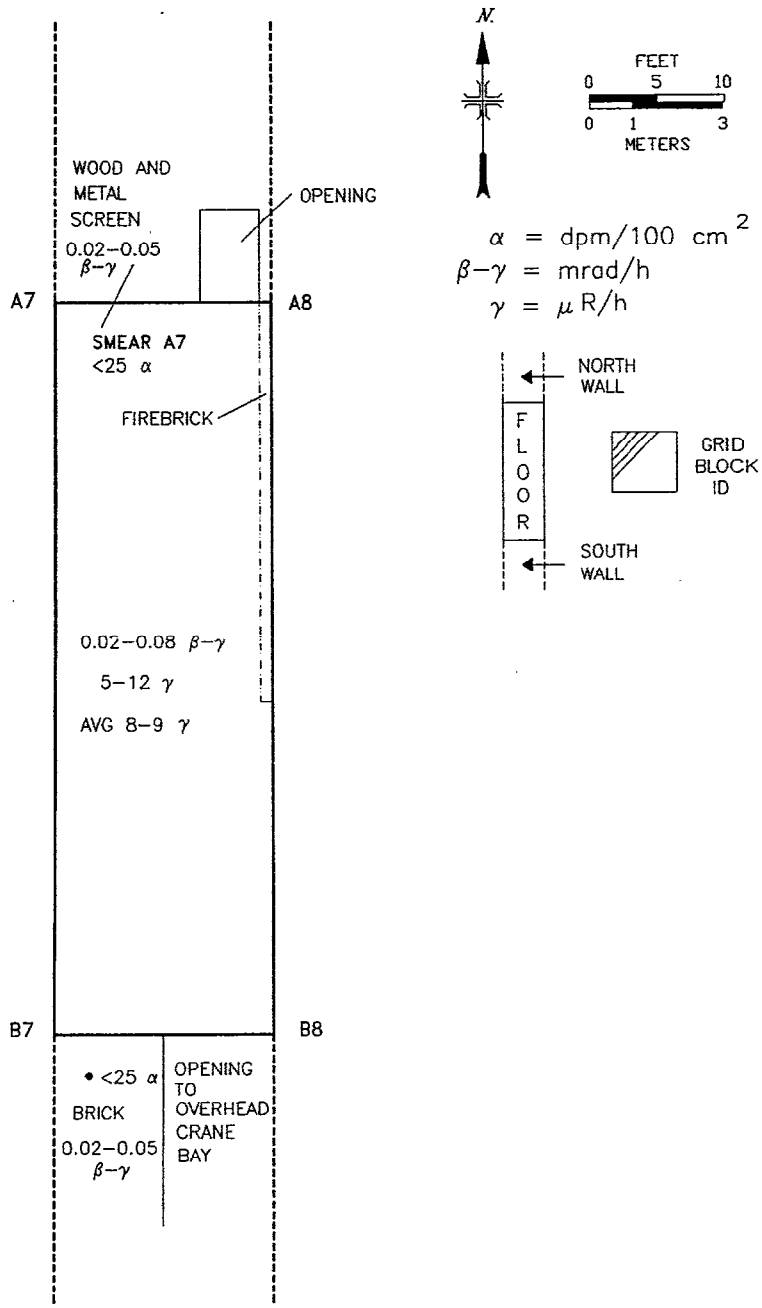

Fig. 15. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A7.

ORNL-DWG 92-6740

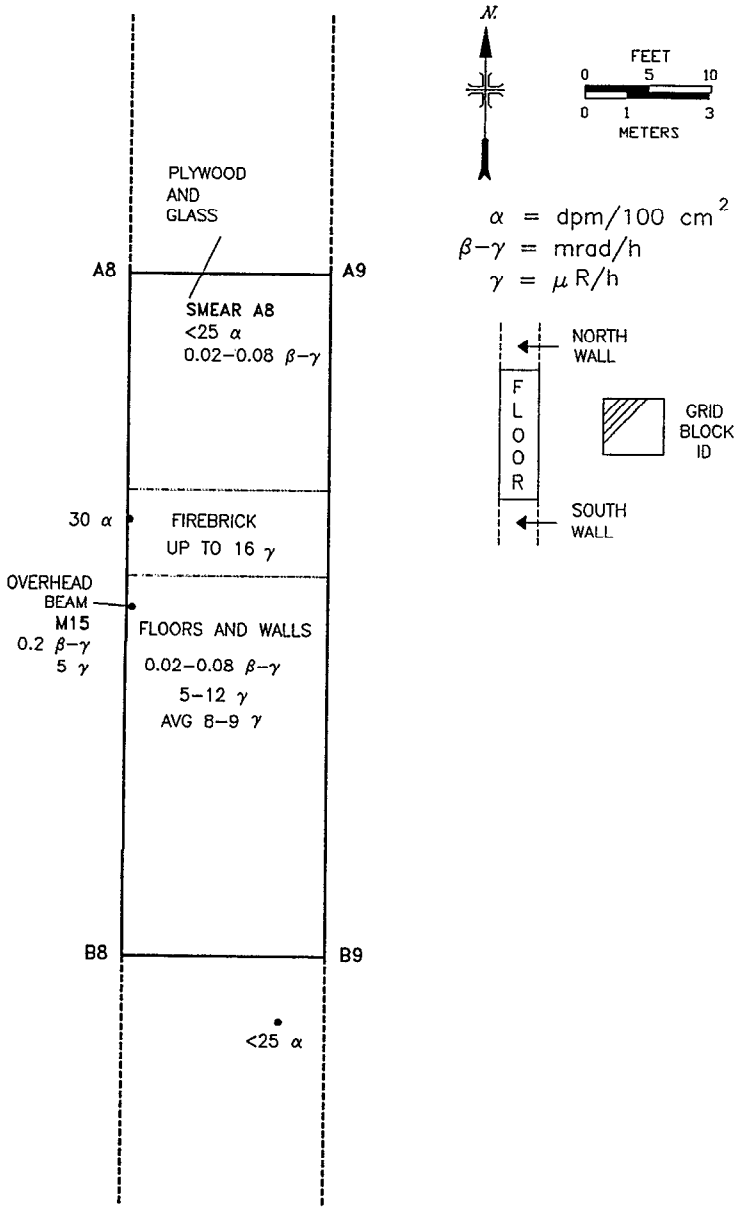

Fig. 16. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm²) in grid block A8.

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ORNL-DWG 92-6741

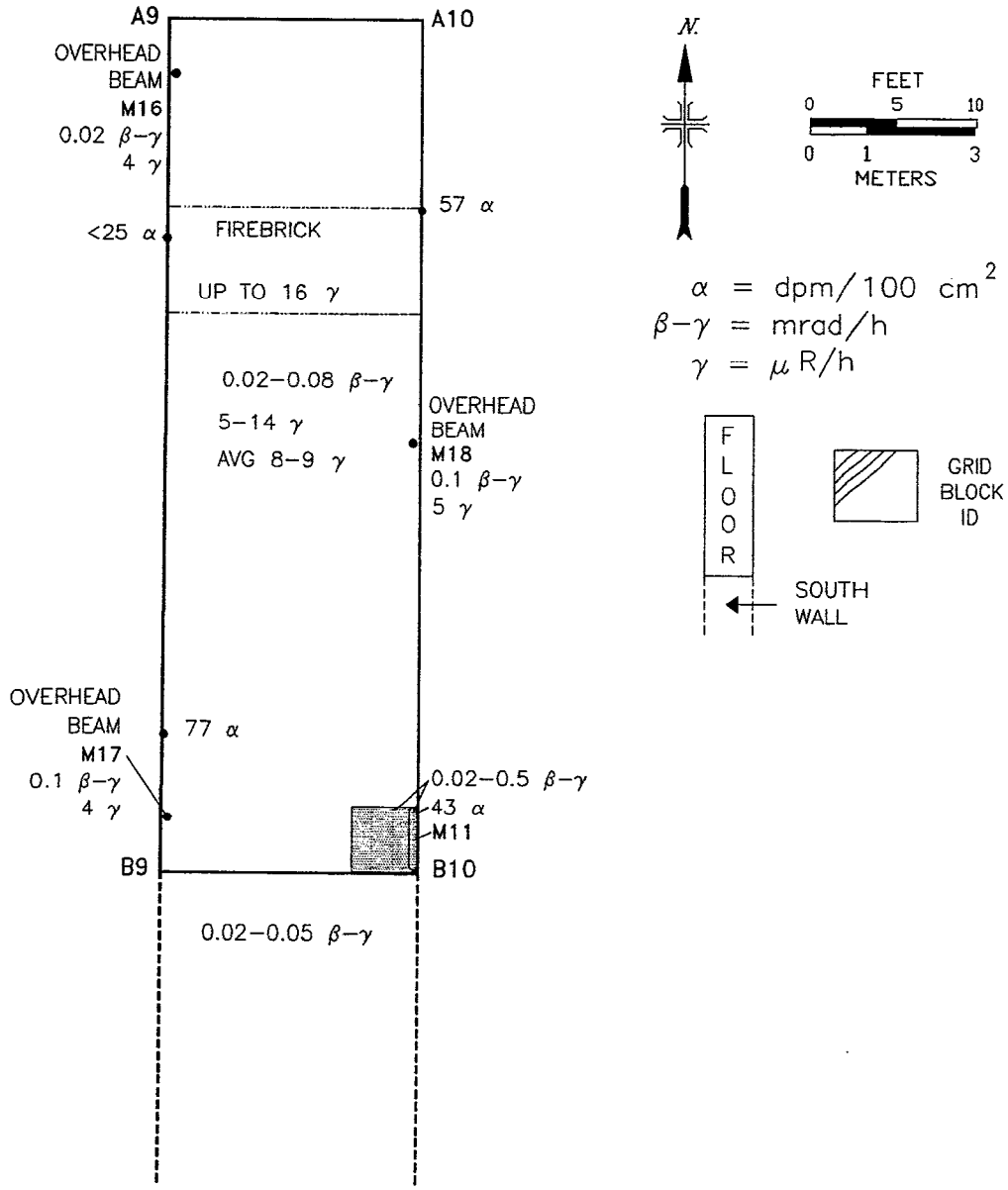

Fig. 17. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A9.

ORNL-DWG 92-6742

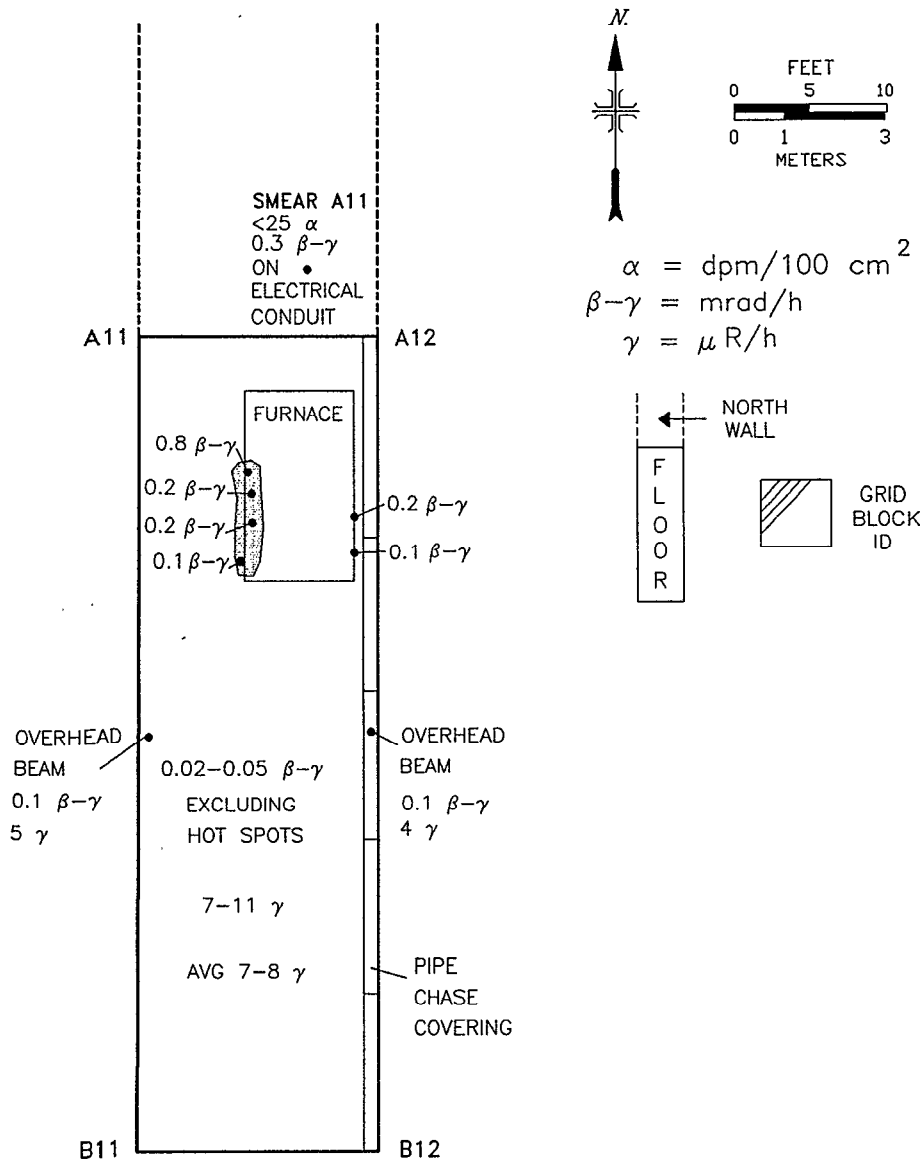

Fig. 18. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A11.

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ORNL-DWG 92-6743

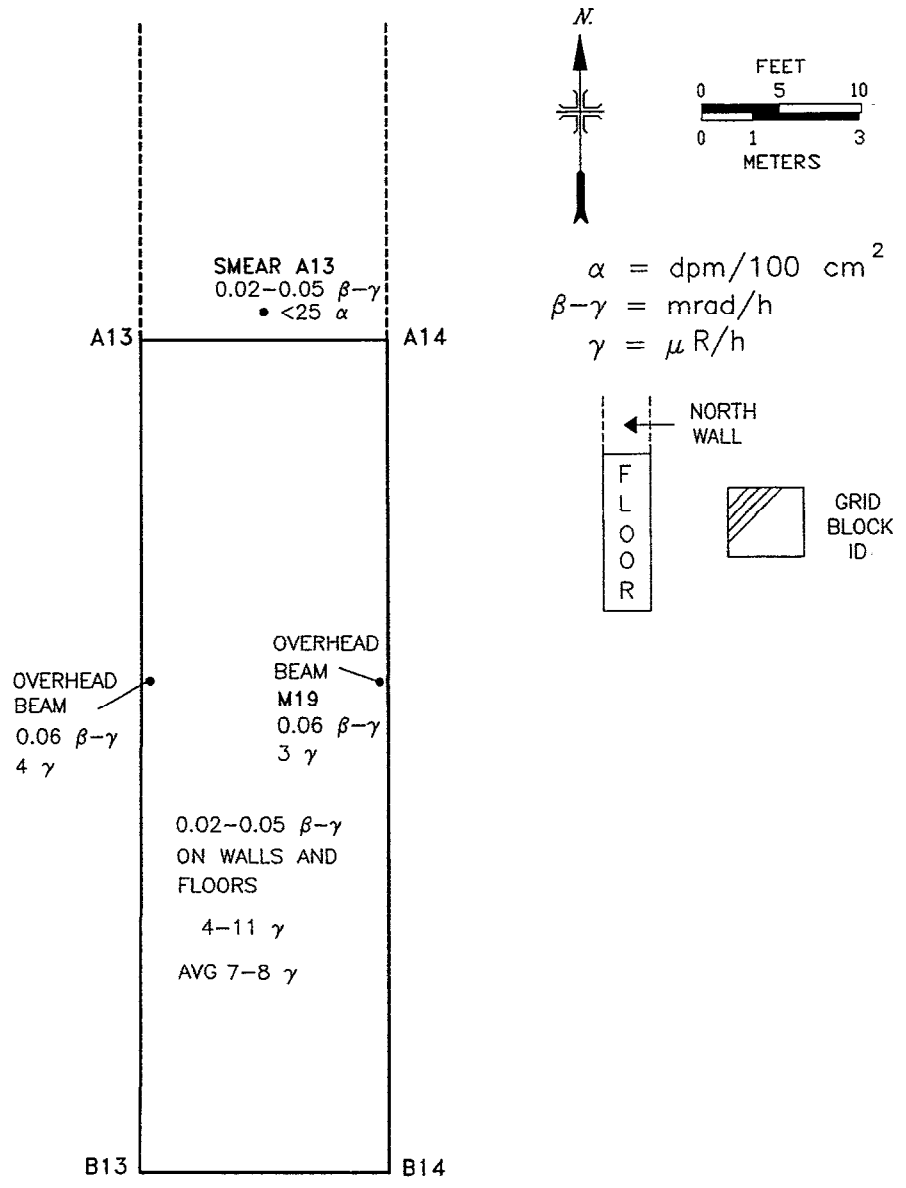

Fig. 19. Surface gamma exposure rates (μR/h), beta-gamma dose rates (mrad/h), and alpha activity levels (dpm/100 cm^2) in grid block A13.

ORNL-PHOTO 1142-92

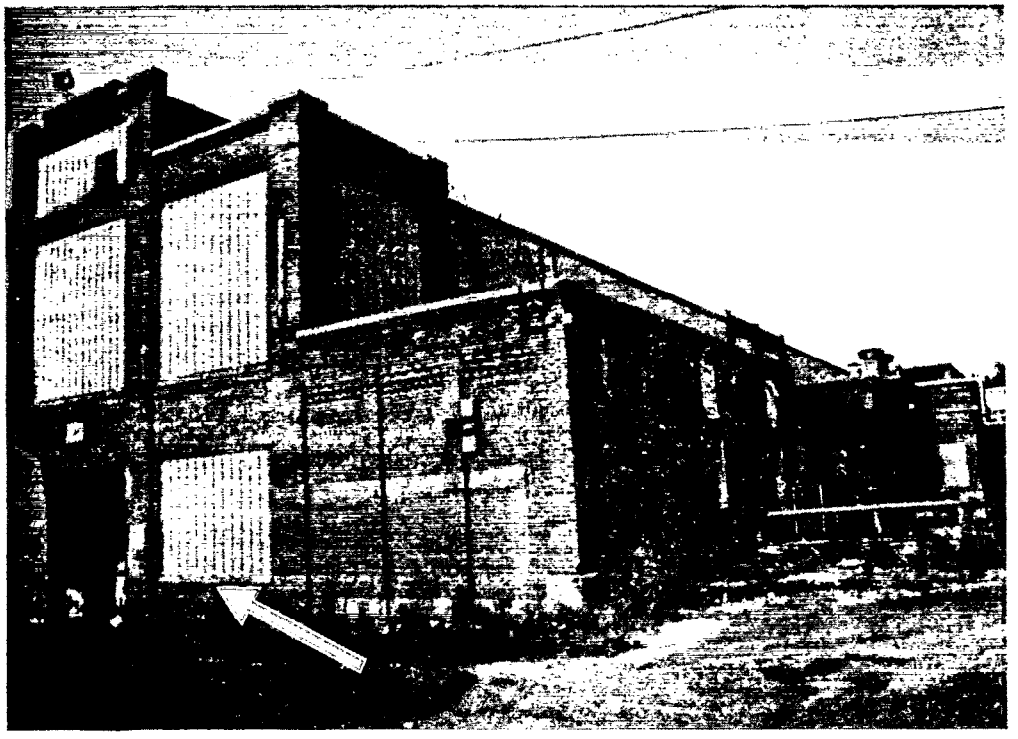

Fig. 20. View looking east northeast at Building 23, Indian Orchard, Massachusetts. Contamination was identified on the exterior of the building in the vicinity of the arrow.

ORNL-PHOTO 1143-92

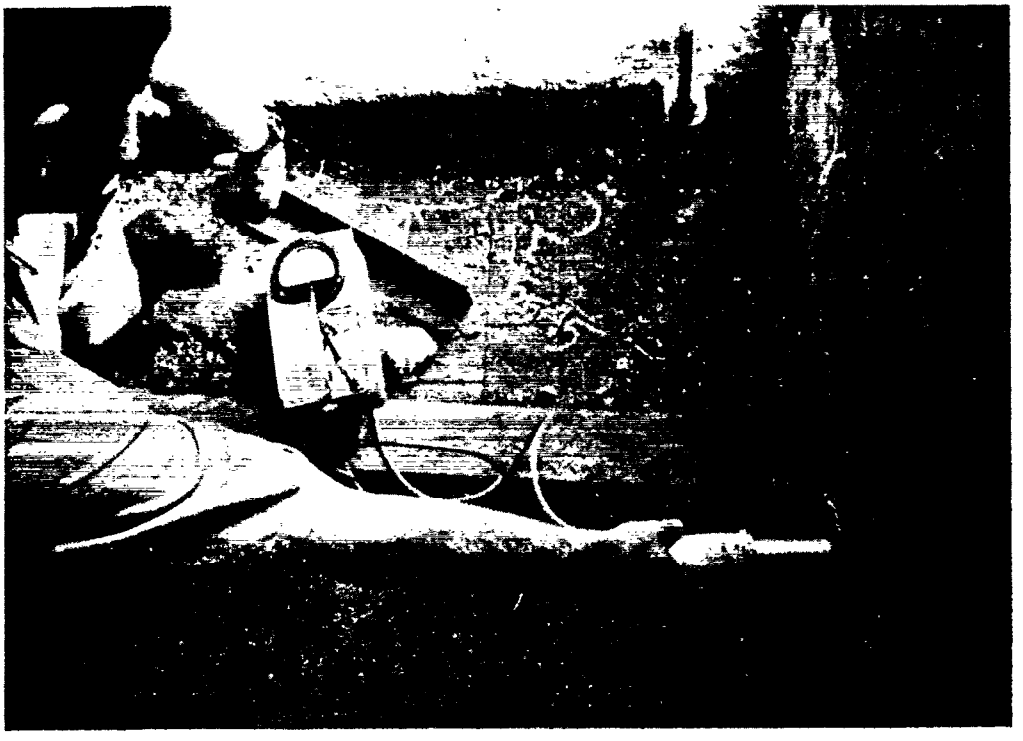

Fig. 21. View of contaminated area located north of the concrete ramp immediately inside the main entrance. The concrete entrance ramp is at the top of the photo; west wall of Building 23 is on the right.

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ORNL-PHOTO 1144-92

Fig. 22. View of area just north of main entrance with elevated beta-gamma levels on the wall, a ledge, and the door jamb.

ORNL-PHOTO 1145-92

Fig. 23. View of brick facing on column B2 (right side of photo behind cabinet). Chips of the brick facing material contained 900 pCi/g ^{238}U.

Table 1. Applicable guidelines for protection against radiation
(Limits for uncontrolled areas)

Mode of exposure	Exposure conditions	Guideline value
Gamma radiation	Indoor gamma radiation level (above background)	20 μR/h[a]
Total residual surface contamination[b]	^{238}U, ^{235}U, U-natural (alpha emitters)	
	or	
	Beta-gamma emitters[c]	
	Maximum	15,000 dpm/100 cm^2
	Average	5,000 dpm/100 cm^2
	Removable	1,000 dpm/100 cm^2
	^{232}Th, Th-natural (alpha emitters)	
	or	
	^{90}Sr (beta-gamma emitter)	
	Maximum	3,000 dpm/100 cm^2
	Average	1,000 dpm/100 cm^2
	Removable	200 dpm/100 cm^2
	^{226}Ra, ^{230}Th, transuranics	
	Maximum	300 dpm/100 cm^2
	Average	100 dpm/100 cm^2
	Removable	20 dpm/100 cm^2
Beta-gamma dose rates	Surface dose rate averaged over not more than 1 m^2	0.20 mrad/h
	Maximum dose rate in any 100-cm^2 area	1.0 mrad/h
Radionuclide concentrations in soil (generic)	Maximum permissible concentration of the following radionuclides in soil above background levels, averaged over a 100-m^2 area ^{226}Ra ^{232}Th ^{230}Th	5 pCi/g averaged over the first 15 cm of soil below the surface; 15 pCi/g when averaged over 15-cm-thick soil layers more than 15 cm below the surface

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Table 1 (continued)

Mode of Exposure	Exposure conditions	Guideline value
Derived concentrations	^{238}U	Site specific[d]

[a]The 20 μR/h shall comply with the basic dose limit (100 mrem/year) when an appropriate-use scenario is considered.

[b]DOE surface contamination guidelines are consistent with *NRC Guidelines for Decontamination at Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material*, May 1987.

[c]Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr, ^{228}Ra, ^{223}Ra, ^{227}Ac, ^{133}I, ^{129}I, ^{126}I, ^{125}I.

[d]DOE guidelines for uranium are derived on a site-specific basis. Guidelines of 35–40 pCi/g have been applied at other FUSRAP sites. *Source:* J. L. Marley and R. F. Carrier, *Results of the Radiological Survey at 4 Elmhurst Avenue, Colonie, New York (AL219)*, ORNL/RASA-87/117, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., February 1988; B. A. Berven et al., *Radiological Survey of the Former Kellex Research Facility, Jersey City, New Jersey*, DOE/EV-0005/29, ORNL-5734, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., February 1982.

Sources: Adapted from U.S. Department of Energy, DOE Order 5400.5, April 1990, and U.S. Department of Energy, *Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites*, Rev. 2, March 1987.

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Table 2. Background radiation levels and concentrations of selected radionuclides in soil for Massachusetts

Type of radiation measurement or sample	Radiation level or radionuclide concentration	
	Range	Average
Gamma exposure rate at ground surface (μR/h)[a]	6–9	7
Concentration of radionuclides in soil (pCi/g)[a]		
^{226}Ra	0.70–1.8	1.1
^{232}Th	0.76–1.2	0.97
^{238}U	0.69–2.7	1.2

[a]Values obtained from six locations in the Beverly, Massachusetts, area.

Source: W. D. Cottrell and R. F. Carrier, *Results of the Radiological Survey at the Ventron Site, Beverly, Massachusetts*, ORNL/TM-10053, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., May 1988 and R. D. Foley and M. S. Uziel, *Radiological Survey Results at 9 and 11 Congress Street, Beverly, Massachusetts (VB002)*, ORNL/RASA-91/13, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., May 1992.

Table 3. Alpha and beta-gamma measurements at Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts

Smear sample ID[a]	Directly measured radioactivity		Removable radioactivity[d] (smears)		Location
	Alpha[b] (dpm/100 cm^2)	Beta-gamma[c] (mrad/h)	Alpha[e] (dpm/100 cm^2)	Beta-gamma[f] (dpm/100 cm^2)	
2	36	0.02–0.05	<10	<200	Block A8, high wall A8–9, ~17 ft above floor
3	<25	0.02–0.05	<10	<200	Block A8, wall B8–9
4	<25	0.02–0.05	<10	<200	Block A6, high wall B6–7, brick ~17 ft above floor
5	<25	0.02–0.05	<10	<200	Block A6, high wall A6–7, metal vent on wall adjacent to A6, ~17 ft above floor
6	*g*	0.02	<10	<200	Block A6, high wall A6–7, inside metal vent on wall adjacent to A6, ~17 ft above floor
7	71	0.05–0.08	<10	<200	Block A4, high wall B4–5
8	43	0.02–0.08	<10	<200	Block A4, high wall A4–5, ~17 ft above floor
9	57	0.05–0.08	<10	<200	Block A2, high wall A2–3, ~17 ft above floor
10	550	0.05–0.2	14	<200	Block A2, high wall B2–3, verticle window sill, contamination washed from horizontal surface
11	650	4	36	<200	Block A1, west wall, 3.5 ft N of column B1, 2 ft above floor
12	33	0.8	<10	<200	Block A1, south wall, 1.5 ft W of column B2, 1 ft above floor
13	300	2.9	32	<200	Block A2, south wall, 1 ft above floor at column B2
14	<25	0.4	<10	<200	Block A3, south wall, 1 ft above floor at column B3
15	<25	0.1	<10	<200	Block A3, south wall, 1 m above floor at column B4
16	<25	0.2	<10	<200	Block A5, south wall, on column B5

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Table 3 (continued)

Smear sample ID[a]	Directly measured radioactivity		Removable radioactivity[d]		Location
	Alpha[b] (dpm/100 cm^2)	Beta-gamma[c] (mrad/h)	Alpha[e] (dpm/100 cm^2)	Beta-gamma[f] (dpm/100 cm^2)	
17	2900	0.7	<10	<200	Outside wall, 4 ft S of main (west) entrance, 1 ft above ground
A1	35	1.4	<10	<200	Block A1, bottom corner of column A2
A2	150	1.9	<10	<200	Block A2, bottom corner of column A2
A3	120	0.02–0.8	<10	<200	Block A3, 6 ft E of Column A3, 2 ft above floor, behind radiator
A4	63	0.2	<10	<200	Block A4, ledge at corner of column A5
A5	<25	0.4	<10	<200	Block A5, 13 ft E column A5, 3 ft above floor
A6	28	0.4	90	<200	Block A6, north wall, 11 ft E of column A6, 2 ft above floor
A7	<25	0.02–0.05	<10	<200	Block A7, north wall, wood wall 6 ft E of column A7, 3 ft above floor
A8	<25	0.02–0.08	<10	<200	Block A8, wood wall 7 ft E of column A8, 3 ft above floor
A11	<25	0.3	<10	<200	Block A11, 12 ft E of column A11, 6 ft above floor, metal box on wall
A12	<25	0.02–0.05	<10	<200	Block A12, 9 ft E of column A12, 2 ft above floor
A13	<25	0.02–0.05	<10	<200	Block A13, 8 ft E of column A13, 2 ft above floor
A15	<25	0.02–0.05	<10	<200	Block A15, 6 ft E of column A15, 1 ft above floor

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Table 3 (continued)

Smear sample ID[a]	Directly measured radioactivity		Removable radioactivity[d]		Location
	Alpha[b] (dpm/100 cm^2)	Beta-gamma[c] (mrad/h)	Alpha[e] (dpm/100 cm^2)	Beta-gamma[f] (dpm/100 cm^2)	
A16	<25	0.02–0.05	<10	<200	Block A16, 6 ft E of column A16, 2 ft above floor
A17	28	0.02–0.05	<10	<200	Block A17, 7 ft E of column A17, 3 ft above floor

[a]Some sample locations are shown on Figs. 9–19 (p. 17-27, Smear #).
[b]Instrument-specific minimum detectable activity (MDA) level = 25 dpm/100 cm^2.
[c]MDA = 0.01 mrad/h.
[d]Removable radioactivity reported as net disintegration rates. Background radiation levels have been subtracted.
[e]MDA = 10 dpm/100 cm^2.
[f]MDA = 200 dpm/100 cm^2.
[g]Not measured.

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Table 4. Concentrations of radionuclide in soil, dust, and debris samples collected at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts

Sample[a]	Location	Radionuclide concentration (pCi/g)[b]		
		^{226}Ra	^{232}Th	^{238}U
		Soil Samples		
S1[c]	West of Bldg. 23	0.33±0.01	0.4 ±0.02	<1.4
B1[d]	Southeast corner of Bldg. 23	1.0 ±0.03	1.4 ±0.06	1.9± 0.8
		Dust and debris samples		
M1[e]	Concrete floor anchor, grid block A5	<48	*f*	36,000 ±400
M2[g]	Concrete floor anchor, grid block A5	<0.9	*f*	220 ± 10
M3	Overhead beam No. 1	0.75± 0.1	0.76±0.2	250 ± 50
M4	Overhead beam No. 1	<0.8	*f*	340 ± 6
M5	Overhead beam No. 1	<0.5	*f*	170 ± 5
M6	Overhead beam No. 2	<0.8	*f*	310 ± 6
M7	Overhead beam No. 3	0.43± 0.09	<0.3	130 ± 5
M8	Overhead beam No. 4	0.7 ±0.07	0.5 ±0.1	48 ± 5
M9	Overhead beam No. 4	0.6 ±0.1	0.7 ±0.1	95 ± 2
M10	Overhead beam No. 6	0.4 ±0.07	0.3 ±0.1	25 ± 7
M11[h]	Composite floor debris, grid block A9	0.6 ±0.1	0.6 ±0.1	240 ± 20
M12	Overhead beam No. 7	0.6 ±0.1	0.7 ±0.2	330 ± 10
M13	Overhead beam No. 7	0.22±0.05	0.23±0.07	60 ± 5
M14	Overhead beam No. 5	0.4 ±0.06	0.5 ±0.07	35 ± 5
M15	Overhead beam No. 8	0.4 ±0.08	0.5 ±0.1	350 ± 50
M16	Overhead beam No. 9	0.16±0.03	0.11±0.03	16 ± 4
M17	Overhead beam No. 9	0.6 ±0.08	0.4 ±0.1	57 ± 10
M18	Overhead beam No. 10	0.20±0.04	0.20±0.07	65 ± 7
M19	Overhead beam No. 14	0.6 ±0.07	0.6 ±0.1	50 ± 5
M20	Overhead beam No. 15	0.6 ±0.1	0.7 ±0.2	37 ± 5

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Table 4 (continued)

Sample[a]	Location	Radionuclide concentraation (pCi/g)[b]		
		^{226}Ra	^{232}Th	^{238}U
M21	Overhead beam No. 16	0.6 ±0.1	0.7 ±0.2	35 ± 2
M22	Overhead beam No. 17	0.7 ±0.08	0.8 ±0.1	20 ± 5
M23	Overhead beam No. 18	0.7 ±0.08	0.6 ±0.1	20 ± 2
M24	Overhead beam No. 19	0.6 ±0.07	0.7 ±0.1	19 ± 1
M25	Overhead beam No. 20	<1.0	<1.0	10 ± 3
M26	Overhead beam No. 23	0.5 ±0.03	0.6 ±0.06	6.0± 1.5
M27	Moveable overhead crane, grid block A23	0.2 ±0.04	0.2 ±0.04	10 ± 3
M28	Overhead beam No. 26	<1.5	*f*	6.4± 0.3
M29	Overhead beam, grid block A2	<5.0	*f*	1,300 ±100
M30	High-wall window sill, grid block A2	<3.0	*f*	1,300 ± 50
M31[h]	West door ramp, north corner	<1.0	<1.0	120 ± 10
M32	Chips from brick facing, column B2	<1.4	*f*	900 ± 20

[a]Sample locations are shown on Fig. 8 (p. 16).
[b]Indicated counting error is at the 95% confidence level (±2σ).
[c]Systematic soil sample (S) taken at location irrespective of gamma exposure rates. Sample depth 0–8 cm.
[d]Biased soil sample (B) taken from area with elevated gamma exposure rates. Sample collected 3 ft under edge of concrete slab. Sample depth 0–15 cm.
[e]Sample depth 0–1 cm.
[f]Not measured.
[g]Sample depth 1–3 cm.
[h]Sample depth 0–2 cm.

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Table 5. Comparison of ^{238}U concentrations in selected dust samples from overhead beams with DOE guideline value for removable surface contamination

Sample[a]	Location	^{238}U (pCi/g)	^{238}U (dpm/100 cm^2)	Percent of guideline value
M3	Overhead beam No. 1	250	5,300	530
M4	Overhead beam No. 1	340	3,600	360
M5	Overhead beam No. 1	170	2,000	200
M6	Overhead beam No. 2	310	3,500	350
M7	Overhead beam No. 3	130	4,000	400
M8	Overhead beam No. 4	48	1,500	150
M9	Overhead beam No. 4	95	2,800	280
M10	Overhead beam No. 6	25	1,000	100
M12	Overhead beam No. 7	330	9,000	900
M13	Overhead beam No. 7	60	2,600	260
M14	Overhead beam No. 5	35	1200	120
M15	Overhead beam No. 8	350	12,000	1,200
M16	Overhead beam No. 9	16	900	90
M17	Overhead beam No. 9	57	1,300	130
M18	Overhead beam No. 10	65	2,800	280
M19	Overhead beam No. 14	50	1,400	140
M22	Overhead beam No. 17	25	500	50
M23	Overhead beam No. 18	20	530	53
M24	Overhead beam No. 19	19	490	49
M26	Overhead beam No. 23	6	200	20
M27	Moveable overhead crane, grid block A23	10	1,600	160
M28	Overhead beam No. 26	6	130	13
M29	Overhead beam, grid block A2	1,300	1,400	140

[a]Samples collected at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts. Sample locations are shown on Fig. 8 (p. 16).

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Table 6. Alpha, beta-gamma, and gamma radiation measurements at dust and debris sample locations at the former Chapman Valve facility, Indian Orchard, Massachusetts

		Alpha (dpm/100 cm^2)	Beta-gamma (mrad/h)		Gamma (μR/h)
Sample[a]	Location	After collection	Prior to collection	After collection	Prior to collection
M1 and M2[b]	Concrete floor anchor under wooden flooring, grid block A5	110	1	4	11 17[c]
M3	Overhead beam No. 1	200	0.2	0.07	5 10[d]
M4	Overhead beam No. 1	170	0.5	0.5	4
M5	Overhead beam No. 1	300	0.4	0.3	6
M6	Overhead beam No. 2	260	0.3	0.3	4
M7	Overhead beam No. 3	43	0.2	0.2	5
M8	Overhead beam No. 4	200	0.06	0.06	4
M9	Overhead beam No. 4	400	0.2	0.4	5
M10	Overhead beam No. 6	46	0.05	0.06	4
M11	Composite floor debris, grid block A9	43[e]	0.02-0.5	*f*	*f*
M12	Overhead beam No. 7	320	0.2	0.06	4
M13	Overhead beam No. 7	180	0.03	0.05	5
M14	Overhead beam No. 5	210	0.08	0.2	4
M15	Overhead beam No. 8	990	0.2	0.1	5
M16	Overhead beam No. 9	300	0.02	0.09	4

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Table 6 (continued)

Sample[a]	Location	Alpha (dpm/100 cm^2) After collection	Beta-gamma (mrad/h) Prior to collection	Beta-gamma (mrad/h) After collection	Gamma (μR/h) Prior to collection
M17	Overhead beam No. 9	65	0.10	0.06	4
M18	Overhead beam No. 10	200	0.1	0.07	5
M19	Overhead beam No. 14	*f*	0.06	0.2	3
M20	Overhead beam No. 15	160	0.06	0.06	3
M21	Overhead beam No. 16	180	0.05	0.06	4
M22	Overhead beam No. 17	220	0.06	0.04	4
M23	Overhead beam No. 18	65	0.03	0.03	4
M24	Overhead beam No. 19	65	0.04	0.03	3
M25	Overhead beam No. 20	140	0.03	0.04	2
M26	Overhead beam No. 23	110	0.03	0.03	2
M27	Moveable overhead crane, grid block A23	55	0.05	0.08	2
M28	Overhead beam No. 26	<25	0.02	0.02	2
M29	Overhead beam, grid block A2	*f*	0.6	0.3	*f*
M30	High-wall window sill, grid block A2	*f*	2	*f*	*f*

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Table 6 (continued)

		Alpha (dpm/100 cm^2)	Beta-gamma (mrad/h)		Gamma (μR/h)
Sample[a]	Location	After collection	Prior to collection	After collection	Prior to collection
M31	West door ramp, north corner	*f*	0.2-1.2	*f*	32
M32	Chips from brick facing, column B2	300	2.9	*f*	*f*

[a]Sample locations are shown on Fig. 8 (p. 16).

[b]Dust samples collected inside concrete floor anchor previously used to anchor machinery. Hole was too small to accommodate measurement instruments. All alpha, beta-gamma, and gamma measurements taken at top of hole.

[c]Gamma measurement at 1 m prior to sample collection.

[d]Gamma measurement on contact after sample collection.

[e]Prior to sample collection.

[f]Not measured.

Editor's Note:

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46-48. G. K. Hovey, Bechtel National, Inc., FUSRAP Department, Oak Ridge Corporate Center, 151 Lafayette Drive, P.O. Box 350, Oak Ridge, TN 37831-0350
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131479

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95.327

United States Government

Department of Energy

Oak Ridge Operations

memorandum

JUN 28 2 17 PM '95

DATE: June 19, 1995

REPLY TO ATTN OF: EW-93:Hartman

SUBJECT: **CX DETERMINATION - REMOVAL ACTION AT THE FORMER CHAPMAN VALVE MANUFACTURING COMPANY, FUSRAP-034**

TO: Robert D. Dempsey, Assistant Manager for Environmental Management, EW-90

Attached is a categorical exclusion (CX) determination describing the proposed removal and disposal of radioactively contaminated materials at the former Chapman Valve Manufacturing Company site, Indian Orchard, Massachusetts. I have determined that this action conforms to an existing National Environmental Policy Act (NEPA) Subpart D CX and may be categorically excluded from further NEPA review and documentation.

If you have any questions concerning NEPA compliance issues, please contact Patricia W. Phillips, ORO NEPA Compliance Officer, at (615) 576-4200.

Robert W. Poe
Assistant Manager for Environment,
Safety & Quality

Attachment

cc w/attachment:
S. C. Golian, EM-22, TREV II
G. S. Hartman, EW-93, ORO
J. D. Kopotic, EW-93, ORO
P. W. Phillips, SE-311, ORO
M. E. Redmon, BNI
S. P. Riddle, EW-91, ORO
J. Russell, EM-421, BAH, TREV II
W. M. Seay, EW-93, ORO
J. D. Waddell, SAIC
J. W. Wagoner II, EM-421, TREV II

95-019

131479

CATEGORICAL EXCLUSION (CX) FOR REMOVAL ACTION AT THE FORMER CHAPMAN VALVE SITE

PROPOSED ACTION: Removal of radioactively contaminated materials at the former Chapman Valve Manufacturing Company site.

LOCATION: Former Chapman Valve Manufacturing Company site, Indian Orchard, Massachusetts [FUSRAP site].
The former Chapman Valve Manufacturing Company site is located at 203 Hampshire Street in Indian Orchard, Massachusetts, a northeast suburb of Springfield. The Chapman Valve Manufacturing Company machined uranium metal for Brookhaven Laboratory under contract to the Manhattan Engineer District (MED) and the Atomic Energy Commission (AEC). The building used for MED-related work has been abandoned since 1987 and has undergone general structural deterioration due to exposure to the elements and disuse and is considered unsafe for occupancy.

DESCRIPTION OF PROPOSED ACTION: The proposed action is to safely remove, transport, and dispose of radioactively contaminated materials at the former Chapman Valve site, thereby eliminating potential exposure of workers and the public to contamination exceeding applicable cleanup guidelines. Proposed site activities include, but are not limited to, the following: Decontamination (removal if necessary) of interior concrete floor areas (wood blocks and tar over a concrete base) and subsurface soils; decontamination of structural surfaces in the portion of the building used for MED contract work; removal of asbestos-containing material; decontamination of a 10-ton traveling crane; removal of inoperable electrical fixtures and equipment (such as electrical wiring, piping, conduit, radiators, HVAC ducts, and transformer boxes); temporary onsite storage of wastes; packaging, transportation, and disposal of materials at existing appropriately licensed disposal facilities; and disposal of waste/debris below DOE contamination/radiological release guidelines in a commercial disposal facility. In the event that disposal delays require temporary staging and/or storage of contaminated wastes, storage would be conducted in accordance with all applicable regulations.

Where failed roof sections allow seepage of the elements inside the building, plans currently call for implementing engineering controls (e.g., temporary supports, netting, or temporary decking) underneath the degraded sections to protect personnel from potentially falling debris. The roof is a tar composite that contains from 2 to 3 percent asbestos. If it becomes necessary to remove the entire roof in the interest of safety, removal of asbestos will be conducted in accordance with federal, state, and local

CATEGORICAL EXCLUSION (CX) FOR REMOVAL ACTION AT THE FORMER CHAPMAN VALVE SITE (cont.)

standards, including certification of removal contractors and technicians. Low-level radioactive waste (estimated at approximately 40 yd^3) and low-level radioactive mixed waste (consisting of small amounts of paint flakes generated from decontamination of the crane and structural steel) would be temporarily stored, packaged, and transported in accordance with applicable federal and state regulations and disposed of at a licensed disposal facility.

The proposed removal action would be conducted under DOE authorities pursuant to the Atomic Energy Act (AEA), would be consistent with the final remedial action for the site, and meets the eligibility criteria for conditions that are integral elements of actions eligible for categorical exclusion as stated in 10 CFR 1021:

1. The proposed action would not threaten a violation of applicable statutory, regulatory, or permit requirements for environment, safety, and health, including requirements of DOE orders. All activities would be managed by FUSRAP.

2. The proposed action would not require siting and construction or major expansion of waste storage, disposal, recovery, or treatment facilities (including incinerators and facilities for treating wastewater, surface water, and groundwater). Wastes generated during the proposed action would be collected, analyzed to determine waste characteristics, and segregated as they are generated into nonhazardous, RCRA-only, mixed, and radioactive-only categories. If hazardous wastes are determined to be commingled with radioactive waste, removal and temporary storage would be done in accordance with applicable requirements; the mixed waste would then be disposed of at an existing facility designed to accept these wastes. Wastes would be transported offsite in accordance with applicable transportation and disposal requirements and disposed of at existing facilities or stored temporarily onsite in accordance with applicable requirements pending evaluation of final disposal options. If temporary storage is required, wastes generated from these activities would be managed in accordance with regulations applicable to the types of wastes being managed.

3. The proposed action would not disturb hazardous substances, pollutants, contaminants, or CERCLA-excluded petroleum and natural gas products that preexist in the environment such that there would be uncontrolled or unpermitted releases. The removal action would be conducted in an environmentally responsible manner to ensure site-specific control of environmental contaminants.

CATEGORICAL EXCLUSION (CX) FOR REMOVAL ACTION AT THE FORMER CHAPMAN VALVE SITE (cont.)

4. The proposed action would not adversely affect any environmentally sensitive resources defined in the Federal Register Notice referenced below, including archaeological or historical sites; potential habitats of threatened or endangered species; floodplains; wetlands; areas having a special designation such as Federally- and state-designated wilderness areas, national parks, national natural landmarks, wild and scenic rivers, state and Federal wildlife refuges, and marine sanctuaries; prime agricultural lands; special sources of water such as sole-source aquifers; and tundra, coral reefs, or rain forests. The proposed action would occur in a previously disturbed/developed area.

There are no extraordinary circumstances related to the proposal that may affect the significance of the environmental effects of the proposal, and the proposal is not precluded by 40 CFR 1506.1 or 10 CFR 1021.211.

The estimated cost for this action is less than $2 million and would take less than 12 months to complete.

CX TO BE APPLIED: From the DOE NEPA Implementing Procedures, 10 CFR 1021, Subpart D, Appendix B, under actions that "Normally Do Not Require EAs or EISs,"

> "B6.1 Removal actions under CERCLA (including those taken as final response actions and those taken before remedial action) and removal-type actions similar in scope under RCRA and other authorities (including those taken as partial closure actions and those taken before corrective action), including treatment (e.g., incineration), recovery, storage, or disposal of wastes at existing facilities currently handling the type of waste involved in the removal action...."

131479

CATEGORICAL EXCLUSION (CX) FOR REMOVAL ACTION AT THE FORMER CHAPMAN VALVE SITE (cont.)

I have concluded that the proposed action meets the requirements for the CX referenced above. Therefore, I recommend that the proposed action be categorically excluded from further NEPA review and documentation.

Patricia W. Phillips 6-5-95

Patricia W. Phillips, ORO NEPA Compliance Officer — Date

Based on my review and the recommendation of the ORO NEPA Compliance Officer, I recommend that the proposed action be categorically excluded from further NEPA review and documentation.

Robert D. Dempsey 6/15/95

Robert D. Dempsey, Assistant Manager for Environmental Restoration and Waste Management, ORO — Date

Based on the recommendations of the ORO NEPA Compliance Officer and the Assistant Manager for Environmental Restoration and Waste Management, I determine that the proposed action is categorically excluded from further NEPA review and documentation.

Robert W. Poe 6/19/95

Robert W. Poe, Assistant Manager for Environment, Safety & Quality — Date

FUSRAP

CCN: 132040

FSRD ☐ COMM TYPE 3 B B

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DOE/ORO FORMER SITES RESTORATION DIVISION (EW-93)

DATE PROCESSED BY PDCC JUL 11 1995

SUBJECT MA Dept of Env. Prot. - Generator Registration

FROM B. Vallem TO L. Rudek COMM DATE 7/5/95

ADDR CODE ____ CLOSES CCN ____ WBS 133

SUBJECT CODE 7480 (NUMBER ST) AFFECTED DOCUMENT

RESPONSE TRACKING INFORMATION

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MESSAGE:

		W/A	W/O			W/A	W/O			W/A	W/O
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D. Adler	FSRD			SECRETARY: S. Heptinstall	SAIC			PROJECT MANAGER: M. Redmon	BPM		
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R. Kirk	FSRD							ENGINEERING & TECHNOLOGY	BET		
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M. Noe	FSRD							ENVIR TECH/DATA	BET		
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SECRETARY: M. Selber/M. Dyke/ T. Patterson	FSRD							SAFETY & HEALTH	BEH		
								ENVIR COMPLIANCE	BEH		
FSRD CHRON FILE								WASTE MGMT & TREATMENT	BEH		
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READING FILE								PROJECT ADMINISTRATION	BPA		
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ORNL:	ORNL			133 Chapman			1	PDCC SENSITIVE CHRON FILE			

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132040

Massachusetts Department of Environmental Protection - Generator Registration

DEP Region: NE CE SE X WE

I am registering as: X Very Small Quantity Generator of Hazardous Waste (less than 27 gallons/month)
Very Small Quantity Generator of Waste Oil (less than 27 gallons/month) or
Small Quantity Generator of Waste Oil (less than 270 gallons/month)

Bechtel National, Inc. for the US Department of Energy
Name of Company

P.O. Box 350, Oak Ridge, TN 37831-0350 (615) 574-3736
Mailing Address: Street City/Town Zip

203 Hampshire Street, Indian Orchard, MA 01151
Address where waste is produced

Environmental Remediation 9999 M V 4 1 3 5 4 3 4 7 3 2
Type of Business SIC Code If you do not have an MAD #, use MV Area Code and Telephone Number

Hazardous Waste Generated (check):	Gallons per Month Prior to Treatment or Recycling:	Disposal, Storage, Treatment and/or Recycling: (Name Company and address where waste is taken or type of treatment on site of generation)
☐ Waste Oil		HAND DELIVERED
☐ Solvent		DATE 7-5-95
☐ Acid or Alkali		RECD. BY G Gates
Other (name):		
☒ D008 less than 10 kg solid		Stabilization with Portland cement
☐		to render non-hazardous

I CERTIFY UNDER PENALTY OF LAW that I have personally examined and am familiar with the information submitted in this document and all attachments and that, based on my inquiry of those individuals immediately responsible for obtaining the information, I believe that the information is true, accurate and complete.

6/30/95 — Signed for USDOE — M. E. Redmon for R. R. Harbert
Date Signed Name of Owner or Operator

KEEP BEIGE COPY - SEND WHITE COPY TO DEP REGIONAL OFFICE

130873

Department of Energy

Oak Ridge Operations
P.O. Box 2001
Oak Ridge, Tennessee 37831—

June 6, 1995

Anthony D. Pantaleoni
Vice President
Crane Company
100 First Stamford Place
Stamford, Connecticut 06902

Dear Mr. Pantaleoni:

REAL ESTATE LICENSE REORDOER-7-95-0139, CHAPMAN VALVE MFG CO., INDIAN ORCHARD, MA

Enclosed for your records is a copy of the fully executed license between Crane Company and the Department of Energy. Thank you for your cooperation and consideration of this issue. If you have any questions concerning the real estate instrument, please feel free to call me at 615-576-0977.

Sincerely,

Katy Kates

Katy Kates
Realty Officer

Enclosure
As stated

cc: Sally Haywood, Bechtel FUSRAP

130873

REAL ESTATE LICENSE NO.
REORDOER-7-95-0139

DEPARTMENT OF ENERGY

LICENSE

PROJECT: CHAPMAN VALVE MANUFACTURING CO., INDIAN ORCHARD, MA
PURPOSE: REMEDIAL ACTION

THIS LICENSE, between Crane Company, known as the "Grantor" and the U.S. Department of Energy, known as the "Grantee", is subject to the following terms and conditions.

1. Rights Granted - The Grantor grants to the Grantee, its agents, employees, or representatives permission to use the premises or facilities, together with ingress and egress, for the purpose of performing remedial action to remove contaminated material

at the location shown depicted on Exhibit(s) "A" attached to this instrument and more specifically identified in whole or in part as Parcel No.(s) N/A filed in Deed Book 2891, Page 53 in the records of Hampden County, Massachusetts.

2. Term/Termination Rights - This License is effective upon the date of execution by the Grantee of this instrument and shall continue in effect for a period of/thru five (5) years unless terminated by either of the parties on not less than thirty (30) days prior written notice given to the other; provided, however, that the Grantor may not terminate this License without the Grantee's approval.

3. Consideration - Upon execution of this License by the Grantee, the Grantee shall initiate action to pay to the Grantor the sum of $ ($) as full and complete payment for the rights granted within this License.

FILED

4. Authority to License - The Grantor represents and warrants that it is the owner of the property and has full right, power, and authority to enter into this License and grant the rights set out in this License.

5. Grantor Responsibility - The Grantor responsibility is set out within the terms and conditions of the rights granted under this License. The Grantor makes no representation as to the suitability or fitness of the premises for the intended purpose.

6. Grantee Responsibility - The Grantee, its agents, employees, or representatives will be responsible for property damage or injury to persons caused by the sole and direct negligence of their respective employees in performing on the Grantor's premises the activities and restoration which are the subject of this License. Grantee shall obtain all necessary permits, licenses, and approvals in connection with the activities to be conducted by the Grantee on the premises. During the performance of the activities specified in this License, the Grantee shall not unreasonably interfere with the use and enjoyment of the premises by the Grantor.

7. Access - During the term of this License, the Grantee, its agents, employees, or representatives shall have the right of access to and egress from the premises as needed and shall have the right to bring necessary equipment upon the premises in connection with the performance of the Grantee's activities as set out in Condition 1.

8. Title to Equipment, Fixtures - Title to all equipment, fixtures, appurtenances, and other improvements furnished and installed in connection with the Grantee's activities under this License shall remain with the Grantee.

9. Restoration - Upon termination of this License, the Grantee shall remove all its equipment, fixtures, appurtenances, and other improvements furnished and installed on the premises in connection with the Grantee's activities under this License. The Grantee shall restore the premises, when such restoration is required in connection with the Grantee's activities, to the extent reasonably practical, to the condition existing at the time of initiation of the Grantee's activities. With the consent of the Grantor, the Grantee may abandon Grantee-owned equipment, fixtures, appurtenances, and other improvements in place in lieu of restoration when it is in the best interests of the Grantee.

10. Successors in Interest - This License and the parties' commitments within, shall be binding on both parties, their successors, and assigns.

11. Funding - Obligations of the Grantee under this License shall be subject to the availability of funds appropriated by the Congress which the Grantee may legally spend for such purposes and nothing in this License implies that Congress will appropriate funds to perform this License.

12. Notices - All notices regarding the specific terms and conditions of this License, and within the restrictions of this License, shall be in writing and shall be deemed effectively given upon personal delivery, upon verified facsimile receipt, or upon mailing by registered or certified mail, postage prepaid, and addressed to the parties at the following respective addresses, or to such other persons or at such other addresses as may be designated in writing by either party to the other.

If to the Grantee:	If to the Grantor:
Katy Kates Realty Officer Department of Energy P.O. Box 2001 Oak Ridge, Tennessee 37831	Anthony D. Pantaleoni Vice President Crane Company 100 First Stamford Place Stamford, Connecticut 06902

13. Entire Agreement - This License represents the entire understanding of the parties on this matter and no oral statements or collateral documents (except as noted within) may modify this License.

14. Amendment - This License may not be amended or superseded except by an agreement in writing executed by the Grantor and Grantee.

That prior to execution of this License certain Conditions were deleted, revised, and/or added (with the additions being as set out below or as designated as Page(s) N/A and being made a part of this License) in the following manner:

Condition No. 3 is deleted in its entirety; Condition No. 15 is added.

15. Termination of Prior Instrument - Right-of-Entry REORDOER-9-95-0416 dated October 28, 1994 between the Grantor and Grantee, which provided for survey and exploration on the same premises is terminated as of the effective date of this License.

REAL ESTATE LICENSE NO.
REORDOER-7-95-0139

The above terms and conditions are acknowledged and agreed upon as indicated by the signatures affixed below:

GRANTOR:	Crane Company	GRANTEE:	U.S. Department of Energy
By:	A.D. Pontelli	By:	Katy Hales DOE Real Estate Office
Title:	V.P. Environment Health + Safety	Title:	Realty Officer
Date:	5/30/95	Date:	6-6-95

148447

Bechtel

Oak Ridge Corporate Center
151 Lafayette Drive
P. O. Box 350
Oak Ridge, Tennessee 37831-0350

Telephone: (423) 220-2000

Job No. 14501, FUSRAP Project
DOE Contract No. DE-AC05-91OR21949
Code: 7330/WBS: 133

NOV 25 1996

U.S. Department of Energy
Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, TN 37831-8723

Attention: James D. Kopotic, Site Manager
Former Sites Restoration Division

Subject: Chapman Valve Site - Publication of PRAR

Dear Mr. Kopotic:

Enclosed is a copy of the subject document, which is being published in accordance with your instructions (CCN 148320). All comments received on the previous draft of this document have been incorporated.

This document and all attachments were prepared under my direction or supervision in accordance with a system designed to ensure that the information submitted was properly gathered and evaluated. To the best of my knowledge and belief, they are true, accurate, and complete.

If you have any questions, please call me at 576-4718.

Sincerely,

M. E. Redmon
Project Manager - FUSRAP

EBS:onl:LR_1881.DOC
Enclosure: Publication of PRAR

cc: S. K. Oldham, w/3

Concurrence: T. King @ ___ M. Kucera MAK E. Speaks for ___

ACTION REQ'D	☐ YES	☒ NO	DUE DATE ___
RESPONSE TO CHRON NO. ___			
☐ FFA ☐ Permit ☐ Milestone ☐ OcR ☐ CCN ☐ CAR ☐ Mid-Yr ☐ Yr-End ☐ Periodic Rpt			

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<u>16, 18, 20, 22, 24, 26, 28, 30, 32, 36, & 38</u>

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DOE/OR/21949-408

Formerly Utilized Sites Remedial Action Program (FUSRAP)
Contract No. DE-AC05-91OR21949

Post-Remedial Action Report for the Chapman Valve Site

Indian Orchard, Massachusetts

November 1996

Printed on recycled/recyclable paper.

Z4.19 0186.1

DOE/OR/21949-408

POST-REMEDIAL ACTION REPORT

FOR THE

CHAPMAN VALVE SITE

INDIAN ORCHARD, MASSACHUSETTS

NOVEMBER 1996

Prepared for

United States Department of Energy

Oak Ridge Operations Office

Under Contract No. DE-AC05-91OR21949

By

Bechtel National, Inc.

Oak Ridge, Tennessee

Bechtel Job No. 14501

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ACRONYMS

AEC	U.S. Atomic Energy Commission
ALARA	as low as reasonably achievable
BNI	Bechtel National, Inc.
CHV	Chapman Valve Site
DAC	derived air concentration
DCG	derived concentration guide
DOE	U.S. Department of Energy
FUSRAP	Formerly Utilized Sites Remedial Action Program
HEPA	high-efficiency particulate air
IVC	independent verification contractor
ORNL	Oak Ridge National Laboratory
PMC	project management contractor
PPE	personal protective equipment
RCRA	Resource Conservation and Recovery Act
TCLP	toxicity characteristic leaching procedure

UNITS OF MEASURE

cm	centimeter
dpm	desintegrations per minute
ft	foot
g	gram
in.	inch
L	liter
µCi	microcurie
µg	microgram
ml	milliliter
mrem	millirem
pCi	picocurie
yr	year

1.0 INTRODUCTION

1.1 BACKGROUND

This report documents the remedial action conducted at the Chapman Valve site (CHV) in Indian Orchard, Massachusetts, from July to September 1995. The remedial action was conducted following the expedited protocol, which is an efficient, cost-effective, and environmentally acceptable approach for this small site. Use of this approach complied with state and local regulations.

Remedial activities at CHV were performed as part of the U.S. Department of Energy (DOE) Formerly Utilized Sites Remedial Action Program (FUSRAP). FUSRAP was established to identify and clean up or otherwise control sites where residual radioactive contamination remains from the early years of the nation's atomic energy program or from commercial operations causing conditions that Congress has authorized DOE to remedy. FUSRAP was established in 1974 and currently includes 46 sites in 14 states. CHV was designated for remedial action under FUSRAP in 1992.

FUSRAP objectives for CHV were to

- remove or otherwise control radioactive contamination above current DOE guidelines, and
- achieve and maintain compliance with applicable criteria for the protection of human health and the environment.

Bechtel National, Inc. (BNI), the project management contractor (PMC) for FUSRAP, assisted DOE's Oak Ridge Operations Office in the planning, management, and implementation of the cleanup of CHV. Oak Ridge National Laboratory (ORNL) was the independent verification contractor (IVC) for the remedial action.

1.2 HISTORY

CHV is located in Indian Orchard, a suburb of Springfield, Massachusetts, at 203 Hampshire Street (Figure 1). The site was formerly owned and operated by the Chapman Valve Manufacturing Company. The Crane Company has owned the site since 1981 but vacated the building in 1987. In 1991, the building was still standing and in reasonably good condition; however, harsh winter conditions had resulted in the deterioration of the building, and a structural inspection indicated that the roof was unsafe (Klotsas to Boyer 1992).

In 1948, Chapman Valve set aside an area in the western end of building 23 (Figure 2) for the machining of uranium rods for the Atomic Energy Commission's (AEC's) Brookhaven National Laboratory. Uranium operations were terminated on November 8, 1948, at which time Chapman Valve had in its possession more than 27,000 pounds of metal scrap, oxides, and sweepings. This material was removed from the site several months after the contract was completed.

1.3 EXTENT OF CONTAMINATION

A 1992 survey indicated that the residual uranium contamination found at CHV was typical of Manhattan Engineer District/AEC operations (ORNL 1992). The contamination was limited to interior areas and included floors, walls, and overhead beams.

The PMC performed additional radiological surveys in November and December 1994 to supplement and refine existing survey information (BNI 1995). Characterization findings confirmed the presence of contamination in the western end of Building 23 (Figures 2, 3, 4, and 5). These findings are in agreement with historical process information obtained during interviews conducted with a former Chapman Valve supervisor. According to the supervisor, a temporary wall was constructed across the center portion of the building from floor to ceiling between columns A-7 and B-7 (Figure 2).

Additional contamination was identified in the southwest corner of grid A-1 at the former location of a chip burner that exhausted to the atmosphere through a nearby window. (Grid numbers are derived from the northwestern column number of the grid.) The exhaust location and the shape of the roof of the building caused more contamination to be deposited on the south roof than on the north roof, as indicated by characterization measurements. The uranium storage area was located in Room B-4 on the south side of the building between columns B-4 and B-7 (Figure 2). Characterization results showed no subsurface soil contamination below the floor slab.

Levels of contamination decreased from west to east in the building, and survey and sampling results from the east end indicated near-background radiological conditions. The exterior of the building, except for the roof and several locations on the west and south exterior walls, indicated near-background radioactivity.

Chemical sampling results indicated the presence of lead-based paint on the cranes, electrical boxes, and structural steel. Asbestos was found in the composite roof material and the electrical box insulation.

2.0 REMEDIAL ACTION GUIDELINES

Radioactive contamination at CHV consisted primarily of natural uranium. Table 1 lists the DOE residual contamination guidelines for release of formerly contaminated properties for use without radiological restrictions. The guidelines listed in Table 1 were applied to the crane, floor, and drain lines. These guidelines were adopted by DOE based on their compatibility with U.S. Environmental Protection Agency (EPA) criteria for remedial action found in 40 CFR 192, "Uranium Mill Tailings Remedial Action Program," and are contained in DOE Order 5400.5, "Radiation Protection of the Public and the Environment."

The remedial action approach for the Chapman Valve site was somewhat unique. Site characterization was performed in 1995, and the results were documented in a technical memorandum (BNI 1995a). A hazard assessment, which focused on future use radiological dose, was completed to assist in determining the appropriate remedial action methods (BNI 1996b). Based on this hazard assessment, supplemental standards as defined in DOE Order 5400.5 were approved by DOE-HQ (Wagoner to Price 1995). Under the future use scenario involving a demolition worker performing commercial operations with no radiological controls, the predicted radiological dose was 5.6 mrem/yr. The predicted dose for a worker at a facility where steel from the building would be smelted was 16 mrem/yr (O'Connell to Kopotic 1995). These scenarios were considered conservative but realistic given future plans for the structure. Based on the results of the hazard assessment and the approved supplemental standards, and consistent with DOE's policy of reducing contaminant levels as low as reasonably achievable (ALARA), the remedial action efforts focused on eliminating removable contamination and decontaminating the most highly contaminated areas of the building. Further details of the remedial action are provided in Section 3.0.

3.0 REMEDIAL ACTION

Because of the potential for salvage and reuse of the crane and the possibility that the base slab would remain intact following future building demolition, the crane and floor were decontaminated to comply with the DOE residual contamination guidelines listed in Table 1. It was determined that a supplemental guideline of 15,000 dpm/100 cm^2 average for the horizontal surfaces of an entire truss was protective of a future demolition worker and that contamination on the walls and roof contributed only a negligible dose to the worker; therefore, decontamination of the wall and roof surfaces was not necessary. However, in accordance with DOE's ALARA policy, removable contamination on horizontal wall surfaces was remediated. This remedial action approach was approved by the Massachusetts Department of Environmental Protection (Weinberg to Pantaleoni 1995) and the Massachusetts Department of Public Health (O'Connell to Kopotic 1995).

During remedial action, areas under failing roof sections were identified, and appropriate precautions such as posting, barricading, and limiting access were taken to ensure the safety of workers.

3.1 DECONTAMINATION ACTIVITIES

Decontamination activities at CHV lasted approximately eight weeks, from July to September 1995. All remediation efforts were confined to the interior of Building 23. Designation and characterization surveys revealed contamination on interior building surfaces in the western third of Building 23, between columns 1 and 8, and on a bridge crane located in grid A-23 (see Figure 2). During floor removal, contamination above criteria was found under a concrete ramp just inside the west equipment door and in a drain line in room B-4. Techniques used in the remedial action are summarized in Table 2.

Volume reduction, waste minimization, and cost saving techniques employed during the remedial action included segregating, sampling, and surveying the wastes produced. Using washable protective clothing and a personnel contamination monitor were also implemented as cost saving and waste minimization techniques. Total costs for remedial action are presented in Table 3. Specific examples of waste minimization and cost saving measures at CHV are described below.

- Concrete and debris were surveyed to determine whether contaminant levels were above criteria. The material below criteria was placed inside the building and left onsite, with property owner concurrence.

- Paint removed from the crane was treated by acid dissolution and solidified with concrete to reduce the content of leachable lead to the extent that toxicity characteristic leaching procedure (TCLP) results were below the limits specified in the Resource Conservation and Recovery Act (RCRA). The levels of lead in the resulting concrete monoliths were determined to be below radiological guidelines, and the monoliths were left onsite, with the concurrence of the property owner and the State of Massachusetts.

- Remediation workers wore washable Protech-2000 coveralls rather than standard Tyvek protective clothing. Using the washable coveralls reduced waste volumes associated with disposable protective clothing. As an additional benefit, workers found that the Protech-2000 fabric coveralls were more comfortable to wear than Tyveks. Interstate Nuclear Services was contracted to provide laundry service for the washable protective clothing.

- An Eberline PCM-2 personnel contamination monitor was used at the access control point instead of a standard manual frisk, reducing the time required to ensure that workers exiting the controlled area are not contaminated. The PCM-2 monitor completes the contamination check in one minute; a standard hand frisk takes approximately five minutes. The PCM-2 also ensures a more consistent exit survey than the conventional hand frisk performed by the individual.

The following sections contain descriptions of decontamination techniques for each remediated area.

10-Ton Bridge Crane

Residual surface contamination was removed from the bridge crane located at grid A-23 (see Figure 2) by hand scraping with putty knives and vacuuming. The crane was decontaminated without disturbing painted surfaces except for several small areas where it was necessary to remove the paint along with the contamination. Before the paint was removed, the high-efficiency particulate air- (HEPA-) filtered vacuums were emptied to minimize the amount of material requiring treatment. Very-small-quantity-generator status permitted treatment and disposal of the lead-based paint onsite. The paint was dissolved in an acid solution and then mixed with concrete. The resulting concrete monoliths were surveyed and determined to be below the DOE residual surface contamination guidelines listed in Table 1. With the concurrence of the property owner and the State of Massachusetts, the concrete monoliths were left onsite. The wooden decking was removed from the crane and decontaminated using the same technique that was used on the crane. The decontaminated boards were also left onsite.

Overhead Trusses

The upper horizontal surfaces of trusses in the main bay of Building 23 from truss 1 to truss 8 were decontaminated by using HEPA-filtered vacuums to remove contaminated dust. In one area of truss 2, abrasive decontamination was required to meet the supplemental limits set for CHV. The radioactive dust generated from the operation was mixed with concrete to eliminate the potential for fugitive dust during transport and handling of the waste. The concrete/dust was placed in an intermodal container for disposal at a licensed disposal facility.

Horizontal Wall Surfaces

Horizontal surfaces on the walls of the main bay from column 1 to column 8 were decontaminated with a HEPA-filtered vacuum. Decontaminated surfaces included the crane rail, window sills, radiators, and a large pipe on the north wall. Preliminary hazard assessment calculations indicated that the wall contamination did not contribute significantly to the dose to a future demolition worker; however, because the dose to the demolition worker was primarily from removable contamination, the horizontal surfaces were vacuumed to remove the contamination, in accordance with DOE's ALARA policy. The radioactive dust generated from the operation was mixed with concrete to eliminate the potential for fugitive dust during transport and handling of the waste. The concrete/dust was placed in an intermodal container for disposal at a licensed disposal facility.

Floor

A backhoe and skid steer loader equipped with a hoe-ram attachment were used to remove the wooden block and concrete pads above the base slab in the main bay and the B-series rooms on the south side of Building 23 between columns 1 and 8. The wooden blocks and concrete pads were considered clean material based on volumetric sampling results (BNI 1995b) and were placed in a clean pile. This

material was left onsite, with the concurrence of the property owner. The floor was vacuumed with HEPA-filtered vacuums, and the dust was collected.

Self-tapping steel floor anchors were removed with a jackhammer and surveyed. Any floor anchors containing above-background contamination were placed in the intermodal container for disposal at the licensed disposal facility. Areas of the floor that required further decontamination were mechanically cleaned with side grinders equipped with vacuum shrouds connected to HEPA-filtered vacuums. During the survey of the floor, it was determined that part of the ramp at the west equipment door of Building 23 needed to be removed to provide access to contamination under the ramp. The portions of the ramp requiring removal were broken up with a jackhammer and a skid steer loader equipped with a hoe-ram attachment.

Shovels were then used to remove the contaminated soil below the ramp. All radioactive dust collected during the decontamination of the floor was mixed with concrete to eliminate the potential for fugitive dust during transport and handling of the waste. The concrete/dust was then placed in an intermodal container for disposal at a licensed disposal facility.

Drain Line

A ductile iron drain line discovered in Room B-4 (see Figure 3) after removal of the wooden blocks was determined to be contaminated above criteria. The drain line was removed by breaking the concrete with jackhammers and a skid steer loader equipped with a hoe-ram attachment. Lead seals in the joints of the pipe were segregated by breaking the pipe away from the seal. A total of 145 ft of 4-in. drain line was removed and placed in an intermodal container for shipment to the disposal facility. The lead seals were surveyed and released for recycling at a local lead recycling company.

3.2 CONTAMINATION CONTROL DURING REMEDIAL ACTION

During the remedial action, engineering and administrative controls such as dust control, hazardous work permits, and personal protective equipment (PPE) were used to protect remediation workers and members of the public from exposure to radiation in excess of applicable standards and in accordance with a site-specific safety and health plan for CHV.

All personnel working in contaminated areas were required to wear protective coveralls, hard hats, safety glasses, hearing protection, boots, and gloves. If conditions warranted, additional PPE such as face shields were used. Site conditions did not necessitate the use of personnel respiratory protection except in the HEPA-filtered vacuum changeout tent.

Workers leaving radioactively restricted work areas were scanned at the control point by a personnel contamination monitor and subjected to a hand and foot frisk to ensure that they were not contaminated and to prevent the spread of contamination.

The potential primary exposure pathways from radioactive material to workers and members of the general public were inhalation and ingestion of radioactively contaminated airborne dust generated during the remedial action. HEPA filtration units were used to control the spread of dust and minimize the potential for contaminants to become airborne. In addition, a fine water mist was sprayed to control dust during floor removal and during transport of material to the intermodal container. All equipment used in the controlled area was surveyed before being released from the site.

During remediation, particulate air monitoring devices were placed in the areas being remediated. Monitoring locations were selected to provide data for the worst-case scenario. Concentrations of uranium-238 ranging from 1.5×10^{-14} to 5.8×10^{-13} µCi/ml were conservatively derived by collecting air particulate samples daily from lapel air samplers worn by workers. After gross activity per volume of air passing through the filter was determined, the source of all activity on the filter was conservatively assumed to be uranium-238. The derived air concentrations (DACs) were then compared with the applicable DOE guideline, which is a DAC of 2.0×10^{-11} µCi/ml for occupational exposures to airborne uranium-238 (DOE Order 5480.11).

Area air particulate sampling was also performed adjacent to areas being remediated to ensure that no member of the general public was exposed to radioactivity above DOE guidelines (DOE Order 5400.5). Because all remediation activities were conducted inside Building 23 and there were no open vents in the building, it was determined that exterior air particulate monitoring was not required. Data were collected daily from an Eberline RAS-1 high-volume monitor and counted after four days to allow for radon decay. The limits in DOE Order 5400.5 are derived concentrations guides (DCGs); a DCG is the concentration of a particular radionuclide that would provide an effective dose equivalent of 100 mrem/yr (the DOE basic dose limit) to an individual continuously exposed to the radionuclide by one pathway for an entire year. Concentrations of uranium-238 measured by area particulate monitors ranged from 1.5×10^{-15} to 3.1×10^{-13} µCi/ml. The DCG is 2.0×10^{-12} µCi/ml for uranium-238.

4.0 POST-REMEDIAL ACTION SAMPLING/SURVEYING RESULTS

After each portion of the property was decontaminated, a radiological survey of the area was conducted to confirm that all radioactive contamination above the cleanup criteria had been removed. Initial post-remediation surveys were conducted by Thermo Nuclear Services (TN) on behalf of the PMC. Survey techniques used during post-remediation and verification surveys included direct surface contamination measurements, removable contamination measurements, external gamma radiation exposure rate measurements, and soil sampling. The initial post-remediation surveys were conducted in accordance with TN procedures and PMC instruction guides. The IVC performed independent verification surveys of the floor, trusses, and bridge crane using survey techniques that were similar or identical to those used by TN. The IVC survey data will be issued in a separate verification report for CHV.

4.1 POST-REMEDIAL ACTION SURVEY DATA

Post-remedial action survey data for each decontaminated area are discussed in the following sections.

10-Ton Bridge Crane

Post-remedial action activities on the crane included surveys of direct and removable contamination and sampling of paint and dust residues to determine whether the waste should be classified as a RCRA mixed waste. Figure 6 shows the survey locations, and Table 4 presents the survey data. A total of 169 locations were surveyed. Direct alpha and beta-gamma average surface contamination readings were 44 and 520 dpm/100 cm^2, respectively. Average removable alpha and beta-gamma readings were 3 and 42 dpm/100 cm^2, respectively. The crane was decontaminated to comply with the DOE residual contamination guidelines listed in Table 1.

Table 9 presents sampling results used in bench-scale tests to determine whether treatment of the paint and dust residues was a feasible waste minimization technique. After the material was dissolved in acid and stabilized in a concrete monolith, a sample was collected to determine the classification of the waste. The sample result for TCLP lead was 338 μg/L; for cadmium the result was 3.0 μg/L. Both results are well below the RCRA limits of 5,000 μg/L for lead and 1,000 μg/L for cadmium.

Overhead Trusses

Surveys of fixed and removable contamination were conducted on the trusses to determine the effectiveness of the decontamination effort. The results of these surveys are included in Table 5. The maximum fixed averages per truss for alpha and beta-gamma contamination were 2,114 and 12,261 dpm/100 cm^2, respectively. Both of these readings, which were from truss 2, were below the supplemental limit of 15,000 dpm/100 cm^2 average per truss. All readings for removable contamination were below 1,000 dpm/100 cm^2. An additional survey conducted on non-horizontal surfaces and between welded angles of the trusses confirmed that the supplemental limits had not been exceeded and that these areas did not require decontamination.

Horizontal Wall Surfaces

The ALARA approach to decontamination of the horizontal wall surfaces was to remove the contamination that exhibited a potential for resuspension or migration. Survey locations for the west, north, and south walls are shown in Figures 7, 8, and 9. Survey results for the west, north, and south walls are presented in Tables 6, 7, and 8. The maximum and average removable alpha measurements for the west wall were 164 and 17 dpm/100 cm^2; for the north wall, the maximum and average measurements were 850 and 35 dpm/100 cm^2; and measurements on the south wall were 111 and 17 dpm/100 cm^2, respectively. The maximum and average removable beta-gamma measurements for the west wall were

191 and 70 dpm/100 cm^2; for the north wall, the measurements were 3,197 and 58 dpm/100 cm^2; and measurements on the south wall were 319 and 45 dpm/100 cm^2, respectively. Except for a single reading on the north wall, all measurements were below the criterion of 1,000 dpm/100 cm^2 for removable contamination.

Floor

Figure 10 shows survey results of 1,598 survey locations on the floor of Building 23; all results were below DOE Order 5400.5 criteria. Fixed beta-gamma contamination was the contaminant of concern on the floor; therefore, only fixed beta-gamma measurements are summarized in Figure 10. The average fixed beta-gamma measurement was 914 dpm/100 cm^2; the minimum was -872 dpm/100 cm^2; and the maximum was 4,934 dpm/100 cm^2. (Negative numbers indicate that the measurement was less than the minimum detectable activity and that after background was subtracted, the numerical value was negative.) A soil sample was also collected in the area where part of the west equipment door ramp was removed. Analytical results are presented in Table 9. The analytical result for uranium-238 was 2.0 pCi/g; for radium-226, the result was 0.47 pCi/g; and for thorium-232, the result was 0.41 pCi/g. A site-specific uranium guideline for CHV was not determined; however, these results were well below the typical uranium-238 guideline of 35 to 50 pCi/g for a FUSRAP site. The radium-226 and thorium-232 results are below the DOE Order 5400.5 criterion of 5 pCi/g for surface soils.

Drain Line

Following the removal of 145 ft of the 4-in. drain line in room B-4, three composite samples were collected in the trench to determine whether the decontamination effort was successful (see Table 9). The maximum level for uranium-238 was 0.62 pCi/g; for radium-226, the maximum was 0.47 pCi/g; and for thorium-232, the maximum was 0.50 pCi/g. A site-specific uranium guideline for CHV was not determined; however, the results are well below the typical uranium-238 guideline of 35 to 50 pCi/g for a FUSRAP site. The radium-226 and thorium-232 results are below the DOE Order 5400.5 criterion of 5 pCi/g for surface soils. Contaminated drain piping was placed in an intermodal container and shipped to a licensed facility for disposal.

4.2 HAZARD ASSESSMENT/EXPOSURE SCENARIOS

A hazard assessment was prepared to document the post-remediation condition of CHV (BNI 1995b). A summary of the exposure scenarios is included in this PRAR, and the results of the hazard assessment are summarized in Table 10. The hazard assessment also includes calculations to determine the total mass of uranium at CHV and the volumetric activity of the rubble resulting from building demolition.

The hazard assessment is based on the fact that contamination at CHV is found only in limited portions of Building 23. The total curie content of all affected building components is approximately 6 μCi. In addition, Building 23 is deteriorating, abandoned, and likely to be demolished.

The hypothetical demolition phases representing each of the building components, potential reuse or recycling of the components, and significant conditions and assumptions are described in the following exposure scenarios.

Roof Demolition

Contamination is consolidated into built-up roofing material. The primary exposure pathway is inhalation of contaminated dust. The primary dust-producing activities would be destroying the roof with heavy equipment (allowing the roof components to fall to the ground) and using heavy equipment to load the roof rubble into trucks. This scenario is plausible but conservative because the roof contains asbestos and the likely scenario would include appropriate asbestos controls, which would mitigate the release of airborne radioactivity and limit worker exposure.

Wall Demolition

Large quantities of dust would be produced during the use of a wrecking ball (or other forceful demolition technique) to dismantle the walls of Building 23. The impact of the ball would be approximately 20 ft above the ground and would dislodge surface contamination on the walls and glass surface. More dust would be produced as clean portions of the brick, mortar, and glass were pulverized. To maximize the potential exposure scenario, it is assumed that demolition is performed one wall at a time, and the rubble from one wall is removed before the next wall is demolished. In addition to the initial plume, contaminated dust would be resuspended as rubble is removed.

Structural Steel Demolition

No airborne contamination would result from demolition of the exposed steel surfaces because the surfaces were decontaminated during remedial action. Some areas such as gusset plates and intersections of steel members that were difficult to access could be a source of airborne contamination as the trusses are removed; therefore, it is assumed that some of the contamination from inaccessible areas will be dislodged as the steel building framework is dismantled. The contamination on the inaccessible surfaces is assumed to be the same as that on the exposed surfaces surveyed during characterization. The contamination is from deposited dust, which is assumed to be 0.1 cm thick.

The primary potential exposure pathways are inhalation and ingestion. The activity generating the most airborne contamination will be toppling the supporting beams and allowing the trusses to fall to the ground, where they will subsequently be salvaged for scrap. It is assumed that torch cutting will be performed to size the metal for transport. Airborne contamination will be released during the felling,

cutting, and removal of the trusses. Assuming that all trusses are felled in a very short time period, the contaminated plume from the felling of any one truss will not have dissipated when the next truss is felled; therefore, the plume of airborne contamination will include all the contamination released from the trusses.

Personal contamination (and subsequent ingestion) will potentially occur as the trusses are handled during cutting and removal.

Steel Recycling

Potential exposures from reuse or recycling of the steel were considered for three activities:

- torch cutting the steel to the size necessary for smelting,
- sand blasting the steel to remove lead-based paint so that the smelter will accept the steel, and
- disposing of the slag after the steel is smelted.

Floor Removal

The floor of Building 23 is composed of concrete, fire brick, or wooden blocks, with wooden blocks covering more than 90 percent of the floor area. A concrete base slab is under the fire bricks and wooden blocks. Isolated areas of contamination were found on the surface of the floor, in adhesive between the wooden blocks, in gaps in the fire brick, and possibly on the concrete base slab. The dose was calculated for a potential future worker removing the blocks left onsite during remedial action.

Reuse of Building Materials

The dose was calculated for an individual living in a small building constructed with the contaminated materials from CHV (Table 10). The building walls are constructed of materials from the west wall, the ceiling includes the roofing material, and the floor is constructed of the wooden blocks from the floor of Building 23.

4.3 CONCLUSIONS OF HAZARD ASSESSMENT

Residual contamination on the roof and walls of Building 23 will not result in a radiation dose above 100 mrem to any member of the general public. The maximum dose to a hypothetical demolition worker is estimated to be 5.6 mrem, and offsite exposure to the general public would be much less. Brick comprises approximately 64 percent of the mass of the contaminated portion of Building 23. The natural (background) radioactivity found in brick at the site is approximately 6.4 pCi/g. The radioactivity added by the surface contamination ranged from 0.18 to 1.5 pCi/g: an increase of approximately 23 percent. Because of the low radiation dose calculated for roof and wall contamination (and the fact that these calculations were very conservative and likely overestimated the dose), supplemental limits for these

building components would probably be the same as those used for the current contamination levels, and no remedial action would need to be performed in these areas. The use of a hazard assessment and the development of supplemental limits resulted in a cost saving of $2 million for the remedial action at CHV.

5.0 POST-REMEDIAL ACTION STATUS

Analytical results from post-remedial action surveys indicate that the levels of radioactivity in the remediated areas meet applicable DOE cleanup guidelines. A summary of remedial action at CHV is provided in Table 11. The IVC has reviewed the post-remedial action surveys and results and has verified that the remediated areas comply with the established DOE guidelines for the site. No areas of contamination above DOE guidelines or supplemental limits remain at the site.

After completing the verification survey, the IVC will report its findings and recommendations to DOE Headquarters and the DOE Oak Ridge Operations Office. DOE will review the report to verify that the remedial action was successful, and a certification docket will then be prepared. The certification docket officially certifies that the site has been successfully remediated to established criteria. The issuance of the certification docket will be documented through publication of a notice in the *Federal Register*.

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GLOSSARY

Alpha-emitting - See **Radiation**

Ambient Background Radiation - Ambient background radiation refers to naturally occurring radiation emitted from either cosmic (e.g., from the sun) or terrestrial (i.e., from the earth) sources. Exposure to this type of radiation is unavoidable, and its level varies greatly depending on geographic location. For example, New Jersey typically receives 100 millirem per year (mrem/yr), Colorado receives about 115 mrem/yr, and some areas in South America receive up to 7000 mrem/yr. Naturally occurring terrestrial radionuclides include uranium, radium, potassium, and thorium (see **Radionuclide**). The dose levels do not include the concentrations of naturally occurring radon inside buildings.

Beta-gamma-emitting - See **Radiation**

Centimeter - A centimeter (cm) is a metric unit of measurement for length; 1 inch is equal to 2.54 cm; 1 foot is equal to approximately 30 cm.

Contamination - The term "contamination" is used generally to mean a concentration of one or more radioactive materials that exceeds naturally occurring levels. Contamination may or may not exceed the DOE cleanup guidelines.

Disintegrations per minute - Disintegrations per minute (dpm) is the measurement indicating the amount of radiation being released from a substance per minute.

Dose - As used in this report, dose is actually dose equivalent and is used to relate absorbed dose (mrad) to an effect on the body. Dose is measured in mrem. For comparison, a dose of 500,000 mrem to the whole body within a short time causes death in 50 percent of the people who receive it; a dose of 5,000,000 mrem may be delivered to a cancerous tumor during radiation treatment; normal background radiation at or near sea level results in an annual dose of about 100 mrem; DOE radiation protection standards limit the dose that may be received by members of the general public to 100 mrem/yr above background levels; living in a brick house typically results in a dose of about 75 mrem/yr above the background level.

Exposure Rate - Exposure rate is the rate at which radiation imparts energy to the air. Exposure is typically measured in microroentgens (μR), and exposure rate is typically expressed as μR/hr. The dose to the whole body can be approximated by multiplying the exposure rate by the number of hours of exposure. For example, if an individual were exposed to gamma radiation at a rate of 20μR/hr for 168 hr/week (continuous exposure) for 52 weeks/yr, the whole-body dose would be approximately 175 mrem/yr.

Gamma Radiation - See **Radiation**

Meter - A meter (m) is a metric unit of length; 1 m is equal to approximately 39 inches.

Microroentgen - A microroentgen (μR) is a unit used to measure radiation exposure. For further information, see **Exposure Rate**.

Millirem - The millirem (mrem) is the unit used to measure radiation dose to man. The DOE dose limit is 100 mrem above background radiation levels within any one-year period for members of the general public. Naturally occurring radioactive substances in the ground result in a yearly exposure of about 100 mrem to each member of the population. To date, no difference can be detected between the health of population groups exposed to 100 mrem/yr above background and the health of groups who are not exposed.

Natural Background Radiation - Natural background radiation refers to radiation emitted from the naturally occurring radionuclides found in manmade materials. The concentrations of the radionuclide, and thus the radiation, will vary widely because of variation in the composition of the materials.

Radiation - There are three primary types of radiation: alpha, beta, and gamma. Alpha radiation travels less than an inch in air before it stops, and it cannot penetrate the outer layers of human skin. Beta radiation can penetrate the outer layers of skin but cannot reach the internal organs. Gamma radiation, the most penetrating type, can usually reach the internal organs.

Radionuclide - Radioactive elements are also referred to as radionuclides. For example, uranium-235 is a radionuclide, uranium-238 is another, thorium-232 is another, and so on.

Remedial Action - Remedial action is a general term used to mean cleanup of contamination that exceeds DOE guidelines. It refers to any action required so that a property may be certified as being in compliance with guidelines and may therefore be released for future use. Remedial action also includes restoring remediated properties to their original conditions as far as possible.

Uranium - Uranium is a naturally occurring radioactive element. The principal use of refined uranium is for the production of fuel for nuclear reactors. Uranium in its natural form is not suitable for use as a fuel source.

FIGURES

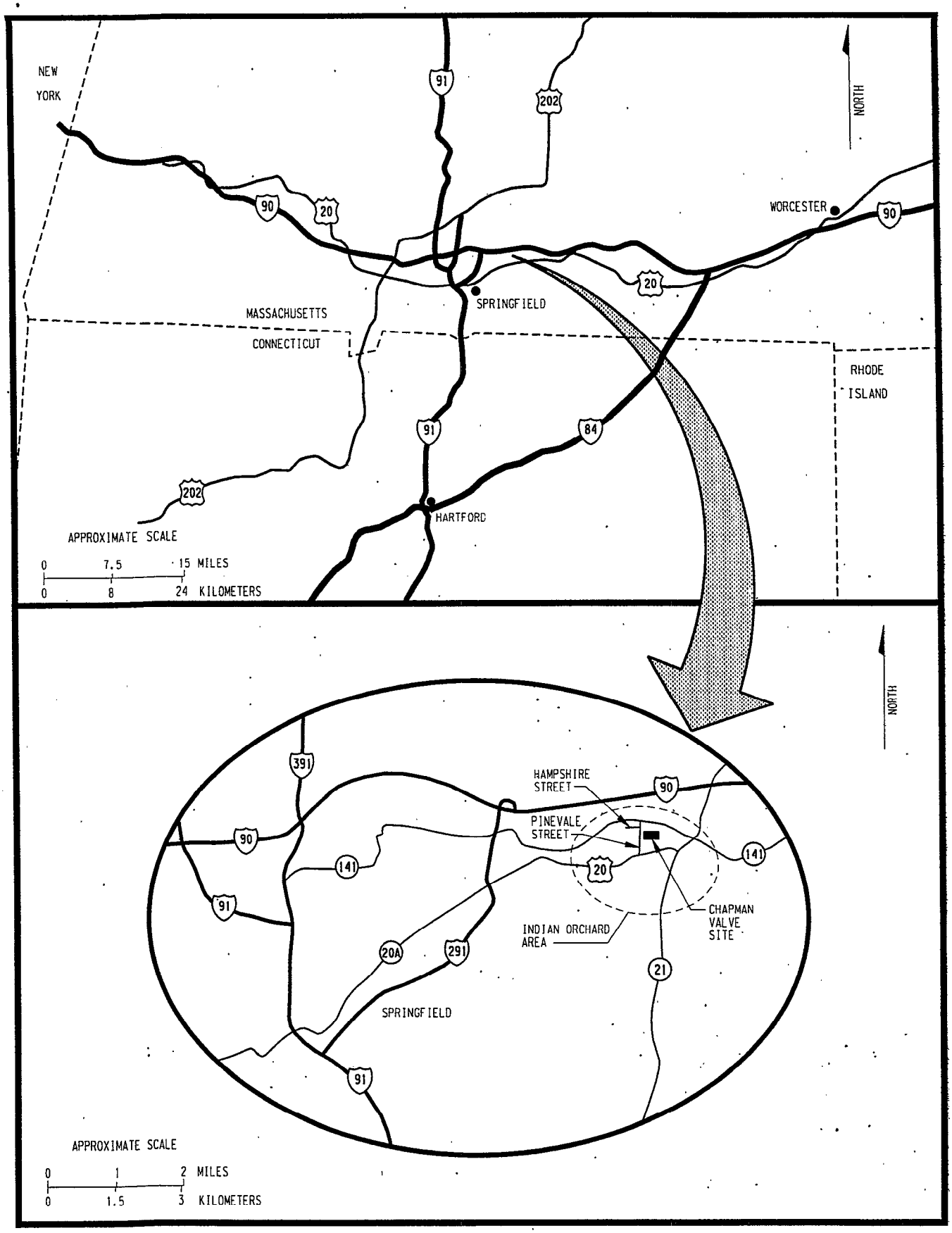

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Figure 1
Approximate Location of Chapman Valve Site

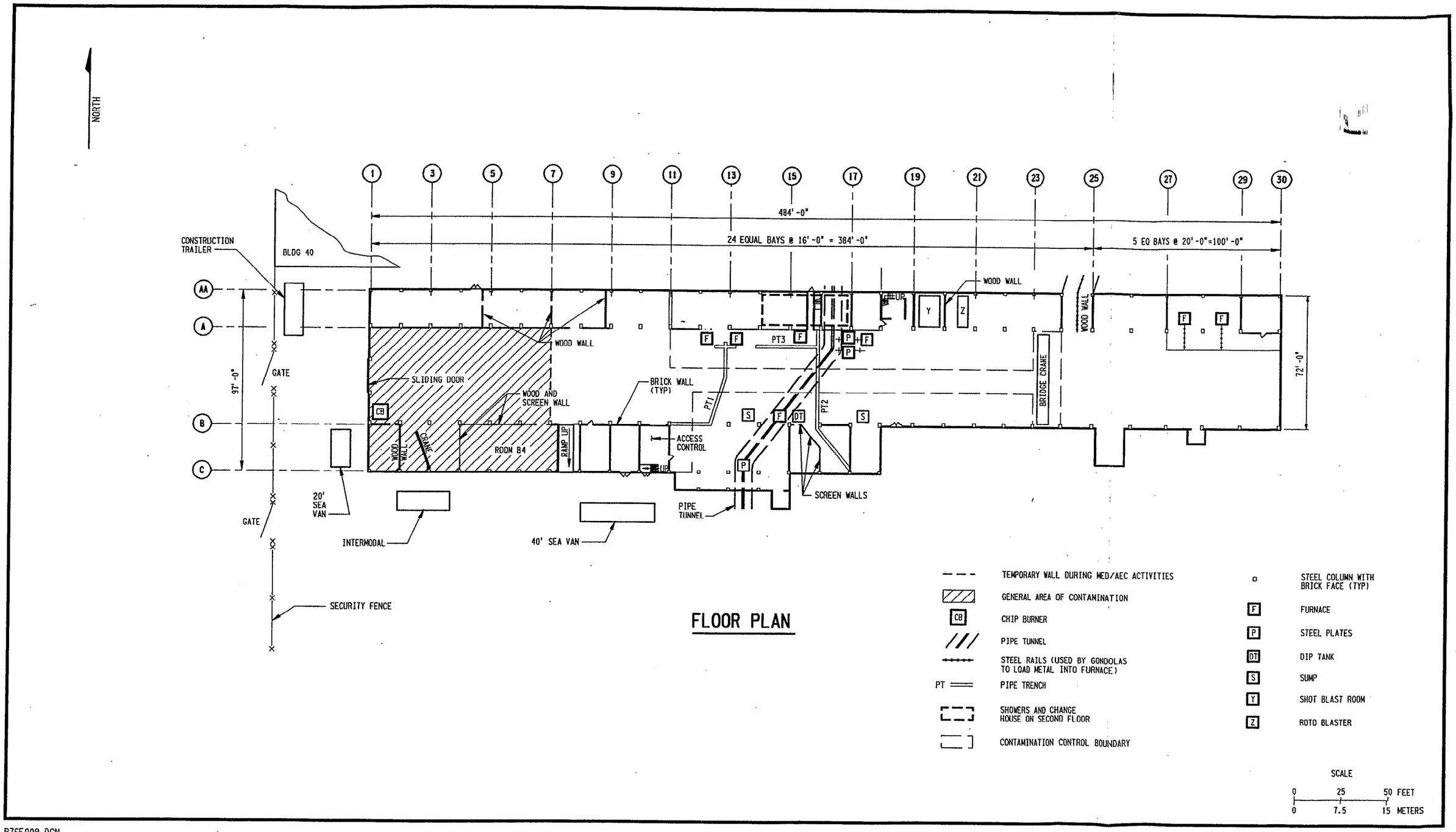

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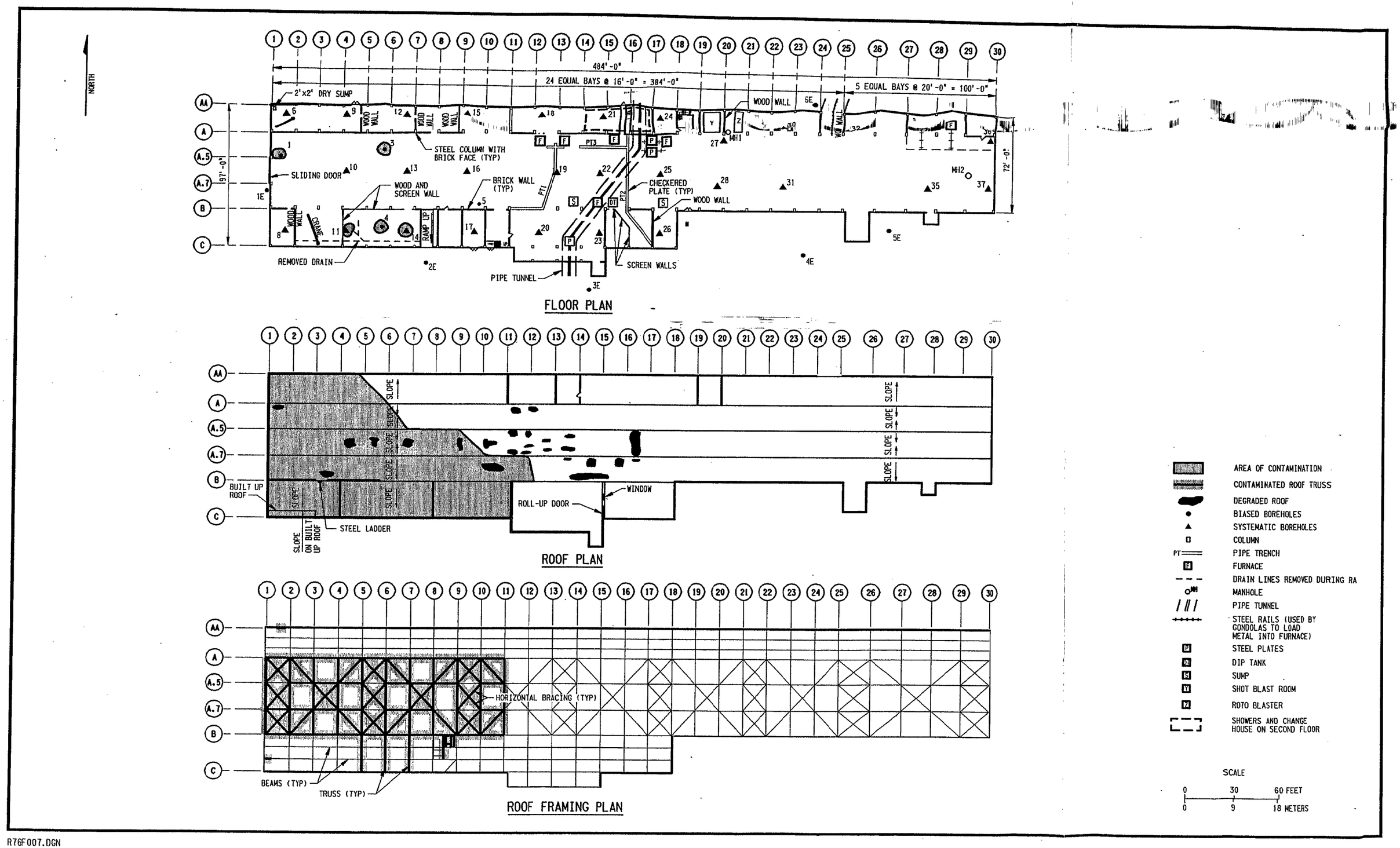
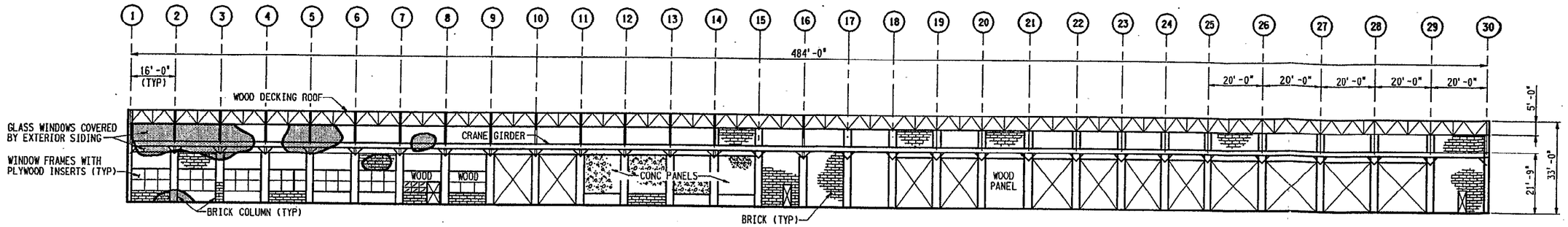
Figure 2
Chapman Valve
Building 23 - Plan View

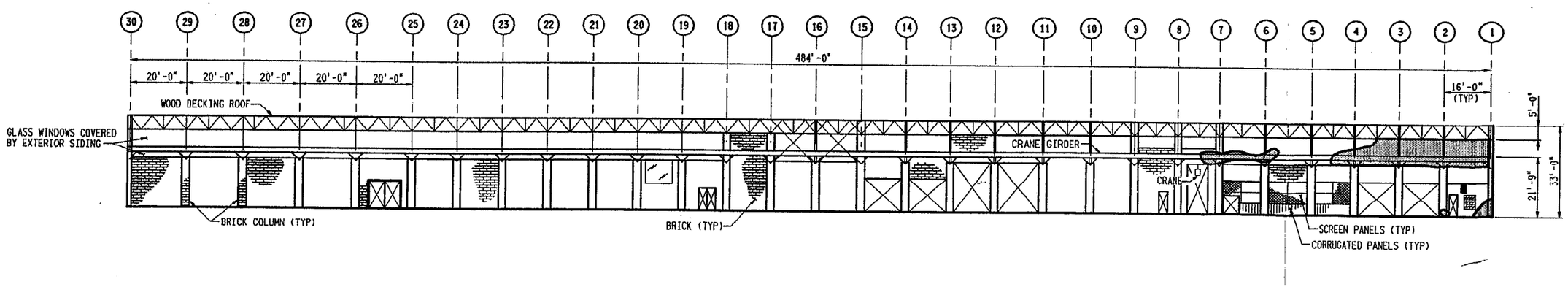


R76F007.DGN

Figure 3
Chapman Valve - 1994 Characterization Results
Plan Views



WALL A - ELEVATION LOOKING NORTH



WALL B - ELEVATION LOOKING SOUTH

CONTAMINATED SURFACE

INDICATES OPENING OR DOORWAY

SCALE

0 20 40 FEET

0 6 12 METERS

R76F006.DGN

Figure 4
Chapman Valve - 1994 Characterization Results
Interior North and South Walls

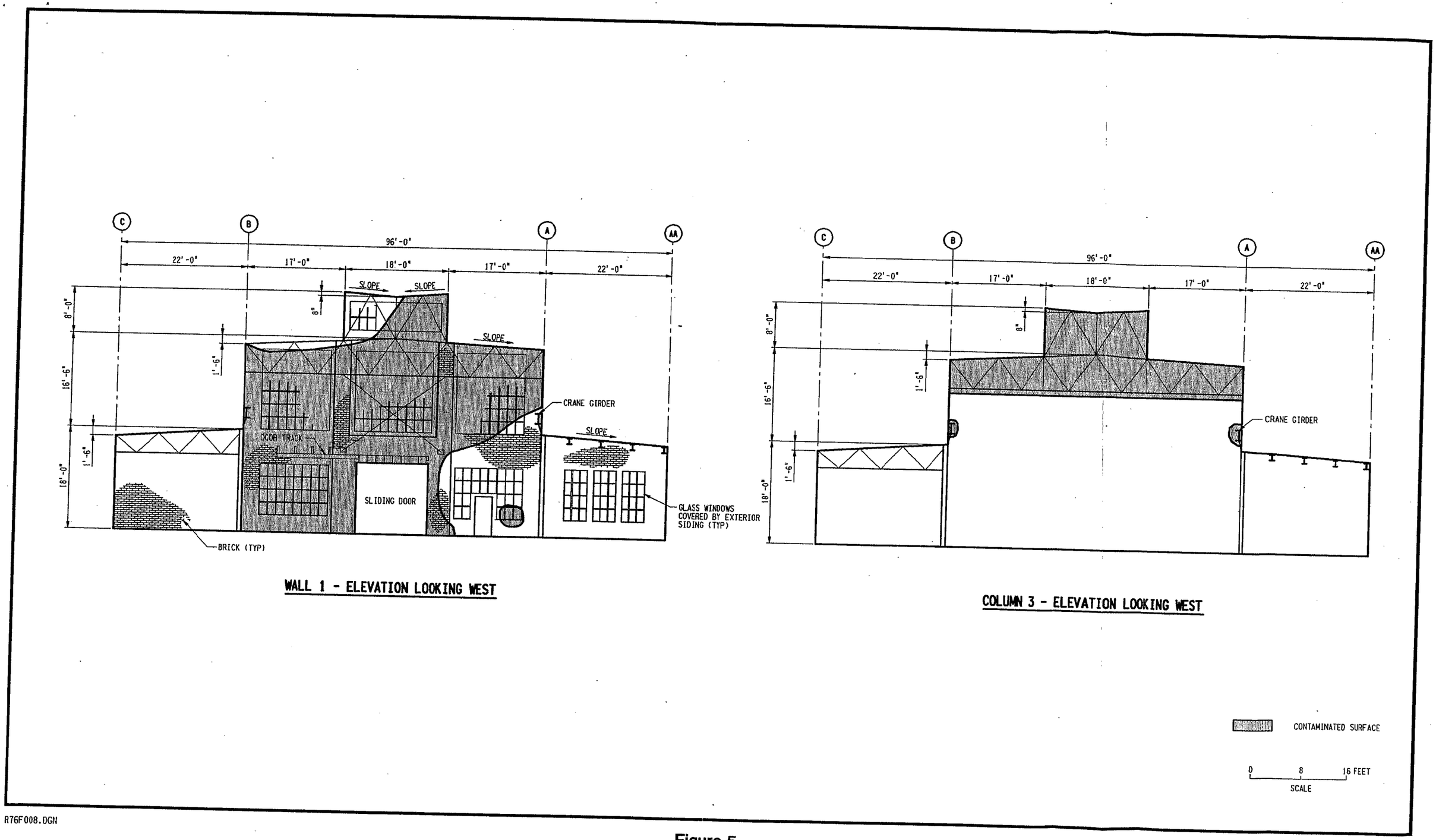
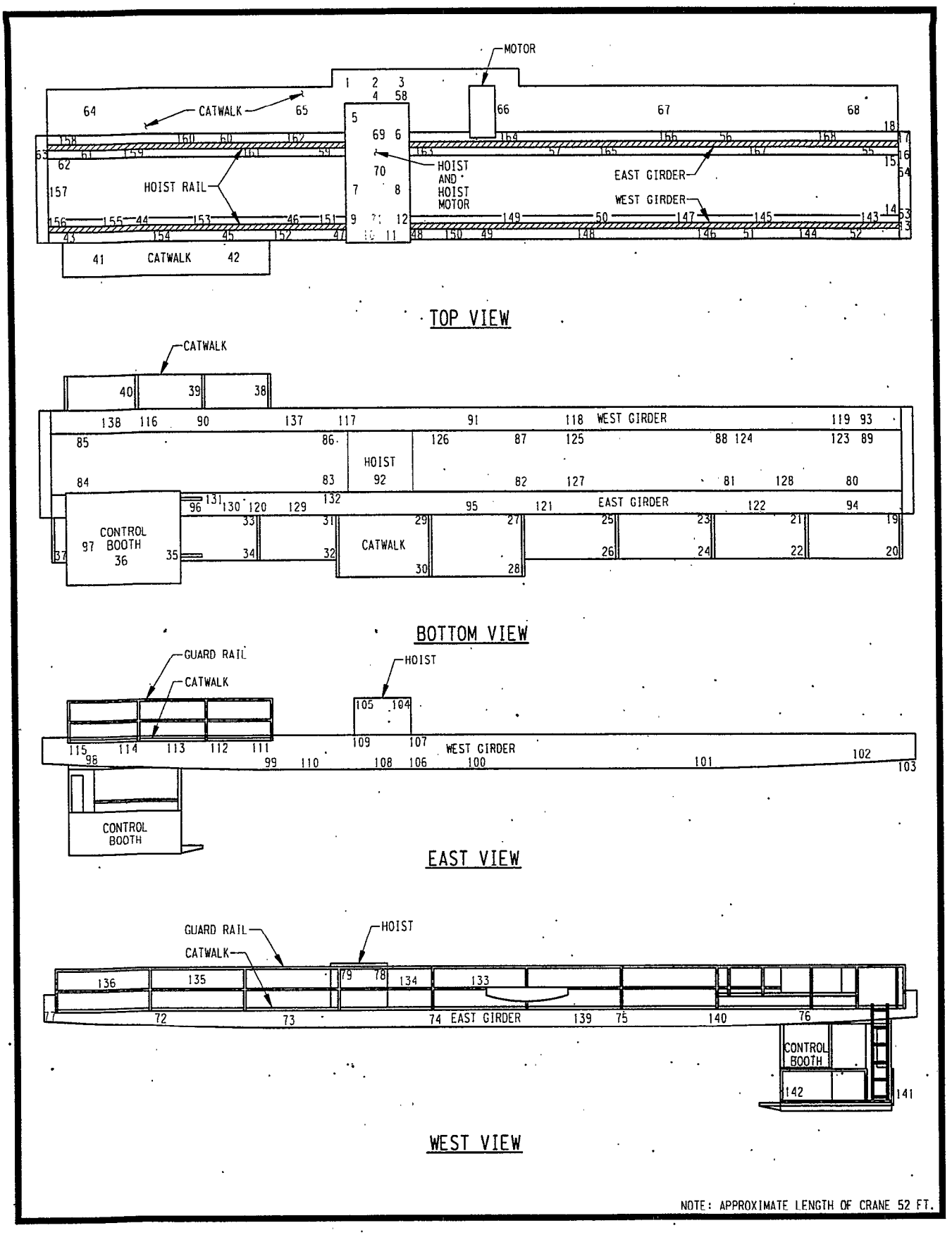
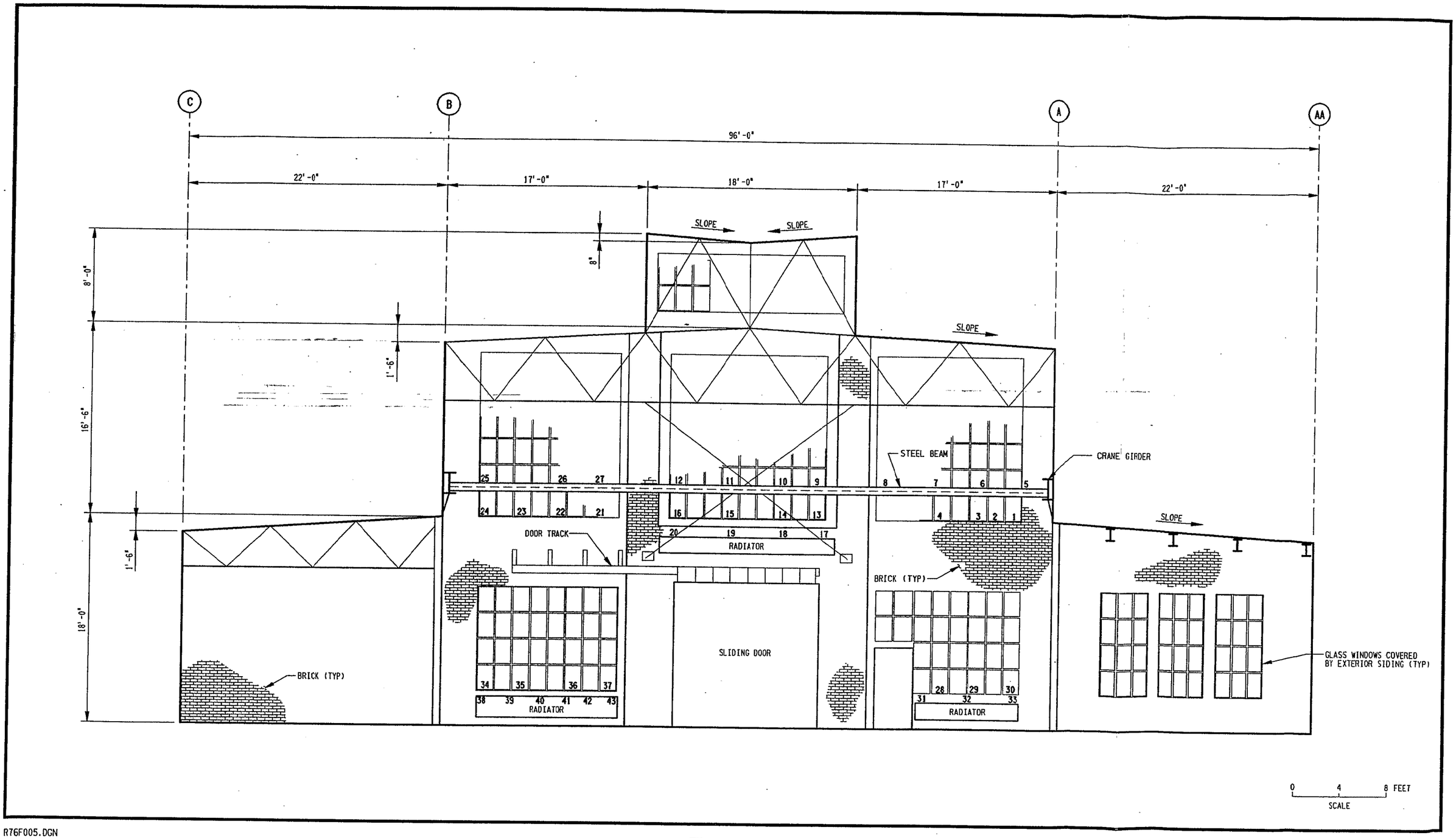


Figure 5
Chapman Valve - 1994 Characterization Results
Interior West Wall



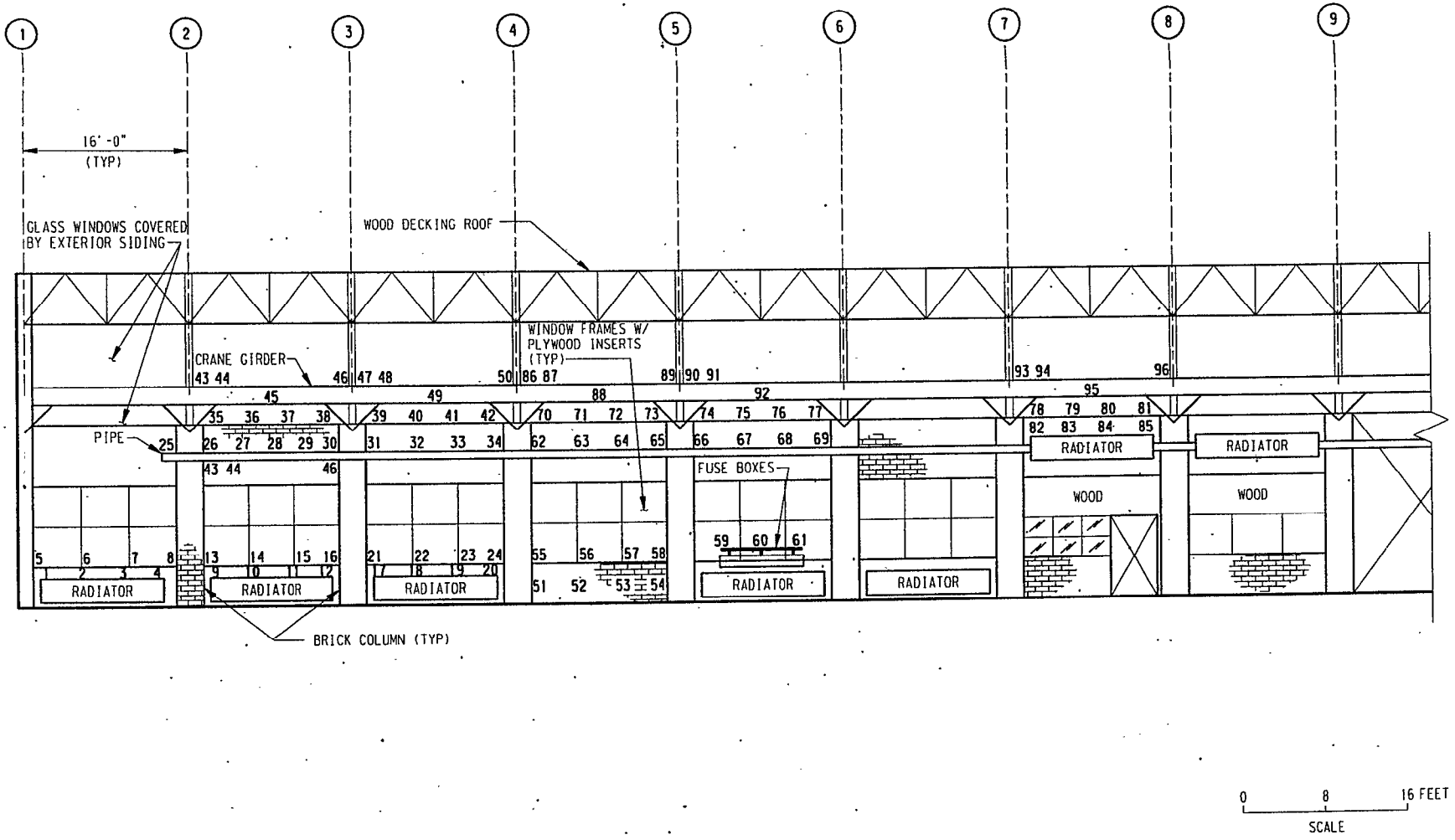
R76F011.DGN

Figure 6
Post-Remedial Action Survey Locations on Overhead Crane



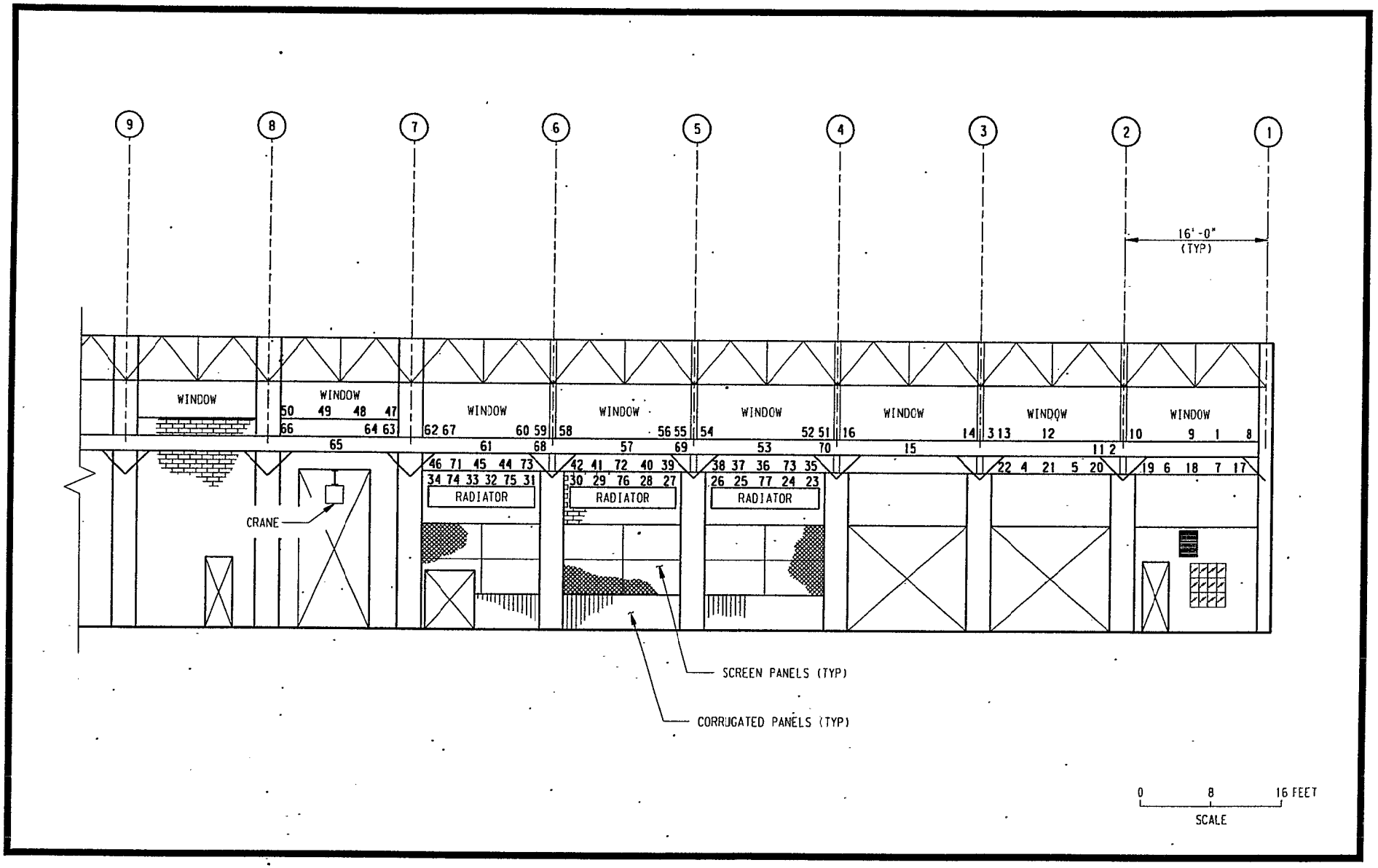
R76F005.DGN

Figure 7
Post-Remedial Action Survey Locations
Interior West Wall



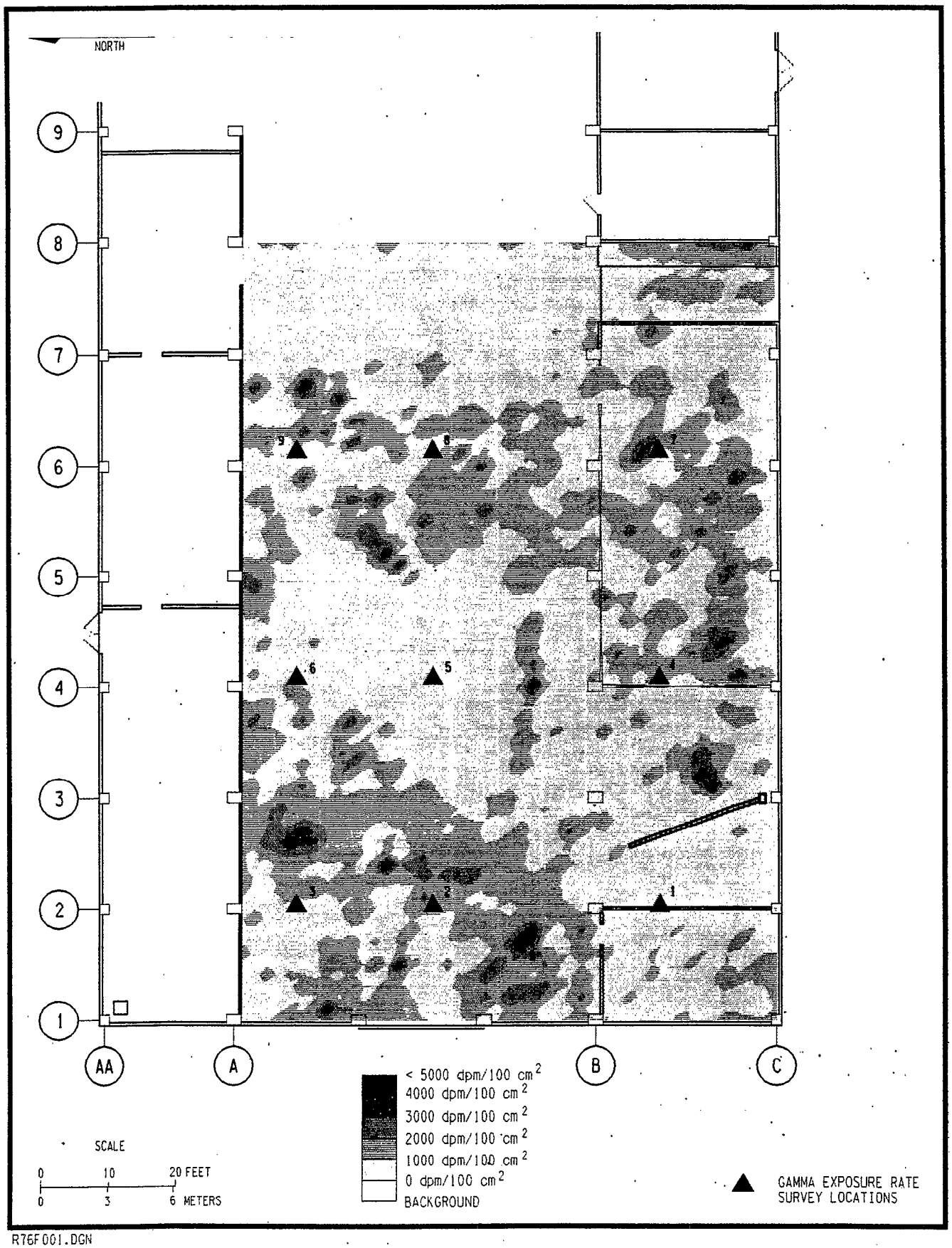
R76F003.DGN

Figure 8
Post-Remedial Action Survey Locations
Interior North Wall



R76F004.DGN

Figure 9
Post-Remedial Action Survey Locations
Interior South Wall



R76F001.DGN

Figure 10
Chapman Valve Post-Remedial Action Floor Survey

TABLES

TABLE 1
SUMMARY OF RESIDUAL CONTAMINATION GUIDELINES

BASIC DOSE LIMITS

The basic limit for the annual radiation dose (including all pathways except radon) received by an individual member of the general public is 100 mrem/yr above background. In implementing this limit, DOE applies as-low-as-reasonably achievable principles to set site-specific guidelines.

SOIL GUIDELINES

Radionuclide	Soil Concentration (pCi/g) Above Background[a,b,c]
Radium-226 Radium-228 Thorium-230 Thorium-232	5 pCi/g when averaged over the first 15 cm of soil below the surface and over any contiguous 100-m^2 surface area; 15 pCi/g when averaged over any 15-cm-thick soil layer below the surface layer and over any contiguous 100-m^2 surface area.
Total Uranium	Site-specific uranium guideline for Chapman Valve was not determined.

STRUCTURE GUIDELINES

Airborne Radon Decay Products

Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property that has no radiological restrictions on its use; structures that will be demolished or buried are excluded. The applicable generic guideline (40 CFR 192) is: In any occupied or habitable building, the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL[d]. In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions are not required in order to comply with this guideline when there is reasonable assurance that residual radioactive materials are not the cause.

External Gamma Radiation

The average level of gamma radiation inside a building or habitable structure on a site that has no radiological restrictions on its use shall not exceed the background level by more than 20 μR/h and will comply with the basic dose limits when an appropriate-use scenario is considered.

Indoor/Outdoor Structure Surface Contamination

Radionuclide[f]	Allowable Surface Residual Contamination[e] (dpm/100 cm^2) Average[g,h]	Maximum[h,i]	Removable[h,j]
Transuranics, Ra-226, Ra-228, Th-230, Th-228 Pa-231, Ac-227, I-125, I-129[k]	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224 U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000 α	15,000 α	1,000 α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above[l]	5,000 β - γ	15,000 β - γ	1,000 β - γ

TABLE 1
(CONTINUED)

[a]These guidelines take into account ingrowth of radium-226 from thorium-230 and of radium-228 from thorium-232, and assume secular equilibrium. If either thorium-230 and radium-226 or thorium-232 and radium-228 are both present, not in secular equilibrium, the guidelines apply to the higher concentration. If other mixtures of radionuclides occur, the concentrations of individual radionuclides shall be reduced so that (1) the dose for the mixtures will not exceed the basic dose limit, or (2) the sum of ratios of the soil concentration of each radionuclide to the allowable limit for that radionuclide will not exceed 1 ("unity").

[b]These guidelines represent allowable residual concentrations above background averaged across any 15-cm-thick layer to any depth and over any contiguous 100-m^2 surface area.

[c]If the average concentration in any surface or below-surface area less than or equal to 25 m^2 exceeds the authorized limit or guideline by a factor of $(100/A)^{1/2}$, where A is the area of the elevated region in square meters, limits for "hot spots" shall also be applicable. Procedures for calculating these hot spot limits, which depend on the extent of the elevated local concentrations, are given in the DOE Manual for Implementing Residual Radioactive Materials Guidelines, DOE/CH/8901. In addition, every reasonable effort shall be made to remove any source of radionuclide that exceeds 30 times the appropriate limit for soil, irrespective of the average concentration in the soil.

[d]A working level (WL) is any combination of short-lived radon decay products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

[e]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 measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[f]Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

[g]Measurements of average contamination should not be averaged over an area of more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

[h]The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at a depth of 1 cm.

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

[j]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping an area of that size with dry filter or soft absorbent paper, applying moderate pressure, and measuring the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area, and the entire surface should be wiped. It is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that total residual surface contamination levels are within the limits for removable contamination.

[k]Guidelines for these radionuclides are not given in DOE Order 5400.5; however, these guidelines are considered applicable based on "DOE Guidelines for Residual Radioactive Materials at FUSRAP and Remote SFMP Sites," Revision 2, March 1987.

[l]This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

Source: DOE Order 5400.5 and 40 CFR 192

Table 2
Decontamination Techniques Used at the Chapman Valve Site

Type	Description
HEPA Vacuuming	High-efficiency particulate air- (HEPA-) filtered vacuum cleaners were used to remove loose contamination and dust. HEPA-filtered vacuums were also used in conjunction with other techniques (grinding, wire brushing, etc.) to eliminate the airborne contamination associated with these techniques.
Jackhammering	Conventional jackhammers were used on small areas to remove anchor bolts from the concrete slab. Skid steer loaders equipped with hoe-ram attachments were used to remove the wooden blocks from the floor and to break up the concrete pads to expose the base slab.
Excavation	Contaminated concrete and debris were removed from the building with a skid steer loader. Removal of contaminated soil from the west ramp and excavation of the pipe were performed with shovels.
Wire Brushing	Small areas on the overhead trusses requiring rework were wire brushed to remove contamination.
Scraping	Putty knives were used to scrape contamination from the surface of the 10-ton crane and from wooden planks removed from the crane deck.

Table 3

Cost of Remedial Action at the Chapman Valve Site

Description	Amount
Direct costs	$269,000
Radiological laboratory/HP support	184,000
Chemical laboratory	11,000
Direct hire labor	273,500
Transportation	12,000
Disposal	24,000
Final engineering reports	51,000
Home office support	143,500
Total RA costs	$968,000
Hazard assessment supplemental limits cost savings	$2,000,000

Table 4

Post-Remedial Action Survey Data - Crane[a]

Location Number	Direct (dpm/100 cm2)		Removable (dpm/100 cm^2)[b]		Location Number	Direct (dpm/100 cm^2)		Removable (dpm/100 cm^2)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	16	201	2	84	33	8	-177	-[b]-	-[b]-
2	1	513	2	110	34	15	25	-[b]-	-[b]-
3	-6	313	-1	-19	35	15	-354	-[b]-	-[b]-
4	9	223	2	20	36	15	-76	-[b]-	-[b]-
5	1	335	-1	-53	37	1	354	-[b]-	-[b]-
6	-6	112	-1	-45	38	-5	-101	-[b]-	-[b]-
7	-6	201	-1	15	39	8	-380	-[b]-	-[b]-
8	-6	402	-1	41	40	21	708	-[b]-	-[b]-
9	-6	380	-1	110	41	26	101	-[b]-	-[b]-
10	16	491	2	3	42	12	557	-[b]-	-[b]-
11	9	313	5	80	43	19	-127	-[b]-	-[b]-
12	1	134	-1	50	44	-3	177	-[b]-	-[b]-
13	9	357	2	114	45	4	633	-[b]-	-[b]-
14	9	201	2	11	46	4	481	-[b]-	-[b]-
15	1	290	-1	-15	47	134	1493	8	73
16	37	0	2	33	48	220	2378	5	87
17	9	112	2	20	49	163	2606	-2	77
18	-6	290	-1	24	50	55	708	-[b]-	-[b]-
19	15	1670	-2	7	51	4	936	-[b]-	-[b]-
20	-5	-329	-[b]-	-[b]-	52	12	354	-[b]-	-[b]-
21	48	1771	4	15	53	19	734	-[b]-	-[b]-
22	15	177	-[b]-	-[b]-	54	19	911	-[b]-	-[b]-
23	21	607	-[b]-	-[b]-	55	19	405	-[b]-	-[b]-
24	1	633	-[b]-	-[b]-	56	-3	607	-[b]-	-[b]-
25	21	253	-[b]-	-[b]-	57	12	531	-[b]-	-[b]-
26	55	3416	1	45	58	148	1695	8	77
27	15	329	-[b]-	-[b]-	59	91	2075	12	39
28	8	2049	1	-53	60	26	253	-[b]-	-[b]-
29	21	-278	-[b]-	-[b]-	61	19	0	-[b]-	-[b]-
30	35	1619	10	63	62	62	354	-[b]-	-[b]-
31	28	1569	4	11	63	12	177	-[b]-	-[b]-
32	48	531	-[b]-	-[b]-	64	-3	633	-[b]-	-[b]-

Table 4 - Continued

Post-Remedial Action Survey Data - Crane[a]

Location Number	Direct (dpm/100 cm^2)		Removable (dpm/100 cm^2)[b]		Location Number	Direct (dpm/100 cm^2)		Removable (dpm/100 cm^2)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
65	4	860	-[b]-	-[b]-	96	4	531	-[b]-	-[b]-
66	19	734	-[b]-	-[b]-	97	-3	405	-[b]-	-[b]-
67	26	961	-[b]-	-[b]-	98	12	304	-[b]-	-[b]-
68	270	3238	5	83	99	112	2758	8	80
69	83	1189	5	90	100	4	455	-[b]-	-[b]-
70	-3	886	-[b]-	-[b]-	101	19	354	-[b]-	-[b]-
71	-3	759	-[b]-	-[b]-	102	40	810	-[b]-	-[b]-
72	148	3669	-2	60	103	26	810	-[b]-	-[b]-
73	40	3087	5	46	104	19	683	-[b]-	-[b]-
74	12	202	-[b]-	-[b]-	105	55	860	-[b]-	-[b]-
75	62	936	-[b]-	-[b]-	106	-43	-578	-[b]-	-[b]-
76	19	101	-[b]-	-[b]-	107	-43	-509	-[b]-	-[b]-
77	4	582	-[b]-	-[b]-	108	-33	-254	-[b]-	-[b]-
78	12	127	-[b]-	-[b]-	109	-43	-462	-[b]-	-[b]-
79	-3	329	-[b]-	-[b]-	110	-43	-185	-[b]-	-[b]-
80	199	3719	8	87	111	-43	-162	-[b]-	-[b]-
81	98	1645	12	107	112	-24	-624	-[b]-	-[b]-
82	40	1113	2	90	113	-43	-509	-[b]-	-[b]-
83	26	961	-[b]-	-[b]-	114	-43	-971	-[b]-	-[b]-
84	12	810	-[b]-	-[b]-	115	-33	-948	-[b]-	-[b]-
85	91	1898	-2	63	116	-24	-879	-[b]-	-[b]-
86	62	3137	5	107	117	-24	-462	-[b]-	-[b]-
87	-3	633	-[b]-	-[b]-	118	-43	-486	-[b]-	-[b]-
88	19	481	-[b]-	-[b]-	119	-33	-832	-[b]-	-[b]-
89	4	708	-[b]-	-[b]-	120	22	-254	-[b]-	-[b]-
90	33	25	-[b]-	-[b]-	121	-43	-624	-[b]-	-[b]-
91	12	329	-[b]-	-[b]-	122	-43	-555	-[b]-	-[b]-
92	12	506	-[b]-	-[b]-	123	-24	-786	-[b]-	-[b]-
93	4	177	-[b]-	-[b]-	124	4	-809	-[b]-	-[b]-
94	12	582	-[b]-	-[b]-	125	13	-439	-[b]-	-[b]-
95	12	734	-[b]-	-[b]-	126	-43	-254	-[b]-	-[b]-

Table 4 - Continued

Post-Remedial Action Survey Data - Crane[a]

Location Number	Direct (dpm/100 cm^2) Alpha	Direct (dpm/100 cm^2) Beta/Gamma	Removable (dpm/100 cm^2)[b] Alpha	Removable (dpm/100 cm^2)[b] Beta/Gamma
127	-24	-786	-[b]-	-[b]-
128	-33	-647	-[b]-	-[b]-
129	689	-324	-[b]-	-[b]-
130	198	-69	-[b]-	-[b]-
131	143	23	-[b]-	-[b]-
132	319	-92	-[b]-	-[b]-
133	87	-370	-[b]-	-[b]-
134	235	3237	-1	-3
135	161	671	-[b]-	-[b]-
136	13	1156	-1	-10
137	4	324	-[b]-	-[b]-
138	-43	69	-[b]-	-[b]-
139	-33	-254	-[b]-	-[b]-
140	-43	-925	-[b]-	-[b]-
141	-43	-416	-[b]-	-[b]-
142	13	-879	-[b]-	-[b]-
143	172	651	-[b]-	-[b]-
144	24	311	-[b]-	-[b]-
145	43	481	-[b]-	-[b]-
146	6	1471	12	17
147	89	396	-[b]-	-[b]-
148	302	283	-[b]-	-[b]-
149	459	736	-[b]-	-[b]-
150	552	5348	9	-17
151	274	4641	1	58
152	98	368	-[b]-	-[b]-
153	43	-57	-[b]-	-[b]-
154	15	85	-[b]-	-[b]-
155	-4	-170	-[b]-	-[b]-
156	24	538	-[b]-	-[b]-
157	24	-311	-[b]-	-[b]-
158	24	-226	-[b]-	-[b]-
159	70	1585	1	31
160	256	538	-[b]-	-[b]-
161	89	198	-[b]-	-[b]-
162	61	-113	-[b]-	-[b]-
163	98	311	-[b]-	-[b]-
164	172	1330	12	20
165	80	-170	-[b]-	-[b]-
166	43	-85	-[b]-	-[b]-
167	33	283	-[b]-	-[b]-
168	70	-198	-[b]-	-[b]-
169	61	-198	-[b]-	-[b]-
AVERAGE	44	520	3	42
DOE Guideline	5000	5000	1000	1000

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Transferable surface readings were not taken because direct readings were less than the transferable criterion of 1000 dpm/100 cm^2 for that location.

Table 5

Post-Remedial Action Survey Data - Trusses and Overheads[a]

Truss Number	Number of Locations		Horizontal Surfaces (dpm/100 cm²) Direct Alpha	Horizontal Surfaces Direct Beta/Gamma	Horizontal Surfaces Removable Alpha	Horizontal Surfaces Removable Beta/Gamma	Non-Horizontal Surfaces (dpm/100 cm²) Direct Alpha	Non-Horizontal Surfaces Direct Beta/Gamma	Non-Horizontal Surfaces Removable Alpha	Non-Horizontal Surfaces Removable Beta/Gamma	Light Fixtures (dpm/100 cm²) Direct Alpha	Light Fixtures Direct Beta/Gamma	Light Fixtures Removable Alpha	Light Fixtures Removable Beta/Gamma
1	160	Average	452	5657	24	112	702	3173	27	128	859	3758	65	227
		Maximum	2221	27841	101	302	1776	7330	78	244	1471	4810	117	339
		Minimum	4	139	-2	0	4	-277	-2	41	161	2798	35	132
2	101	Average	2114	12261	18	125	1869	6175	20	107	794	2336	2	62
		Maximum	17577	111967	193	539	17397	18280	88	573	1300	3677	7	79
		Minimum	-9	277	-2	-25	102	948	-2	-17	59	-185	-2	44
3	69	Average	907	4266	29	70	889	5383	17	61	-[b]-	-[b]-	-[b]-	-[b]-
		Maximum	9973	28806	368	634	5916	27419	55	129	-[b]-	-[b]-	-[b]-	-[b]-
		Minimum	26	113	-2	-20	7	-85	-2	0	-[b]-	-[b]-	-[b]-	-[b]-
4	68	Average	361	1773	11	55	483	2221	21	48	-[b]-	-[b]-	-[b]-	-[b]-
		Maximum	1024	17827	42	180	1450	10017	192	502	-[b]-	-[b]-	-[b]-	-[b]-
		Minimum	26	-792	-2	14	7	-481	0	-19	-[b]-	-[b]-	-[b]-	-[b]-
5	150	Average	422	1783	7	27	316	1245	3	19	432	1186	18	17
		Maximum	4724	14591	58	98	741	4185	9	62	808	4324	35	49
		Minimum	17	-347	-2	-36	-7	-162	-1	-19	4	-254	-1	-39
6	67	Average	243	1395	9	27	114	815	1	10	-[b]-	-[b]-	-[b]-	-[b]-
		Maximum	1400	17940	131	310	715	8319	8	76	-[b]-	-[b]-	-[b]-	-[b]-
		Minimum	-8	-453	-1	-16	-7	-283	-1	-29	-[b]-	-[b]-	-[b]-	-[b]-
7	71	Average	329	1923	3	36	201	865	2	18	-[b]-	-[b]-	-[b]-	-[b]-
		Maximum	1773	12507	15	173	902	5263	9	156	-[b]-	-[b]-	-[b]-	-[b]-
		Minimum	13	-509	-1	-31	-6	-509	-1	-14	-[b]-	-[b]-	-[b]-	-[b]-
8	43	Average	344	2567	14	32	287	1592	10	-10	-[b]-	-[b]-	-[b]-	-[b]-
		Maximum	1389	20141	42	123	556	5388	28	42	-[b]-	-[b]-	-[b]-	-[b]-
		Minimum	0	-439	0	-33	93	-90	-1	-114	-[b]-	-[b]-	-[b]-	-[b]-
1 to 8[c]	40	Average	161	627	2	8	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-
		Maximum	1895	11084	25	256	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-
		Minimum	-22	-315	0	-62	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-	-[c]-
Supplemental Limit[d]			15000	15000	1000	1000	15000	15000	1000	1000	15000	15000	1000	1000

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted the numerical value was negative.

[b] These truss areas did not contain light fixtures.

[c] This survey was performed on the underside of horizontal surfaces and in the area between back-to-back welded angles on all eight trusses. Non-horizontal surfaces and light fixtures were not surveyed.

[d] The supplemental limit is an average for the truss; there is no maximum limit.

Table 6

Post-Remedial Action Survey Data - West Wall[a]

Location Number	Direct (dpm/100 cm²)		Removable (dpm/100 cm²)[b]		Location Number	Direct (dpm/100 cm²)		Removable (dpm/100 cm²)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	100	4075	-1	18	22	989	8263	19	187
2	82	3792	-1	24	23	878	80985	9	119
3	193	8489	-1	4	24	322	5490	-1	-6
4	137	14375	5	45	25	626	23147	19	52
5	1304	37295	12	119	26	774	11149	2	65
6	878	61743	42	401	27	4470	14544	108	170
7	2508	15903	9	123	28	6	2999	2	-30
8	693	22977	5	106	29	6	6112	12	18
9	600	59479	75	221	30	70	4980	2	18
10	2230	130787	164	919	31	43	1471	-1	18
11	4824	60045	9	113	32	61	170	-[b]-	-[b]-
12	3731	5433	138	594	33	6	1075	9	18
13	44	5999	5	38	34	154	30730	2	-37
14	248	4301	-1	28	35	302	12620	2	-16
15	359	34692	2	89	36	395	26033	15	4
16	433	10922	2	14	37	376	10413	9	31
17	174	14997	9	31	38	24	4075	9	4
18	63	16921	5	92	39	43	3679	5	-9
19	137	22354	5	102	40	43	1585	2	-64
20	433	6508	5	18	41	61	6452	-1	-57
21	878	15167	2	85	42	24	2999	2	-30
					Average	708	19410	17	70
					Doe Guideline	-[c]-	-[c]-	-[c]-	-[c]-

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Removable surface readings were not taken because direct readings were less than the removable criteria of 1000 dpm/100 cm² for that location.

[c] The walls were decontaminated to comply with DOE's ALARA policy, and the data collected are intended to be used in the final hazard assessment calculation; therefore, there are no specific surface criteria that apply to the walls.

Table 7

Post-Remedial Action Survey Data - North Wall[a]

Location Number	Direct (dpm/100 cm^2)		Removable (dpmM/100 cm^2)[b]		Location Number	Direct (dpm/100 cm^2)		Removable (dpm/100 cm^2)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	20	5018	5	-16	34	419	1711	19	-16
2	48	2428	15	-3	35	270	2844	22	18
3	104	2914	22	-3	36	622	2960	12	18
4	187	2336	9	11	37	270	3908	29	68
5	2	3839	-1	-13	38	761	2567	22	24
6	39	2243	2	-9	39	270	3307	2	-37
7	2	2752	2	-16	40	94	3422	2	-23
8	57	2613	12	-16	41	187	4902	12	62
9	2	2729	-1	-37	42	215	4394	32	18
10	113	3839	22	-23	43	57	1156	9	126
11	39	2312	9	-57	44	743	3006	5	143
12	30	2567	5	11	45	919	8949	12	11
13	104	2798	15	-43	46	1762	36698	850	3197
14	94	1896	2	45	47	39	301	-[b]-	-[b]-
15	113	2613	12	-30	48	947	25714	15	45
16	57	2891	5	-9	49	1104	3839	65	265
17	11	3908	2	4	50	965	6914	-1	-43
18	57	2359	15	-6	51	11	1781	5	46
19	11	2474	2	-9	52	11	1017	-2	-28
20	178	1734	-1	4	53	30	185	-[b]-	-[b]-
21	94	2914	5	11	54	2	1295	1	39
22	85	2497	2	-23	55	215	1457	11	19
23	2	2289	-1	31	56	30	925	-[b]-	-[b]-
24	48	3006	12	-64	57	11	1226	1	-15
25	233	12071	32	-40	58	11	647	-[b]-	-[b]-
26	919	10082	15	4	59	76	139	-[b]-	-[b]-
27	243	3422	25	48	60	11	439	-[b]-	-[b]-
28	122	856	9	18	61	48	69	-[b]-	-[b]-
29	39	5272	5	-9	62	150	1017	11	-8
30	326	5365	35	72	63	530	1503	28	-28
31	345	1688	5	-43	64	307	717	-[b]-	-[b]-
32	317	1526	2	-9	65	30	301	-[b]-	-[b]-

Table 7 - continued

Post-Remedial Action Survey Data - North Wall[a]

Location Number	Direct (dpm/100 cm^2)		Removable (dpm/100 cm^2)[b]		Location Number	Direct (dpm/100 cm^2)		Removable (dpm/100 cm^2)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
33	919	1503	22	79	66	104	856	-[b]-	-[b]-
67	363	740	-[b]-	-[b]-	82	169	1064	18	-35
68	132	763	-[b]-	-[b]-	83	39	509	-[b]-	-[b]-
69	539	1434	-2	33	84	548	6544	31	185
70	11	2127	5	-5	85	132	1572	21	29
71	178	1064	8	22	86	48	23	-[b]-	-[b]-
72	280	1202	18	39	87	1021	14268	31	155
73	243	2428	21	9	88	48	3645	15	39
74	94	555	-[b]-	-[b]-	89	576	3076	15	-62
75	30	1804	11	-59	90	20	-416	-[b]-	-[b]-
76	169	532	-[b]-	-[b]-	91	1437	27749	25	243
77	85	2405	18	-28	92	2595	4648	213	266
78	94	-185	-[b]-	-[b]-	93	345	-46	-[b]-	-[b]-
79	910	2012	38	-45	94	372	2359	34	100
80	11	1526	-2	39	95	446	7307	101	209
81	335	1896	11	46	96	48	4116	5	-32
					Average	1001	2823	35	58
					Doe Guideline	-[c]-	-[c]-	-[c]-	-[c]-

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Removable surface readings were not taken because direct readings were less than the removable criterion of 1000 dpm/100 cm^2 for that location.

[c] The walls were decontaminated to comply with DOE's ALARA policy, and the data collected are intended to be used in the final hazard assessment calculation; therefore, there are no specific surface criteria that apply to the walls.

Table 8

Post-Remedial Action Survey Data - South Wall[a]

Location Number	Direct (dpm/100 cm²)		Removable (dpm/100 cm²)[b]		Location Number	Direct (dpm/100 cm²)		Removable (dpm/100 cm²)[b]	
	Alpha	Beta/Gamma	Alpha	Beta/Gamma		Alpha	Beta/Gamma	Alpha	Beta/Gamma
1	2564	41709	22	265	34	354	740	-[b]-	-[b]-
2	7	283	-[b]-	-[b]-	35	261	1272	21	33
3	2230	7074	5	102	36	761	8140	34	134
4	44	12677	-1	11	37	67	1665	11	-39
5	304	7866	5	52	38	428	4255	18	5
6	82	3056	-1	1	39	196	5850	15	94
7	545	2264	-1	-23	40	483	7538	28	-5
8	3710	17714	5	55	41	122	971	11	-25
9	191	849	39	150	42	539	2451	8	-56
10	3358	9508	111	319	43	215	694	-[b]-	-[b]-
11	691	26005	9	31	44	85	971	-[b]-	-[b]-
12	209	3735	5	24	45	122	2336	1	-25
13	1450	9508	22	153	46	641	2497	28	-1
14	135	2490	15	52	47	122	6428	8	-8
15	432	2490	72	252	48	132	1873	1	-35
16	80	1471	9	-3	49	30	1734	5	-1
17	7804	20543	88	241	50	39	3214	8	26
18	784	14488	12	72	51	67	-162	-[b]-	-[b]-
19	580	44822	19	126	52	446	4509	38	56
20	2228	67572	32	153	53	558	925	-[b]-	-[b]-
21	5414	11375	25	99	54	1947	5758	68	161
22	691	13639	22	24	55	-7	-324	-[b]-	-[b]-
23	11	2867	1	-59	56	354	1665	5	-76
24	196	4671	1	-62	57	169	902	-[b]-	-[b]-
25	67	324	-[b]-	-[b]-	58	409	5943	44	175
26	187	2428	11	-15	59	48	-370	-[b]-	-[b]-
27	39	2960	-2	-45	60	261	2127	5	49
28	67	2035	5	-12	61	372	2197	21	16
29	1067	1202	8	26	62	187	2359	18	138
30	104	2821	1	16	63	30	763	-[b]-	-[b]-
31	178	5319	15	-52	64	2	-624	-[b]-	-[b]-
32	-7	7215	-2	2	65	169	694	-[b]-	-[b]-

Table 8 - Continued

Post-Remedial Action Survey Data - South Wall[a]

Location Number	Direct (dpm/100 cm^2) Alpha	Direct (dpm/100 cm^2) Beta/Gamma	Removable (dpm/100 cm^2)[b] Alpha	Removable (dpm/100 cm^2)[b] Beta/Gamma	Location Number	Direct (dpm/100 cm^2) Alpha	Direct (dpm/100 cm^2) Beta/Gamma	Removable (dpm/100 cm^2)[b] Alpha	Removable (dpm/100 cm^2)[b] Beta/Gamma
33	567	4278	11	121	66	104	578	-[b]-	-[b]-
67	283	1188	32	106	73	450	22864	2	11
68	6	4358	2	-50	74	43	3679	2	-23
69	1469	3452	12	45	75	98	2320	15	-6
70	580	1075	9	-30	76	43	2037	5	-9
71	265	1981	9	85	77	6	1358	-1	-37
72	450	10696	12	75					
					Average	642	6387	17	45
					Doe Guideline	-[c]-	-[c]-	-[c]-	-[c]-

[a] Negative numbers indicate that the measurement was less than the minimum detectible activity and that after background was subtracted, the numerical value was negative.

[b] Transferable surface readings were not taken because direct readings were less than the transferable criterion of 1000 dpm/100 cm^2 for that location.

[c] The walls were decontaminated to comply with DOE's ALARA policy, and the data collected are intended to be used in the final hazard assessment calculation; therefore, there are no specific surface criteria that apply to the walls.

Table 9

Post-Remedial Action and Bench - Scale Sampling Results[a]

Location	TCLP Pb ug/L	TCLP Cd ug/L	U-238 pCi/g	Ra-226 pCi/g	Th-232 pCi/g
Dust from crane (mixed with concrete)	338	< 3.0	-[b]-	-[b]-	-[b]-
Bench Scale Test (Ratio 2)	315	< 3.0	-[b]-	-[b]-	-[b]-
Bench Scale Test (Ratio 4)	102	< 3.0	-[b]-	-[b]-	-[b]-
Bench Scale Test (Ratio 6)	302	< 3.0	-[b]-	-[b]-	-[b]-
Bench Scale Test (Ratio 3)	94.2	< 3.0	-[b]-	-[b]-	-[b]-
Bench Scale Test (Ratio 5)	< 46.6	< 3.0	-[b]-	-[b]-	-[b]-
Bench Scale Test (Ratio 1)	171	< 3.0	-[b]-	-[b]-	-[b]-
West Equipment Door Ramp	-[b]-	-[b]-	< 2.00	0.47	0.41
Room B4	-[b]-	-[b]-	0.26	0.47	0.50
Room B4	-[b]-	-[b]-	< 0.62	0.38	0.39
Room B4	-[b]-	-[b]-	< 0.54	< 0.3	0.42
DOE Soil Guideline	-[c]-	-[c]-	-[d]-	-[e]-	-[e]-
RCRA TCLP Limits	5000	1000	-[c]-	-[c]-	-[c]-

[a] Less than values (<) are results less than the minimum detectable activity, and the number reported is less than the minimum detectable activity.

[b] Sample was not analyzed for this analyte.

[c] This set of guidelines does not apply to this analyte.

[d] There was no site-specific uranium guideline developed for CHV. A typical U-238 guideline for FUSRAP sites ranges from 35 to 50 pCi/g.

[e] DOE soil cleanup guideline for radium and thorium is 5 pCi/g in the top 6 inches of soil and 15 pCi/g greater than 6 inches below the surface of the soil.

Table 10
Hazard Assessment Summary

Dose Scenario	Calculated Number	Units
Building Demolition		
Roof Demolition	4.9	mrem
Wall Demolition	0.02	mrem
Structural Steel Demolition	0.76	mrem
Removal of Floor Blocks	0.0007	mrem
Total	5.6	mrem
Structural Steel Recycling		
Torch Cutting of Trusses	5.3	mrem
Sand Blasting of Trusses	10.7	mrem
Slag From Melting Trusses	0.29	mrem
Total	16.3	mrem
Reuse of Building Materials	17.3	mrem/yr
Activity of Building Rubble	4.4	pCi/g
Mass of Uranium in Rubble	20	lbs

TABLE 11
REMEDIAL ACTION SUMMARY

WBS 133
SITE Chapman Valve

OWNER Crane Company
SITE ADDRESS 203 Hampshire Street
CITY, STATE Indian orchard, MA 05515

REMEDIATION AUTHORITY
- ☒ NEPA/CERCLA
- ☐ SUPERFUND
- ☐ RCRA

ACTION	DATE	RESPONSIBLE ENTITY	DOCUMENT
DESIGNATION	December 15, 1992	DOE	Authorization for RA
CHARACTERIZATION	July 1992	ORNL	Results of Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts
CHARACTERIZATION	May 4, 1995	BNI	FUSRAP Technical memorandum - Chapman Valve Characterization Results
FINAL RA	September 1996	DOE/ORNL/BNI	FUSRAP Technical memorandum - Chapman Valve Post-Remedial Action Report

TOTAL VOLUME 750 cy
- To Remain In Situ 731 cy
- Volume Reduction 0
- Net Disposal 19 cy

Documentation Used: Hazard Assessment in FUSRAP Technical Memorandum - Chapman Valve Post-Remedial Action Report

TYPE OF WASTE FOR NET DISPOSAL:

REGULATORY	VOLUME	DISPOSAL SITE
☒ LLRW	19 cy	Clive, Utah
☐ 11(E)2		
☐ MIXED		Clive, Utah
☐ CHEMICAL		

PHYSICAL		
☒ BUILDING RUBBLE	6 cy	Clive, Utah
☒ SOIL	3 cy	Clive, Utah
☐ LIQUID		
☒ OTHER Dust Mixed with Concrete	10 cy	Clive, Utah

TREATMENT TECHNOLOGIES APPLIED AT THE SITE:
Chemical Stabilization

134984

OAK RIDGE NATIONAL LABORATORY
MANAGED BY LOCKHEED MARTIN ENERGY SYSTEMS
FOR THE U. S. DEPARTMENT OF ENERGY

POST OFFICE BOX 2008
OAK RIDGE, TENNESSEE 37831

OCT 2 1 18 PM '95

September 25, 1995

Dr. W. Alexander Williams
Designation and Certification Manager
Office of Eastern Area Programs
Cloverleaf Building (EM-421)
Department of Energy
19901 Germantown Road
Germantown, Maryland 20874-1290

Dear Dr. Williams:

Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts

An independent assessment of remedial action related activities at Building 23 of the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts, is underway by the Oak Ridge National Laboratory, Measurement Applications and Development Group. The purpose of the assessment is to verify the data supporting the remedial action and the adequacy of the remedial action and to confirm the site's compliance with remedial action criteria. The assessment will include both a review of the post remedial action report, as well as the already completed verification surveys of the property. Based on the independent surveys performed by the Measurement Applications and Development group in August 1995, and partial review of the post remedial action survey data collected by Eberline, the site conforms to the supplemental criteria that are established for this site. However, final verification of the site is dependent upon the approval of the hazard assessment and supplemental limits by the Department of Energy, and the post demolition verification surveys. A formal report will be drafted at a later date.

If you have any questions please call R. E. Rodriguez (423-574-5750).

Sincerely,

R. E. Rodriguez

R. E. Rodriguez
Measurement Applications
and Development Group

RER:lec

c: J. D. Kopotic (DOE-ORO)

133-LOA-7AA-00001

ornl

ORNL/RASA-95/17

OAK RIDGE
NATIONAL
LABORATORY

LOCKHEED MARTIN

Results of the Independent Radiological Verification Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001V)

R. E. Rodriguez
C. A. Johnson

MANAGED AND OPERATED BY
LOCKHEED MARTIN ENERGY RESEARCH CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL-27 (3-96)

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LIFE SCIENCES DIVISION
Environmental Restoration and Waste Management Non-Defense Programs
(Activity No. EX 20 20 01 0; ADS1310AA)

Results of the Independent Radiological Verification Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001V)

R. E. Rodriguez and C. A. Johnson

Date issued —May 1997

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R. E. Rodriguez—Field Survey Team Leader

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ABSTRACT

The Chapman Valve Manufacturing Company in Indian Orchard, Massachusetts was one of many companies performing work during the 1940s associated with the development of nuclear energy for defense-related projects for the Manhattan Engineer District (MED) and the Atomic Energy Commission (AEC). In 1947 Chapman set aside approximately one-third of Building 23 at the Chapman site for the machining of uranium rods. The "set aside" portion of the building measured ~ 200 by 60 ft and was separated from the remainder of the building by a floor-to-ceiling wooden partition.

In 1991, a radiological survey was conducted at the former Chapman Valve Manufacturing Company by members of the Measurement Applications and Development (MAD) Group of the Oak Ridge National Laboratory (ORNL) at the request of the U. S. Department of Energy (DOE). The survey report, published in 1992, concluded that survey results indicated ^{238}U contamination in excess of the DOE criteria for surface contamination, and some ^{235}U residues in the west end of the building.

Decontamination of the facility to supplemental guidelines, derived by a hazard assessment based on appropriate scenarios for this building, was conducted by subcontractor personnel in 1995 under the direction of Bechtel National Incorporated (BNI), the project management contractor for FUSRAP. The independent radiological verification survey detailed in this report was performed in July and August 1995 under the FUSRAP program by members of ORNL at the request of DOE.

The radiological verification survey of the west end of the building included gamma, alpha, and beta-gamma scans for fixed contamination, smear sampling for transferable contamination, and radionuclide analysis of soil samples taken from outside the building and from excavations in the concrete floor inside the building.

Additionally, DOE has committed to conduct an additional radiological survey after demolition of the building by the propety owner.

Based on the results of the remedial action and verification data in this report, all radiological measurements fall below the limits prescribed by DOE radiological guidelines established for this site.

Results of the Independent Radiological Verification Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001V)*

INTRODUCTION

The Chapman Valve Manufacturing Company in Indian Orchard, Massachusetts, was one of many companies performing work during the 1940s associated with the development of nuclear energy for defense-related projects for the Manhattan Engineer District (MED) and the Atomic Energy Commission (AEC). Operations conducted under government contract at such sites included the procurement, storage, and processing of uranium oxides, salts, and metals, and the subsequent machining of these products. As a result of activities involving these materials, equipment, buildings, and land at some of the sites became radiologically contaminated with small amounts of the material resulting in low levels of contamination on the properties. The Formerly Utilized Sites Remedial Action Program (FUSRAP) was established by DOE in 1974 to assist in the assessment and cleanup activities at these sites.[1]

In 1947 Chapman set aside approximately one-third of Building 23 at the Chapman site, 203 Hampshire Street, Indian Orchard, Massachusetts, for the machining of uranium rods for Brookhaven Laboratory. The "set aside" portion of the building measured ~ 200 by 60 ft and was separated from the remainder of the building by a floor-to-ceiling wooden partition, which has since been removed. Chapman became part of the Crane Company in 1959. Building 23 has been vacant since Crane discontinued all manufacturing at Indian Orchard early in 1987.[2]

In 1991, a radiological survey was conducted at the former Chapman Valve Manufacturing company by members of the Measurement Applications and Development (MAD) Group of the Oak Ridge National Laboratory (ORNL) at the request of the U. S. Department of Energy (DOE). The survey and sampling at this site covered selected areas outdoors and accessible areas inside Building 23. Survey emphasis was on interior floors, walls and overhead beams. The survey report, published in 1992, concluded that survey results indicated ^{238}U contamination in excess of the DOE criteria for surface contamination, and some ^{235}U residues in the west end of the building.[3]

Decontamination of the facility to supplemental guidelines, derived by a hazard assessment based on appropriate scenarios for this building, was conducted by subcontractor personnel in 1995 under the direction of Bechtel National Incorporated

*The survey was performed by members of the Measurement Applications and Development Group of the former Health Sciences Research Division (now the Life Sciences Division) of Oak Ridge National Laboratory under DOE contract DE-AC05-96OR22464.

(BNI), the project management contractor for FUSRAP. Thermo NUtech was the radiological support subcontractor.

The independent radiological verification survey detailed in this report was performed in July and August 1995 under the FUSRAP program by members of ORNL at the request of DOE. The policy to assign an independent verification contractor ensures the effectiveness of remedial actions performed within FUSRAP and confirms the site's compliance with DOE guidelines.

Figure 1 is a drawing of the former Chapman Valve Manufacturing Company property showing the location of Building 23.

VERIFICATION PROCEDURES

OBJECTIVES

The objective of the verification activities was to confirm (1) that available documentation adequately and accurately describes the post-remedial condition of the facility that is to be verified, and (2) that the remedial action reduced contamination levels to within authorized limits. Applicable DOE residual radioactivity guidelines for protection of the general public are summarized in Table 1.

SURVEY METHODS

Survey methods followed guidelines for a generic site as outlined in References 4 and 5. Figures 2-10 are diagrams of the building indicating locations of scans, smears, and soil samples. The building has been divided into a series of survey blocks, with the east-west support columns and overhead beams numbered 1 to 31, and the north-south columns labeled AA, A, B, and C (Fig. 2). Instrument calibrations were verified and background checked before each survey session.

Residual uranium was concentrated in the west end of the building, specifically blocks A1 through A7. The radiological verification survey of the west end of the building included: (1) a gamma scan at the surface of floors, using sodium iodide (NaI) gamma scintillation detectors; (2) beta-gamma scans of the floors and overhead structures with "pancake" GM detectors, and limited alpha scans on overhead structures with ZnS scintillation detectors; (3) a comprehensive scan of the floor surface for alpha-beta-gamma activity, using large area gas flow proportional detectors; (4) smear sampling at selected locations; and (5) radionuclide analysis of soil samples taken from outside the building and from core holes drilled through the concrete floor inside the building.

VERIFICATION SURVEY AND ANALYSIS

Current guidelines for sites included within FUSRAP are summarized in Table 1. Typical background radiation levels for the Massachusetts area are presented in Table 2. These data are provided for comparison with the survey results presented in this section. Gamma measurements presented in this report are gross readings; background radiation levels have not been subtracted. Similarly, background concentrations have not been subtracted from radionuclide concentrations in soil.

Beta-gamma contamination levels were recorded in gross counts per minute (cpm), background adjusted and converted to disintegrations per minute (dpm/100 cm2) using standard geometry factors for beta-gamma pancake probe/Bicron ratemeter combination. Transferable radioactivity levels (smears) are reported as net counts with background subtracted.

DIRECT AND REMOVABLE RADIOACTIVITY LEVELS

Gamma exposure rates, taken at the initial survey, ranged from 5 to 11 μR/h at 1 m above the floor in grid blocks A1-A30 (see Fig. 2), while surface exposure rates ranged from 5 to 13 μR/h. These values are comparable to the typical average background levels for the area (6 to 9 μR/h, Table 2). Gamma levels at accessible floor and wall surfaces were higher on contact with bricks, concrete, and other materials that contain naturally occurring radioactivity.

Direct beta-gamma activity levels were measured on the overhead crane and at 81 locations on seven roof-support trusses (beams) in the west end of the building, where contamination was found in the first survey. The eighth truss was not included in the verification survey; after review of the post remedial survey data for truss no. 8 it was determined unnecessary. Some activity above guidelines was found on the crane during the initial survey by the verification team but was remediated and again surveyed and verified below guidelines. Beta-gamma activity measured on the trusses ranged from 400 to 60,000 dpm/100 cm^2. When averaged over the first 8 trusses, these measurements meet the supplemental guidelines established for this site (i.e., an average surface level of not more than 15,000 dpm/ 100 cm^2 of uranium activity for the first eight trusses).[6] Results are given in Table 3.

Direct alpha activity was measured on the west beams and in spots on the floors underneath the beams. None of the measurements were above guidelines. A comprehensive beta-gamma scan of the floors under these beams showed no beta-gamma measurements above guidelines. The building is scheduled for demolition by the owner.

Smears were collected at selected locations throughout the area surveyed to ensure that remedial efforts left no residual transferable radioactivity above established guidelines. Results from smear analyses indicate no removable beta-gamma activity

above guidelines and the instrument-specific minimum detectable activity (MDA)*. All alpha measurements were below guidelines (Table 1) for both fixed and transferable alpha activity.

SOIL SAMPLES

Systematic surface (0-6 in, or 0-15 cm) soil samples were taken outdoors as well as indoors under the concrete floor where excavations had been dug by the remedial action contractor to remove the drainlines. Sample locations are shown on Fig. 3.

All samples were analyzed to determine the concentrations of ^{238}U, ^{235}U, ^{226}Ra, and ^{232}Th. Results of the radionuclide analysis are shown in Table 4. Concentrations of ^{235}U and ^{238}U ranged from <0.05 to 0.46 pCi/g, and 0.71 to 14 pCi/g, respectively. The site specific guidelines for ^{238}U concentrations of 35 to 40 pCi/g have been applied at other FUSRAP sites (Table 1). Concentrations for ^{232}Th, and ^{226}Ra in surface soil ranged from 0.27 to 0.43 pCi/g and from 0.43 to 0.65 pCi/g, respectively. All these values are comparable to background levels in the area, and well below DOE guidelines.

CONCLUSIONS

Prior to remedial efforts, uranium residuals exceeded current DOE surface contamination guidelines in the west quadrant of Building 23. With the exception of the overhead crane, the rest of the building had been found to meet guidelines as reported in ORNL/RASA-92/1. The building will be demolished at a future date, when further verification of the site will be performed.

All known radioactivity at the site has been addressed during the remedial action, either by cleanup or by evaluation for Supplemental Standards. In addition, DOE has committed to the property owner and to the community to conduct an additional radiological survey after demolition of the building by the property owner. This will provide the property owner and community with additional confidence that the property does not contain residual uranium at levels that would affect health or safety of future site occupants.

Decontamination of the facility was performed by subcontractors under the direction of BNI. Initially, spotty contamination on the crane was identified by the ORNL verification team during the remediation efforts; however, these residuals were removed

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* The MDA for transferable alpha and beta activity is 9 and 140 dpm/100 cm^2, respectively. The critical detection level (L_c) for transferable alpha and beta activity is 3 and 85 dpm/100 cm^2, respectively.
The MDA for direct alpha and beta measurements is 60 dpm/100 cm^2 and 970 dpm/100 cm^2, respectively. The L_c for direct alpha and beta measurements is 20 and 440 dpm/100 cm^2, respectively.

and the areas resurveyed by Thermo NUtech personnel, followed by another radiological verification effort by the ORNL team.

Results of this independent radiological verification survey of Building 23 at the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts confirm that residual uranium contamination in the west end of the building has been remediated to levels meeting the supplemental DOE guidelines for this site. The results of the direct scans and the removable smear analyses showed that all direct and transferable activity on the overhead beams and crane was below applicable guidelines. The results of soil radionuclide analyses indicate that all soil concentration measurements are below the limits prescribed by DOE radiological guidelines.

Based on the results of the remedial action and verification data in this report, all radiological measurements fall below the limits prescribed by DOE radiological guidelines established for this site.

REFERENCES

1. U. S. Department of Energy, *A Background Report for the Formerly Utilized Manhattan Engineer District/Atomic Energy Commission Sites Program*, DOE/EV-0097, September 1980.

2. T. J. Ungerland, Associate General Council, Crane Company, New York, NY, letter (with attachments) to J. J. Fiore, U.S. Department of Energy, Washington, DC, December 1987.

3. R. D. Foley, M. S. Uziel, *Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001)*, ORNL/RASA-92/1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1992.

4. T. E. Myrick, B. A. Berven, W. D. Cottrell, W. A. Goldsmith, and F. F. Haywood, *Procedures Manual for the ORNL Radiological Survey Activities (RASA) Program*, ORNL/TM-8600, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., April 1987.

5. Oak Ridge National Laboratory, Health Sciences Research Division, *Measurement Applications and Development Group Guidelines*, ORNL-6782, Martin Marietta Energy Systems, Inc., January 1995.

6. J. W. Wagoner II, Director, Division of Off-Site Programs, Office of Eastern Area Programs, Office of Environmental Restoration, U. S. DOE, memorandum to L. K. Price, Director, Former Sites Restoration Division, Oak Ridge Field Office, U. S. DOE, July 27, 1995.

ORNL DWG. No. 96-4417

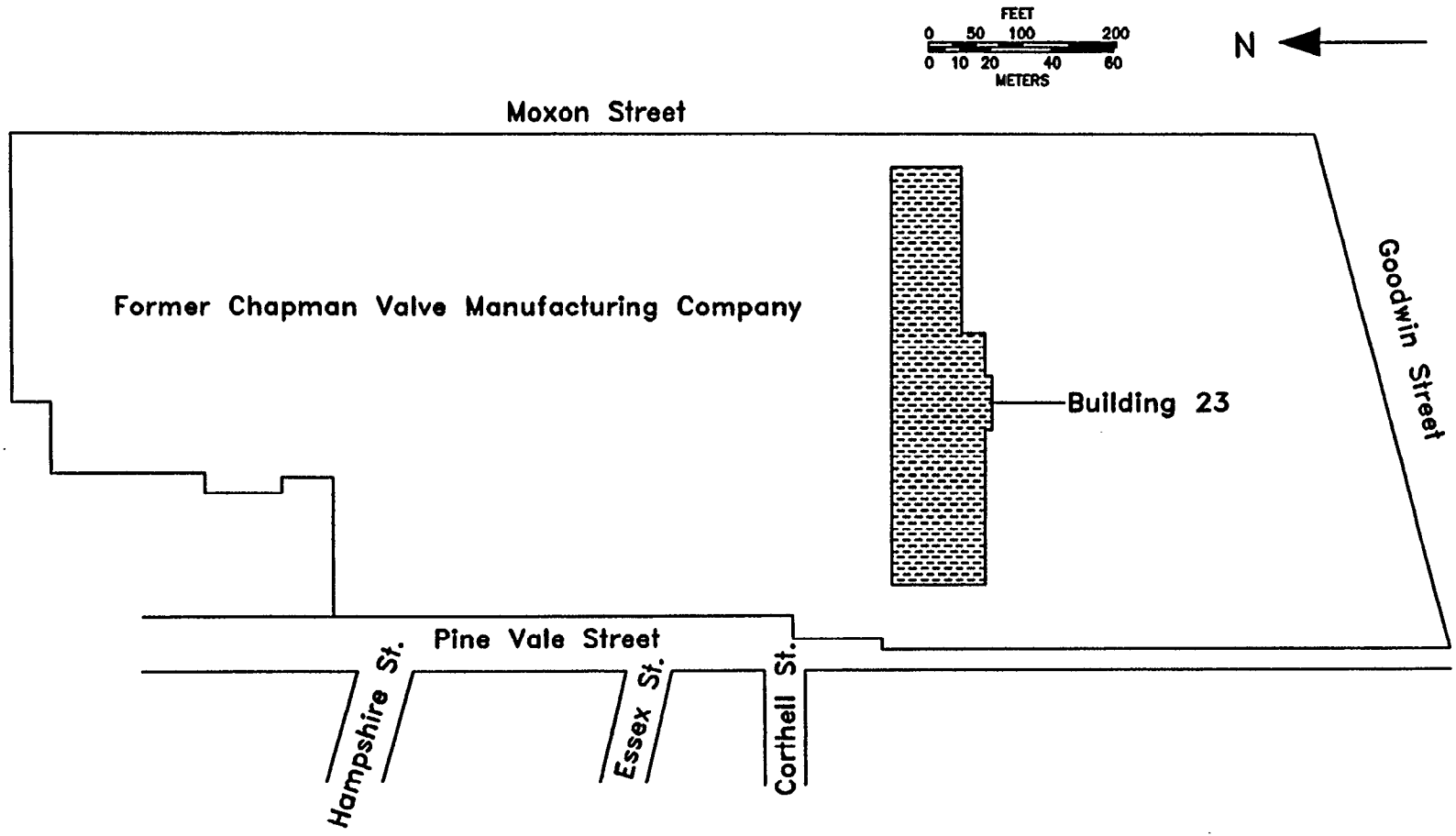

Fig. 1. Location of Building 23 at the site of the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts.

ORNL Dwg.No. 96-4418

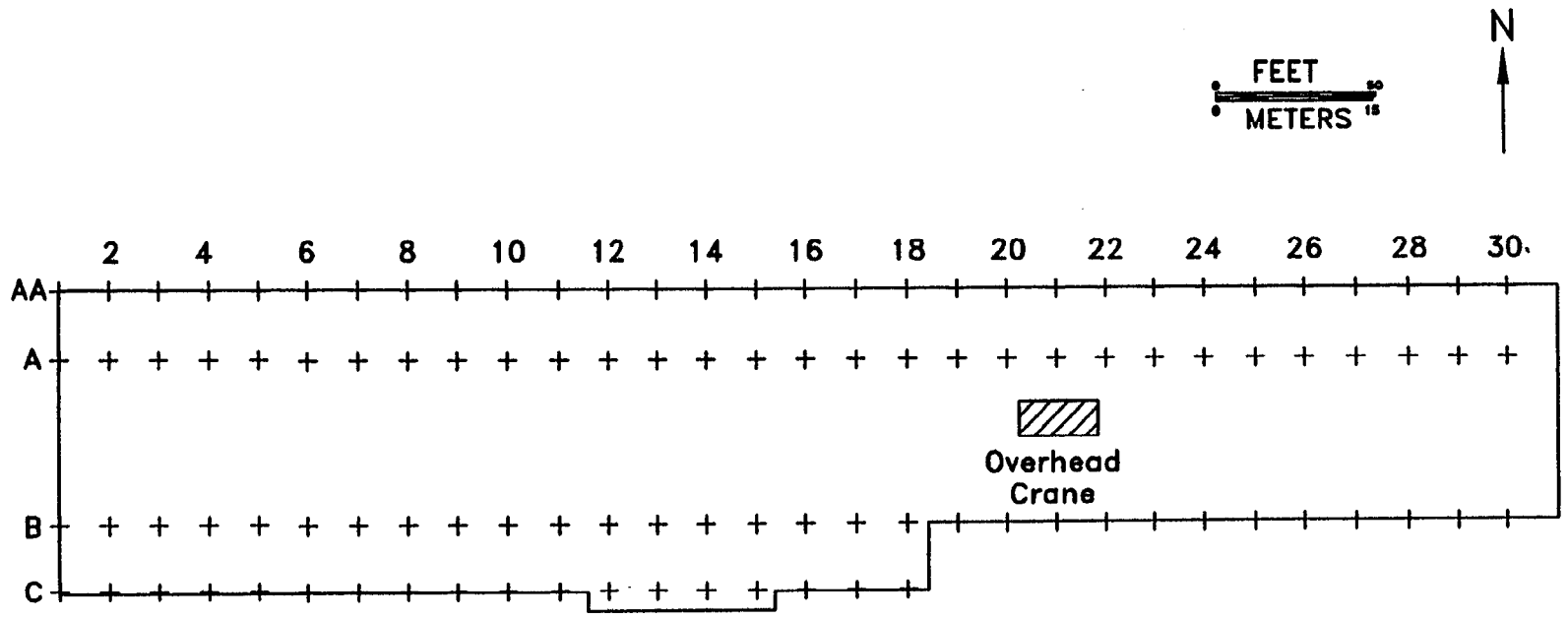

Fig. 2. Building 23 at the site of the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts. Overhead roof support trusses are numbered 1 to 31 and columns are labeled AA to C. The mobile overhead crane is shown in the eastern half of the building.

ORNL Dwg.No. 96-4419

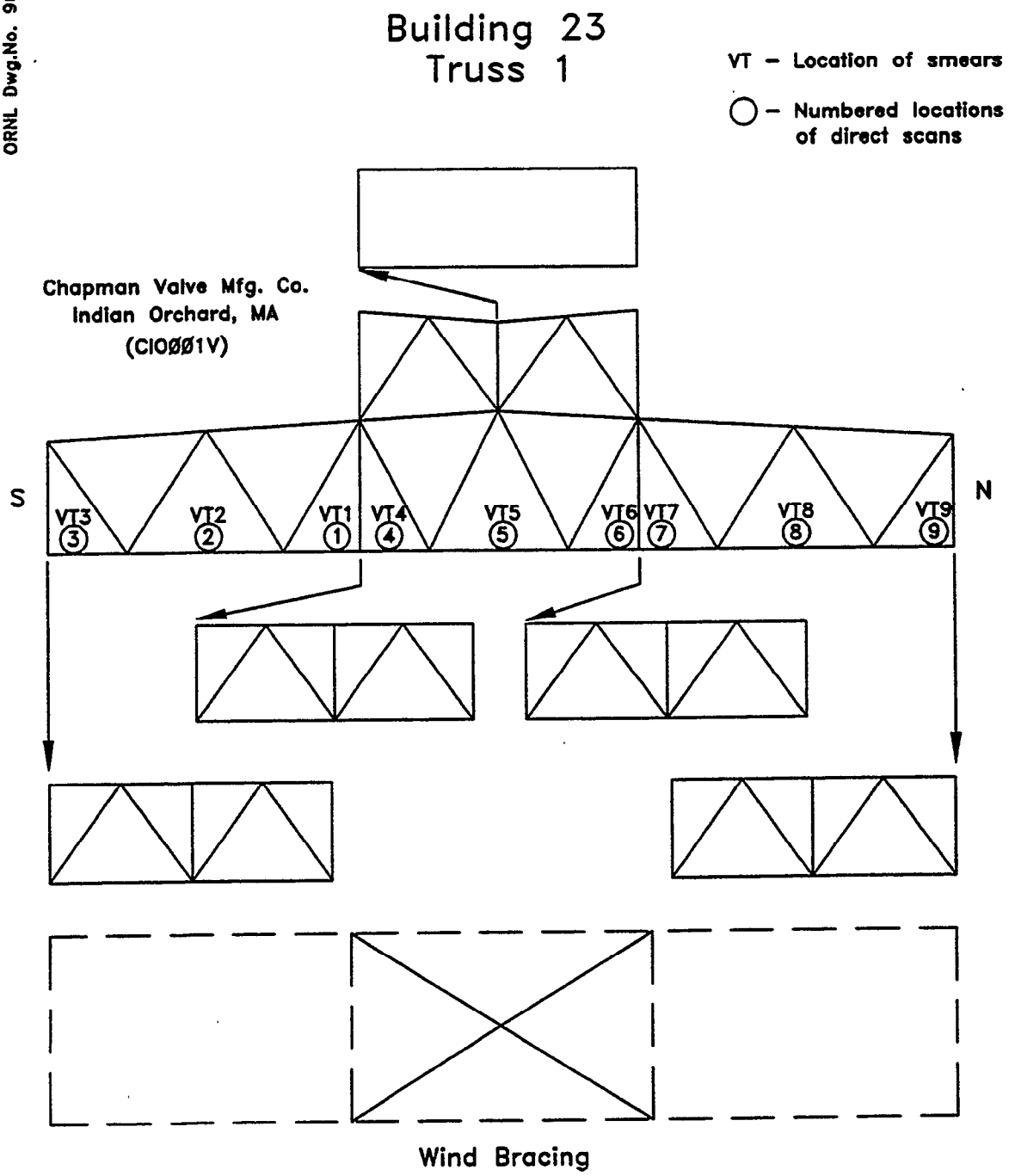

Fig. 3. Locations of directly measured beta–gamma activity levels (dpm/100 cm^2) and smears collected on Truss #1 in Building 23 at the site of the former Chapman Valve facility. Results of alpha and beta–gamma scans on all trusses are shown in Table 3.

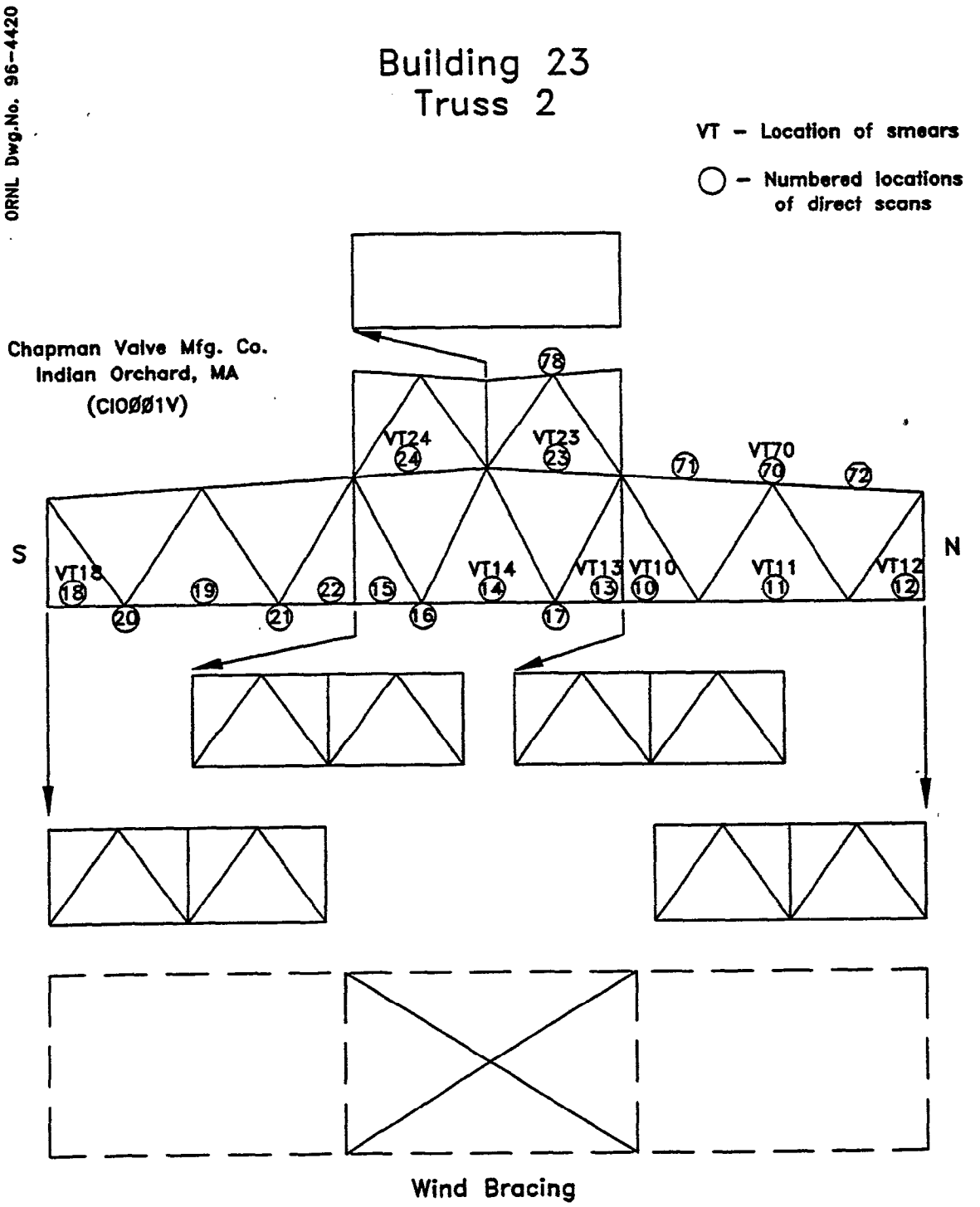

Fig. 4. Locations of directly measured beta–gamma activity levels (dpm/100 cm^2) and smears collected on Truss #2 in Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

ORNL Dwg.No. 96-4421

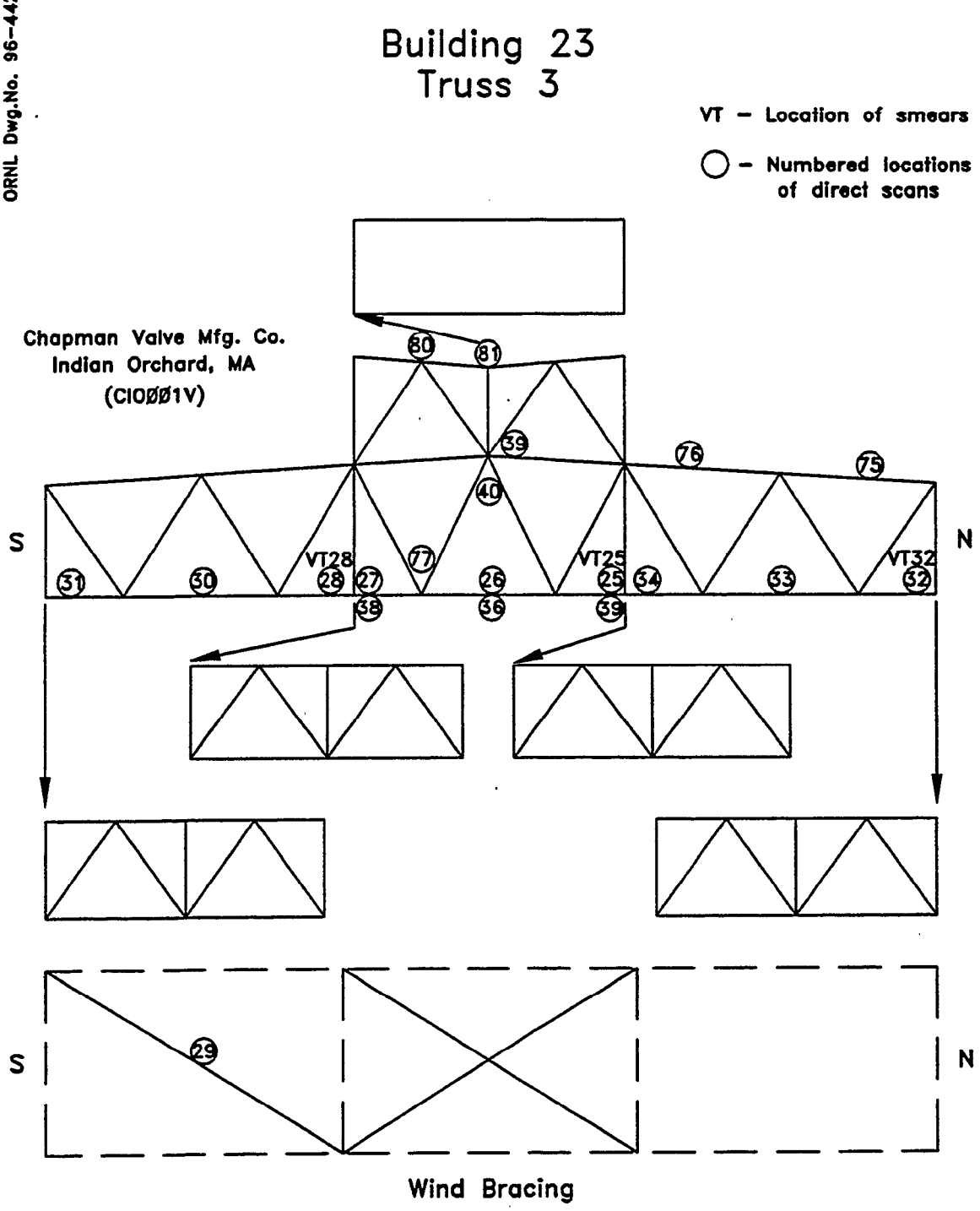

Fig. 5. Locations of directly measured beta–gamma activity levels (dpm/100 cm^2) and smears collected on Truss #3 in Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

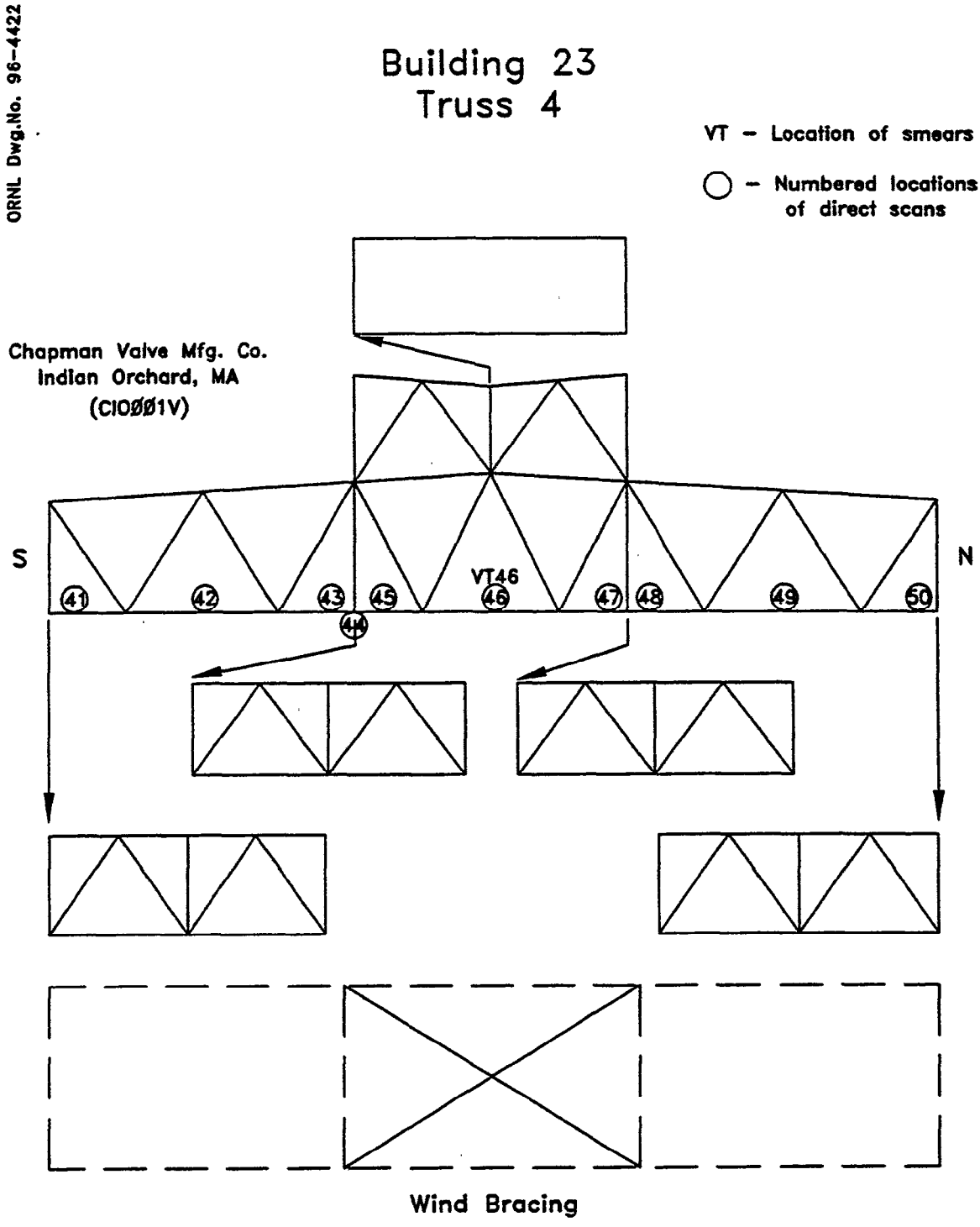

Fig. 6. Locations of directly measured beta–gamma activity levels (dpm/100 cm^2) and smears collected on Truss #4 in Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

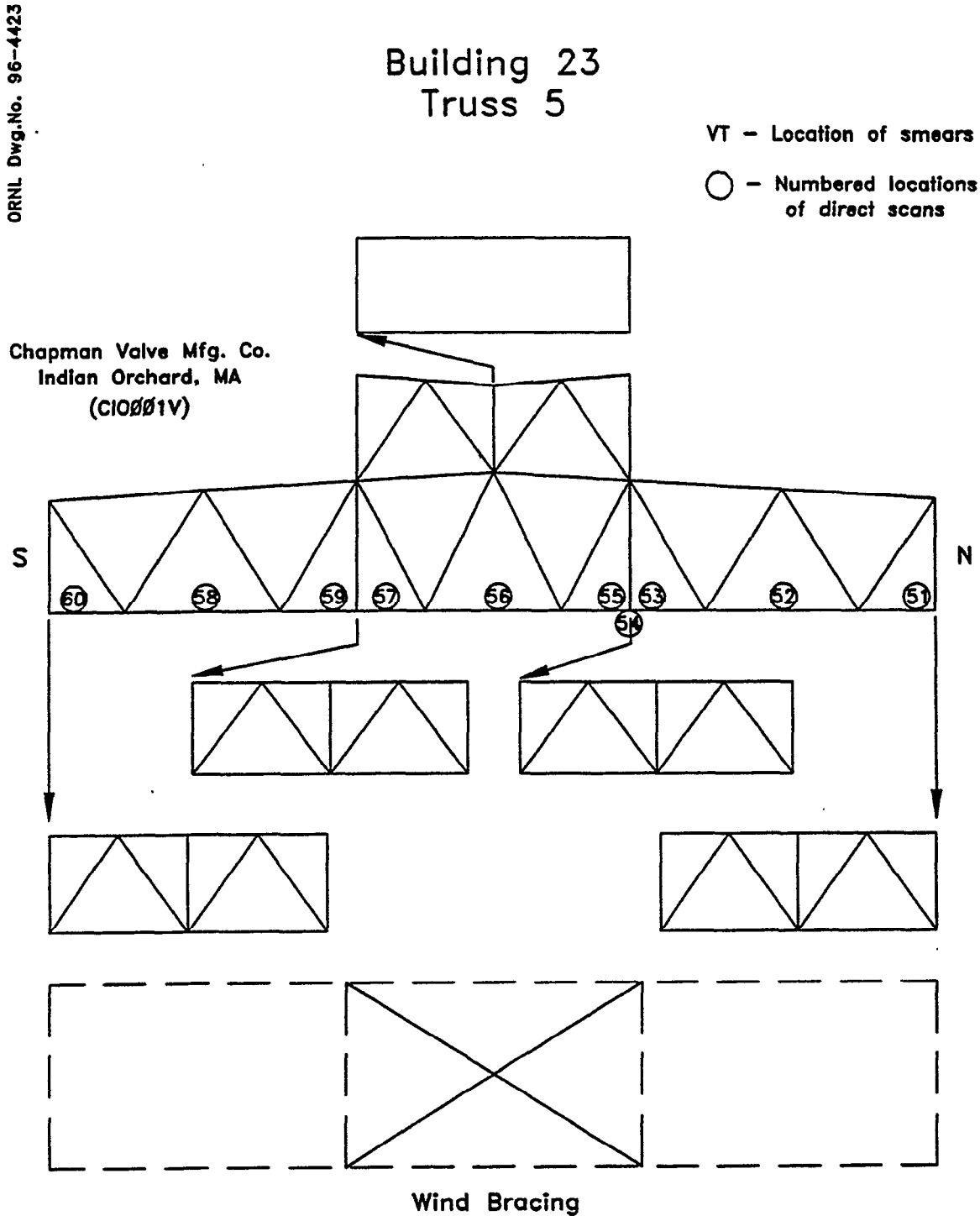

Fig. 7. Locations of directly measured beta–gamma activity levels (dpm/100 cm^2) and smears collected on Truss #5 in Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

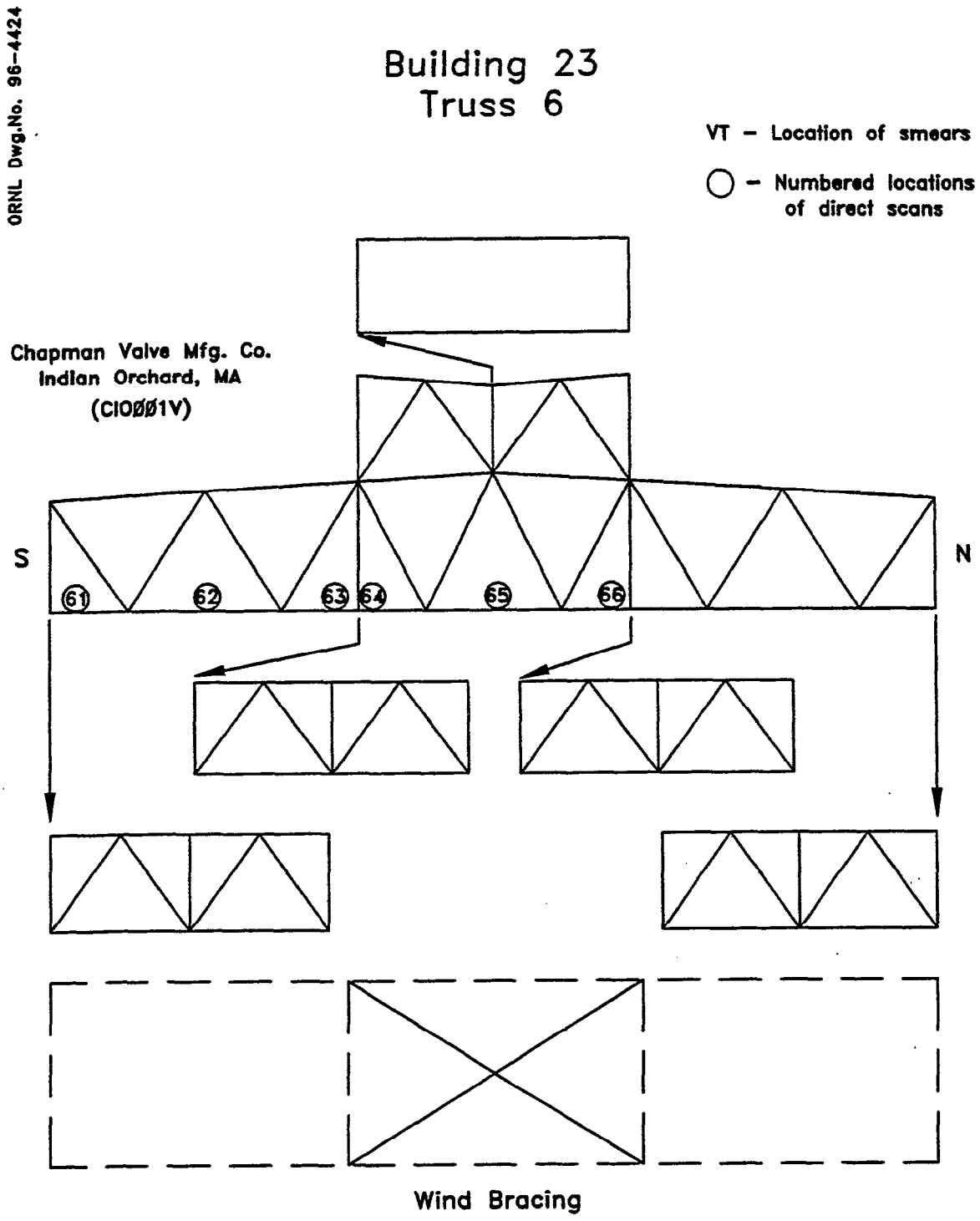

Fig. 8. Locations of directly measured beta-gamma activity levels (dpm/100 cm^2) and smears collected on Truss #6 in Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

ORNL Dwg.No. 96-4425

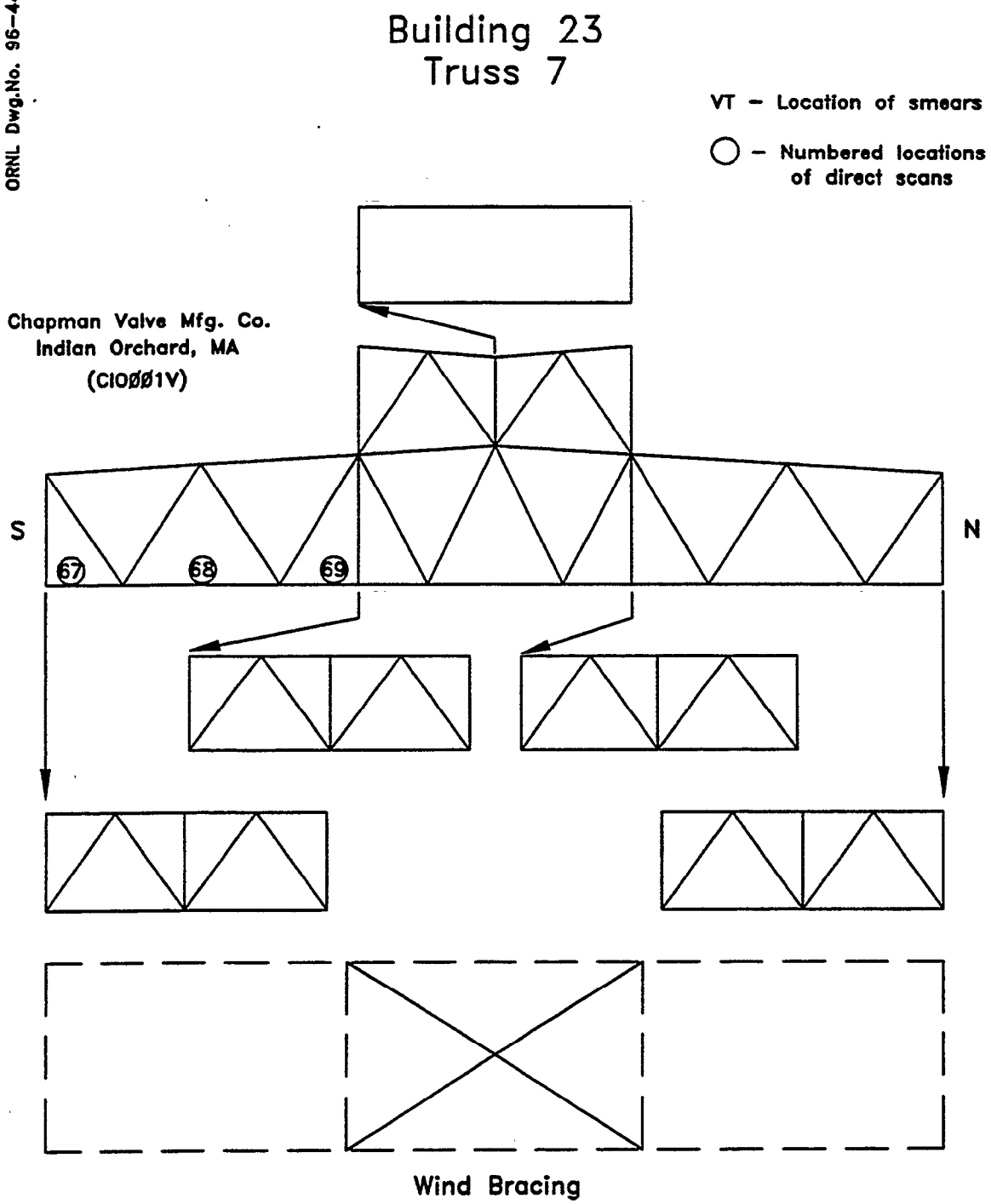

Fig. 9. Locations of directly measured beta–gamma activity levels (dpm/100 cm^2) and smears collected on Truss #7 in Building 23 at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

ORNL Dwg.No. 96-4426

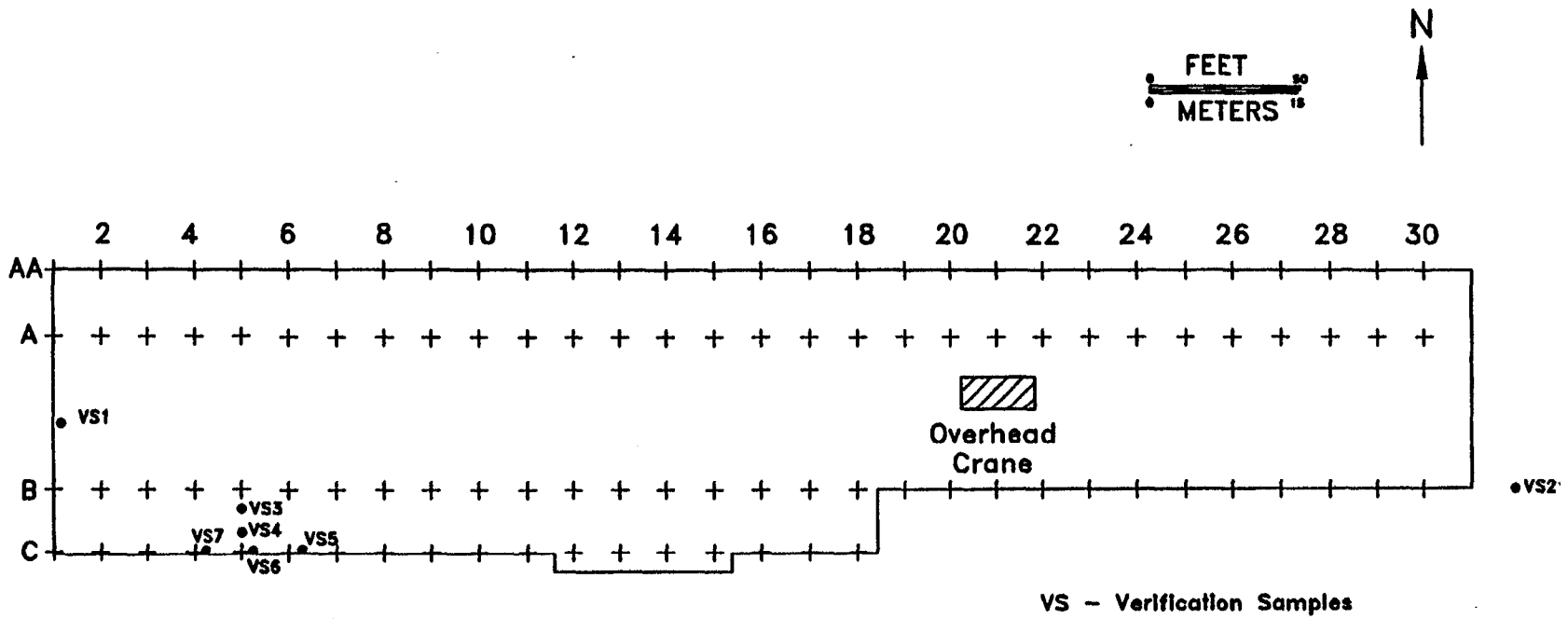

Fig. 10. Locations of soil samples collected at the site of the former Chapman Valve facility, Indian Orchard, Massachusetts.

Table 1. Applicable guidelines for protection against radiation
(Limits for uncontrolled areas)

Mode of exposure	Exposure conditions	Guideline value
Gamma radiation	Indoor gamma radiation level (above background)	20 μR/h[a]
Total residual surface contamination[b,c]	^{238}U, ^{235}U, U-natural (*alpha emitters*) or Beta-gamma emitters[d]	
	Maximum	15,000 dpm/100 cm^2
	Average	5,000 dpm/100 cm^2
	Removable	1,000 dpm/100 cm^2
	^{232}Th, Th-natural (*alpha emitters*) or ^{90}Sr (*beta-gamma emitter*)	
	Maximum	3,000 dpm/100 cm^2
	Average	1,000 dpm/100 cm^2
	Removable	200 dpm/100 cm^2
	^{226}Ra, ^{230}Th, transuranics	
	Maximum	300 dpm/100 cm^2
	Average	100 dpm/100 cm^2
	Removable	20 dpm/100 cm^2
Beta-gamma dose rates	Surface dose rate averaged over not more than 1 m^2	0.20 mrad/h
	Maximum dose rate in any 100-cm^2 area	1.0 mrad/h
Radionuclide concentrations in soil (generic)	Maximum permissible concentration of the following radionuclides in soil above background levels, averaged over a 100-m^2 area	5 pCi/g averaged over the first 15 cm of soil below the surface; 15 pCi/g when averaged over 15 cm-thick soil layers > 15 cm below the surface
	^{226}Ra	
	^{232}Th	
	^{230}Th	
Derived concentrations	^{238}U	Site specific[e]

Table 1. (continued)

Mode of exposure	Exposure conditions	Guideline value
Guideline for non-homogeneous contamination (used in addition to the 100-m^2 guideline)[f]	Applicable to locations with an area ≤25 m^2, with significantly elevated concentrations of radionuclides ("hot spots")	$G_A = G_i(100/A)^{1/2}$, where G_A = guideline for "hot spot" of area (A) G_i = guideline averaged over a 100-m^2 area

[a]The 20 μR/h shall comply with the basic dose limit (100 mrem/yr) when an appropriate-use scenario is considered.

[b]DOE surface contamination guidelines are consistent with *NRC Guidelines for Decontamination at Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material*, May 1987.

[c]Limits not applicable for the first eight trusses on the west end of the building. Source: Memo, J. W. Wagoner II, Dirctor, Division of Off-Site Programs, Office of Eastern Area Programs, Office of Environmental Restoration, U. S. DOE, to L. K. Price, Director, Former Sites Restoration Division, Oak Ridge Field Office, U.S. DOE, July 27, 1995.

[d]Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr, ^{228}Ra, ^{223}Ra, ^{227}Ac, ^{133}I, ^{129}I, ^{126}I, ^{125}I.

[e]DOE guidelines for uranium are derived on a site-specific basis. Guidelines of 35-40 pCi/g have been applied at other FUSRAP sites. *Sources*: J. L. Marley and R. F. Carrier, *Results of the Radiological Survey at 4 Elmhurst Avenue, Colonie, New York (AL219)*, ORNL/RASA-87/117, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., February 1988; B. A. Berven et. al., *Radiological Survey of the Former Kellex Research Facility, Jersey City, New Jersey*, DOE/EV-0005/29, ORNL-5734, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., February 1982.

[f]DOE guidelines specify that every reasonable effort shall be made to identify and to remove any source that has a concentration exceeding 30 times the guideline value, irrespective of area (adapted from *Revised Guidelines for Residual Radioactive Material at FUSRAP and Remote SFMP Sites, April 1987)*.

Sources: Adapted from U.S. Department of Energy, *Radiation Protection of the Public and the Environment*, DOE Order 5400.5, April 1990, and U.S. Department of Energy, *Guidelines for Residual Radioactive Material at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites*, Rev. 2, March 1987; and *U. S. Department of Energy Radiological Control Manual*, DOE N 5480.6 (DOE/EH-256T), June 1992.

Table 2. Background radiation levels and concentrations of selected radionuclides in soil in the Beverly, Massachusetts, area

Type of radiation measurement or sample	Radiation level or radionuclide concentration	
	Range	Average
Gamma exposure rate at ground surface (μR/h)[a]	6-9	7
Concentration of radionuclides in soil (pCi/g)[a]		
^{226}Ra	0.70-1.8	1.1
^{232}Th	0.76-1.2	0.97
^{238}U	0.69-2.7	1.2

[a]Values obtained from six locations in the Beverly, Massachusetts, area.

Source: R. D. Foley, M. S. Uziel, *Results of the Radiological Survey at the Former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001)*, ORNL/RASA-92/1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1992.

Table 3. Direct and transferable beta-gamma and alpha measurements on overhead trusses in Building 23 for the verification of the former Chapman Valve Manufacturing Company facility

Location No.[a]	Directly measured activity: Beta-gamma (dpm/100 cm²)	Directly measured activity: Alpha (dpm/100 cm²)	Removable activity: Smear No.[b]
	Truss # 1		
1	24,000	c	VT1
2	5,000	330	VT2
3	13,500	c	VT3
4	6,000	c	VT4
5	1,800	430	VT5
6	8,400	c	VT6
7	3,000	112	VT7
8	1,800	c	VT8
9	6,000	c	VT9
	Truss # 2		
10	21,000	30	VT10
11	1,800	c	VT11
12	2,700	c	VT12
13	24,000	10[d]	VT13
14	9,500	c	VT14
15	18,000	c	e
16	27,000	c	e
17	4,500	c	e
18	24,000	10[d]	VT18
19	21,000	c	e
20	7,500	c	e
21	24,000	c	e
22	15,000	c	e
23	18,000	780	VT23
24	10,500	c	VT24
70	15,000	c	VT70
71	24,500	c	e
72	10,500	c	e
78	6,600	c	e
	Truss # 3		
25	6,600	c	VT25
26	4,800	c	e

Table 3 (continued)

Location No.[a]	Directly measured activity Beta-gamma (dpm/100 cm²)	Directly measured activity Alpha (dpm/100 cm²)	Removable activity Smear No.[b]
	Truss # 3 (cont)		
27	12,000	c	e
28	60,000	150	VT28
29	6,600	c	e
30	17,000	c	e
31	4,500	c	e
32	3,000	c	VT32
33	2,400	c	e
34	30,000	c	e
35underside	5,500	c	e
36underside	2,500	c	e
38underside	0[d]	c	e
39	24,000	c	e
40verticle truss	870	c	e
75	4,500	c	e
76	9,000	c	e
77	21,000		
80	600	c	e
81	1,350	c	e
	Truss # 4		
41	25,000	c	e
42	7,500	c	e
43	1,500	160	e
44underside	60[d]	c	e
45	900	c	e
46	4,200	c	VT46
47	13,000	c	e
48	13,000	c	e
49	2,400	c	e
50	3,000	c	e
	Truss # 5		
51	600	c	e
52	3,600	c	e
53	27,000	c	e
54underside	0[d]	c	e
55	9,000	c	e
56	2,700	c	e
57	900	c	e

Table 3 (continued)

Location No.[a]	Directly measured activity Beta-gamma (dpm/100 cm²)	Directly measured activity Alpha (dpm/100 cm²)	Removable activity Smear No.[b]
	Truss # 5 (cont)		
58	3,000	c	e
59	3,000	c	e
60	2,700	c	e
	Truss # 6		
61	4,500	c	e
62	~400	c	e
63	3,000	c	e
64	6,600	c	e
65	150[d]	c	e
66	12,000	c	e
	Truss # 7		
67	4,800	c	e
68	210[d]	c	e
69	6,000	c	e

[a]Locations of the direct readings on trusses are shown on Figs. 3-9.

[b]Smears (VT samples) are located on Figs. 3-9. All smears were below the guidelines for transferable alpha and beta measurements.

[c]Measurement was not taken. Review of post remedial action data indicated that because of dust and rust the primary measurable radiation was beta-gamma.

[d]Values not discernible from background.

[e]No smear taken.

Table 4. Concentrations of radionuclides in soil at the former Chapman Valve Manufacturing Company, Indian Orchard, Massachusetts (CIO001V)

Sample number[a]	Depth (cm)	Radionuclide concentration (pCi/g)[b]			
		^{226}Ra	^{232}Th	^{238}U	^{235}U
		Systematic soil samples[c]			
VS1	0-15	0.47±0.1	0.43±0.08	14 ±1	0.46 ±0.07
VS2	0-15	0.56±0.1	0.34±0.07	0.71±0.3	0.09±0.04
VS3	0-15	0.65±0.2	0.31±0.1	0.82±0.3	<0.10
VS4	0-8	0.57±0.1	0.36±0.07	1.1 ±0.4	<0.08
VS5	0-15	0.50±0.2	0.37±0.2	0.98±0.3	0.11±0.05
VS6	0-15	0.43±0.1	0.33±0.1	0.90 ±0.3	<0.05
VS7	0-15	0.63±0.1	0.27±0.06	0.87 ±0.3	0.08±0.03

[a]Locations of soil samples are shown on Fig. 10.

[b]Indicated counting error is at the 95% confidence level (± 2σ).

[c]Systematic samples are taken at locations irrespective of gamma exposure rates.

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memorandum

AUG 15 8 36 AM '95

DATE: June 9, 1995

REPLY TO ATTN OF: EW-93:Kopotic

SUBJECT: CHAPMAN VALVE - REMEDIATION SUCCESS DEFINITION

TO: James W. Wagoner II, Director, Division of Off-Site/Savannah River Programs, Office of Eastern Area Programs, EM-421, CL

The purpose of this memorandum is to document our verbally agreed upon definition for the successful completion of the upcoming Chapman Valve remedial action effort.

As you know, an integral part of the proposed remedial action includes the current site owner's plans to demolish the former industrial complex within the next couple of years. This coupled with the results from the December 1994 characterization effort resulted in FSRD proposing a cost effective and technically protective site remediation strategy which includes the use of supplemental standards. Current remediation plans are to remove radioactive material from the structure such that the maximum dose to a hypothetical demolition worker would not exceed 10 mrem during the demolition activities. Verification of the success of the remediation effort would be based upon the post remedial action survey and the IVC's survey. As an added protective measure it was suggested by Alexander Williams the IVC could also conduct an additional verification of the former Building 23 location after the owner has completed demolition activities.

Since the Department has little to no control as to when the owner would demolish the building, we will declare the cleanup complete when DOE's remediation activities have been completed. If you agree please sign below and return a copy of this letter to me.

Lester K. Price

Lester K. Price, Director
Former Sites Restoration Division

CONCUR:

James W. Wagoner II

J. W. Wagoner II, Director
Off-Site and Savannah River
Program Division

7/27/95
Date

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TO McDaniel, T ORG USACE

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MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

1060 Commerce Park Drive
Oak Ridge, TN 37830-6480
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July 16, 2003

Ms. Tomiann McDaniel
United States Army Corps of Engineers
CEMP-RS
441 G. Street, NW
Washington, D.C. 20314

Dear Ms. McDaniel:

Final Radiological Verification Survey at the Crane Valve Site, Indian Orchard, Massachusetts

A final radiological verification survey was performed at the Crane Valve site from May 27 to May 30, 2003, in accordance with DOE Proposal No. 2239-S532-A1 and the approved Statement of Work. Conflicts with other projects and inclement weather delayed the survey from the proposed Fall of 2002 until May of this year. The survey team consisted of Robert L. Coleman, Certified Health Physicist, and Michael E. Murray, Professional Engineer, both senior staff members at Oak Ridge National Laboratory (ORNL). Arrangements were made with Anthony Pantaleoni of Crane Company to gain site access and coordinate the survey.

Since the previous survey (ORNL/RASA-95/17) the structure (building 23) containing residual uranium contamination has been demolished and removed; only the concrete floors remain. The purpose of this survey was to verify that the demolition activities did not redistribute any of the structural contamination. Any possible contamination would be detected on the concrete surface or in the soil footprint of the building. As documented in previous work, the contamination that remained was on the overhead structures near the front of the building. Using engineering drawings and photographs, the potentially affected area was located and the existing physical conditions documented. Cover dirt and debris have been brought on to the site and approximately half of the area of concern has been covered. The cover material was estimated to slope up to a depth of 5 feet.

The survey consisted of a walkover scan for gamma emissions using sodium iodide detectors, surface scanning with GM pancake detectors, and collecting soil samples. The only area found with any radiation distinguishable from background radiation was a small area of about 2 square meters that contained firebrick. The firebrick had been found during previous surveys and analyzed to contain naturally occurring nuclides. A small piece of the firebrick was collected for analysis. All of the accessible surfaces were scanned for surface contamination with none found. Four samples were collected. Two of the samples were of soil in the footprint area, one of the fill material, and the aforementioned firebrick. The firebrick again contained only naturally occurring nuclides consistent with firebrick found elsewhere. The two footprint soil samples contained uranium-238 concentrations of 1.4 and 1.9 picocuries per gram which is slightly higher than the background concentration of 0.8 picocuries per gram but far below the cleanup criteria of 35 picocuries per gram.

Given the data from previous surveys and this last survey we conclude that the property meets the criteria for unrestricted use and that all survey obligations have been completed. Raw data and survey maps will be maintained in our files and available for inspection upon request. Please direct any questions to me at 865-574-5838 or murrayme@ornl.gov.

Sincerely,

Michael Murray

Michael Murray

cc: Robert Atkin, DOE, Oak Ridge
Laboratory Records - RC
Anthony Pantaleoni, Crane
Elizabeth Rudek, Bechtel National
Verden Spicer, Bechtel National
Sharon Wagner, U.S. Army Corps of Engineers

130172

Department of Energy

Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, Tennessee 37831— 8723

May 22, 1995

Mr. Stephen S. Ball
Massachusetts Department of
Environmental Protection
436 Dwight Street
Springfield, Massachusetts 01103

Mr. Thomas F. O'Connell
Massachusetts Department of
Public Health and Radiation
Control Program
305 South Street
Boston, Massachusetts 02130

Dear Messrs. Ball and O'Connell:

CHAPMAN VALVE SITE - TRANSMITTAL OF DATA AND RA APPROACH

Enclosed is the Technical Study and supporting documents for the Remedial Action planned for the Chapman Valve Site in Indian Orchard, Massachusetts. As a follow up to our meeting of 3 May 1995, this information is being provided to furnish you with the data and the technical details for the Department of Energy's (DOE) cleanup approach for the site. In particular, the Technical Study presents the characterization results and calculations showing the maximum dose to a hypothetical site worker involved in all phases of building demolition.

The former Chapman Valve property, currently owned by the Crane Company, was designated for inclusion into DOE's FUSRAP Program in 1992. A copy of the designation report along with DOE information about FUSRAP were provided to you at the May 3rd meeting. The characterization study of the site conducted in December 1994 confirmed that radiological contamination above DOE guidelines is confined to portions of the roof's western end, localized spots on the south and west exterior walls, and interior areas in the western end of Building 23. Based upon discussions with the property owner, the future use scenario for Building 23 is demolition. There are no plans to occupy or utilize the structure in the interim.

As outlined in the Technical Study, DOE residual contamination guidelines for the release of contaminated structures are contained in DOE Order 5400.5 (proposed rule 10 CFR 834), "Radiation Protection of the Public and the Environment." Although the Order is used by DOE to establish requirements for the control and cleanup of radiological contamination at FUSRAP sites, it is important to note that the basis for the Order was adopted from other regulatory sources such as: the U.S. Environmental Protection Agency's "Standards for Uranium Mill Tailings," (40 CFR 192); and, the Nuclear Regulatory Commission's "Standards for Protection Against Radiation," (10 CFR 20), and NRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors." In addition, data and information from the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP) were used in developing the technical basis for the Order.

The guidelines for surface contamination are specified in disintegrations per minute (dpm) per unit area (100 square centimeters). The guidelines applicable to this site are 5000 dpm/100 cm^2 average, 15000 dpm/100cm^2 maximum, and 1000 dpm/100 cm^2 removable for alpha and beta/gamma emitting radionuclides. Soil (volumetric) guidelines for uranium are site-specific, based upon site characteristics and use. A site-specific uranium soil guideline has not been developed for Chapman Valve, but values typically range between 50 and 100 pCi/g for total uranium. A site-specific uranium guideline was not developed because no soil contamination was identified.

Also outlined in DOE Order 5400.5 are radiation dose limits for members of the general public. The maximum allowable annual dose to a member of the general public is 100 millirem. As discussed in our meeting, DOE Order 5400.5 also allows for the application of supplemental limits if specific circumstances indicate that the cleanup guidelines are not appropriate for a given property or portion of that property. If the cost of a remedial action is unreasonably high relative to the benefit that is achieved with removal of the radioactive material, supplemental limits may be warranted. These limits must be shown to be protective under very conservative scenarios and the potential dose from the remaining residual contamination must be below the allowable dose limit of 100 millirem per year.

Average radiological contamination levels in Building 23 on structural surfaces ranged 1000-29000 dpm/100 cm^2 beta/gamma with isolated spots of elevated activity. Results of analysis of dust samples taken from interior trusses ranged from approximately 7 to 8000 pCi/g U-238 with average levels of 870 pCi/g. The highest concentrations were observed on truss #2 and decreased significantly from west to east. Systematic analysis of roof composite materials collected from the western portion of the roof yielded average results of 83 pCi/g for U-238.

Work plans for remediation at Chapman Valve include the decontamination of the west bridge crane, the mono hoist area and adjacent rooms, interior trusses 1-7, and the concrete base slab in bays 1-8. In addition, debris located in the building's western end will be surveyed for radiological contamination and either decontaminated for release or packaged for waste disposal. Areas in which we plan to take no action and utilize supplemental limits are the roof, interior trusses East of truss #7, interior walls, and the upper layer of flooring (wooden blocks).

As demonstrated in the Technical Study, following DOE's remediation efforts at the site, the calculated dose to the hypothetical demolition worker from any residual uranium will be less than 10 millirem. This dose was derived assuming that the worker participates in the demolition of all structural components. In addition, the assumptions used in the dose calculation regarding the exposure and the nature and extent of contamination are very conservative. This low dose and our remedial action approach will allow us to still certify and release the property for future use without radiological restrictions.

We would be glad to meet with you to discuss this matter in more detail and provide you with a technical briefing on our remediation plans should you so desire. As we have secured funding for this project in fiscal year 1995, your concurrence with our approach is requested by June 7, 1995. If you agree please sign the concurrence line below and return a copy to me. We would like to begin mobilizing to the site by mid to late June in order to complete remediation activities by September 1995.

We believe this is an appropriate approach given site conditions and probable future use of the property. Please contact me at (615) 576-9441 or Michael E. Redmon of Bechtel at (615) 576-4718 with any questions you may have regarding this data.

Sincerely,

James D. Kopotic

James D. Kopotic, Site Manager
Former Sites Restoration Division

Enclosures

cc w/o enclosures:
A. D. Pantaleoni, Crane Company
T. J. O'Brien, CON-TEST

Concurrence: _____ _____

Stephen S. Ball
MA Department of
Environmental Protection

Date

_____ _____

Thomas F. O'Connell
MA Department of Health/
Radiation Control Program

Date

131182

Department of Energy

Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, Tennessee 37831—8723

June 15, 1995

Mr. Stephen S. Ball
Massachusetts Department of
Environmental Protection
436 Dwight Street
Springfield, Massachusetts 01103

Mr. Thomas F. O'Connell
Massachusetts Department of
Public Health and Radiation
Control Program
305 South Street
Boston, Massachusetts 02130

Dear Messrs. Ball and O'Connell:

CHAPMAN VALVE SITE - TRANSMITTAL OF REVISED TECHNICAL STUDY

Enclosed is a copy of the revised Chapman Valve Technical Study for your records. As was discussed during our telephone conversation, during document processing the "micro" symbols were incorrectly changed to "milli" symbols in Appendix F, Hazard Assessment Calculations, causing the numbers stated in the document text to be much larger than the actual calculated numbers. I hope this error did not inconvenience you during your review of the document.

If you should have any questions or need additional information please feel free to contact me at the above address or call (615) 576-9441.

Sincerely,

James D. Kopotic

James D. Kopotic, Site Manager
Former Sites Restoration Division

131136

Commonwealth of Massachusetts
Executive Office of Environmental Affairs
Department of Environmental Protection
Western Regional Office

William F. Weld
Governor
Trudy Coxe
Secretary, EOEA
David B. Struhs
Commissioner

June 15, 1995

Anthony Pantaleoni
Crane Company
100 First Stamford Place
Stamford, Connecticut

Re: Springfield 1-0170
Crane Company
Pinevale, Goodwin, Moxon Sts.

Dear Mr. Pantaleoni:

The Department of Environmental Protection (the Department) has received a report entitled Technical Study for the Remedial Action at the Chapman Valve Site. This report was prepared by the Department of Energy (DOE) and received by the Department on 26 May 1995.

During the 1940s when Chapman Valve operated this site, there was some contamination from a uranium fire in the west end of building 23. This contamination consists primarily of uranium dust on the horizontal surfaces of beams and bracings in the roof of the interior of the building.

The DOE proposes to use a combination of vacuuming, washing, and removal to decontaminate this area of residual uranium in the west end of building 23. Following removal of the uranium, the DOE will survey the building for any residual contamination.

The DOE has asked the Department to review the report and work plan. The Department has reviewed the report and concurs with the remedial action plan.

The Department requests that a summary report be submitted following completion of all decontamination activities.

Thank you for your cooperation. If you have any questions regarding this matter, please contact Stephen Ball of this office.

Sincerely,

Alan Weinberg
Regional Engineer
Bureau of Waste Site Cleanup

AW:SSB
cranedoe.ltr
cc: Springfield, Mayor's Office
Springfield, Board of Health
Springfield, Office of Emergency Preparedness
Indian Orchard Citisens Council
ConTest, Inc.
James D. Kopotic, Department of Energy, Former Sites Restoration Division, Oak Ridge Operations Office, P.O. Box 2001, Oak Ridge, Tennessee 37831

436 Dwight Street • Springfield, Massachusetts 01103 • FAX (413) 784-1149 • Telephone (413) 784-1100

131136

The Commonwealth of Massachusetts

Executive Office of Health and Human Services
Department of Public Health
Radiation Control Program
305 South Street, Jamaica Plain, MA 02130
(617) 727-6214 · FAX (617) 727-2098

WILLIAM F. WELD
Governor
ARGEO PAUL CELLUCCI
Lieutenant Governor
GERALD WHITBURN
Secretary
DAVID H. MULLIGAN
Commissioner

June 14, 1995

James D. Kopotic, Site Manager
Former Sites Restoration Division
Department of Energy
Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, Tennessee 37831-8723

Dear Mr. Kopotic,

The Radiation Control Program has reviewed the Department of Energy's Technical Study (DOE/OR/21949) and the supporting documents for the remediation action proposed for the Chapman Valve Site in Indian Orchard, MA.

The proposed remediation of the Chapman Valve Site, now owned by Crane Company, is taking place under the Former Utilized Sites Remedial Action Plan (FUSRAP). The Chapman Valve Manufacturing Company performed work for the Manhattan Engineer District and the Atomic Energy Commission during the 1940's.

Based on the information received by the Radiation Control Program from the Department of Energy, the processing of uranium metals for the Manhatten Engineer District and the Atomic Energy Commission activities took place in Building 23 of the Chapman Valve Site. The Technical Study indicates that the forementioned activities have created areas of residual contamination which are above the unrestricted use release criteria.

The remedial action proposed for the Chapman Valve Site is based on the characterization studies performed for the Department of Energy during 1991 and 1994. The characterization studies indicate that radiological contamination above the Department of Energy's guidelines is confined to portions of the western end of Building 23.

James D. Kopotic, Site Manager
Former Sites Restoration Division
Department of Energy
Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, Tennessee 37831-8723

The remediation activities proposed are based on the Department of Energy Order 5400.5 which contains guidelines for the release of contaminated structures and for limits on the radiation doses to members of the general public.

The Radiation Control Program uses the same guidance documents (i.e. 10CFR 20 , NRC Regulatory Guide 1.86, etc) that the Department of Energy utilized to generate Order 5400.5. One difference is that the Radiation Control Program's unrestricted use criteria utilizes a total effective dose equivalent of 10 mrem per year.

As stated in your letter of May 25, 1995, the future use scenario of Building 23 is demolition. It is the Radiation Control Program's understanding that the demolition material should be able to be disposed of in an ordinary landfill. There are no plans to occupy or utilize the structure after remediation.

The Radiation Control Program concurs with the Department of Energy's proposed remediation action of the Chapman Valve Site. This concurrence is based on the information known about the Chapman Valve Site as of this date.

Sincerely,

Thomas F. O'Connell

Thomas F. O'Connell
Radiation Scientist
Radiation Control Program

cc: R. Hallisey
file
Steve Ball, DEP
A.D. Pantaleoni, Crane Company