2.8 References

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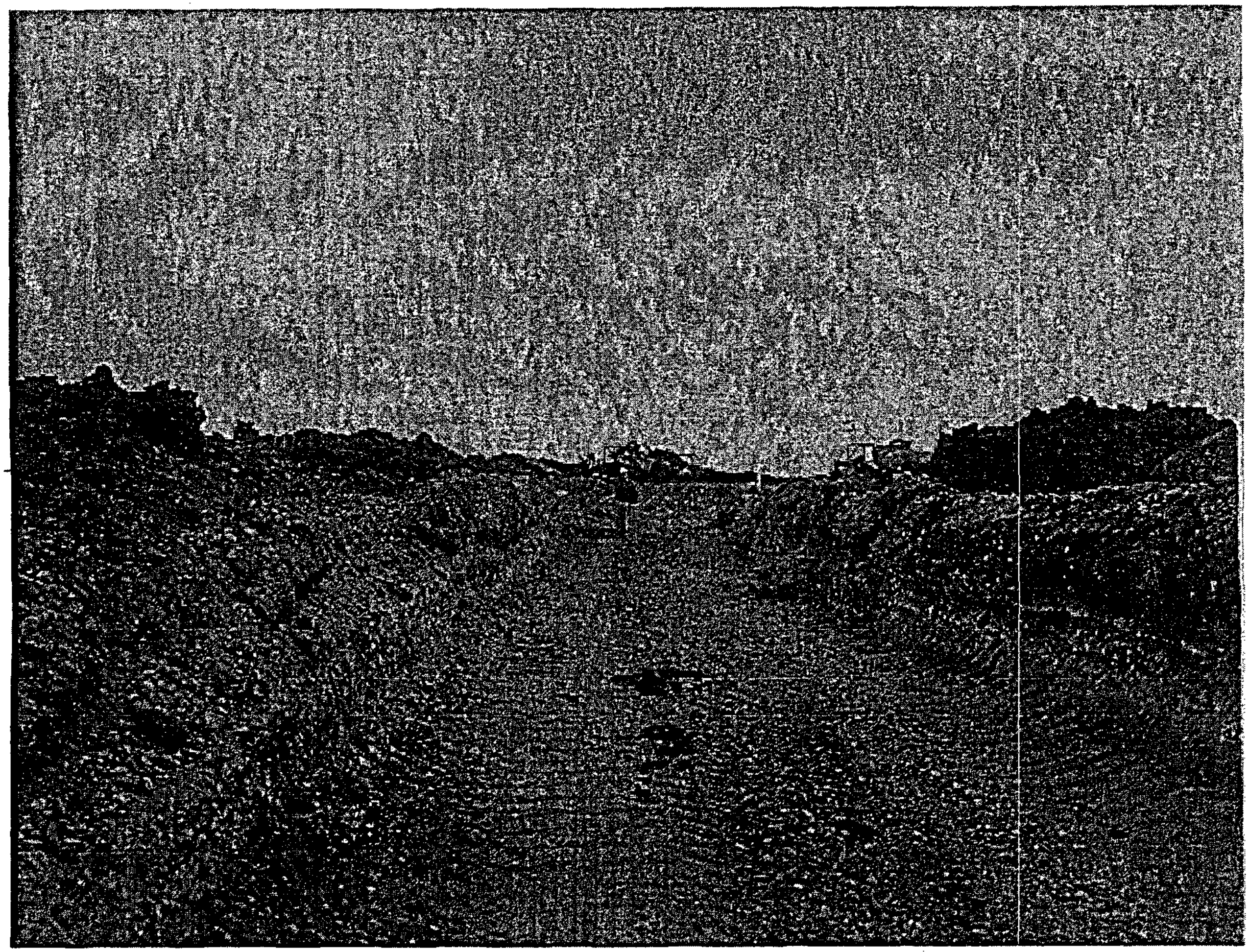
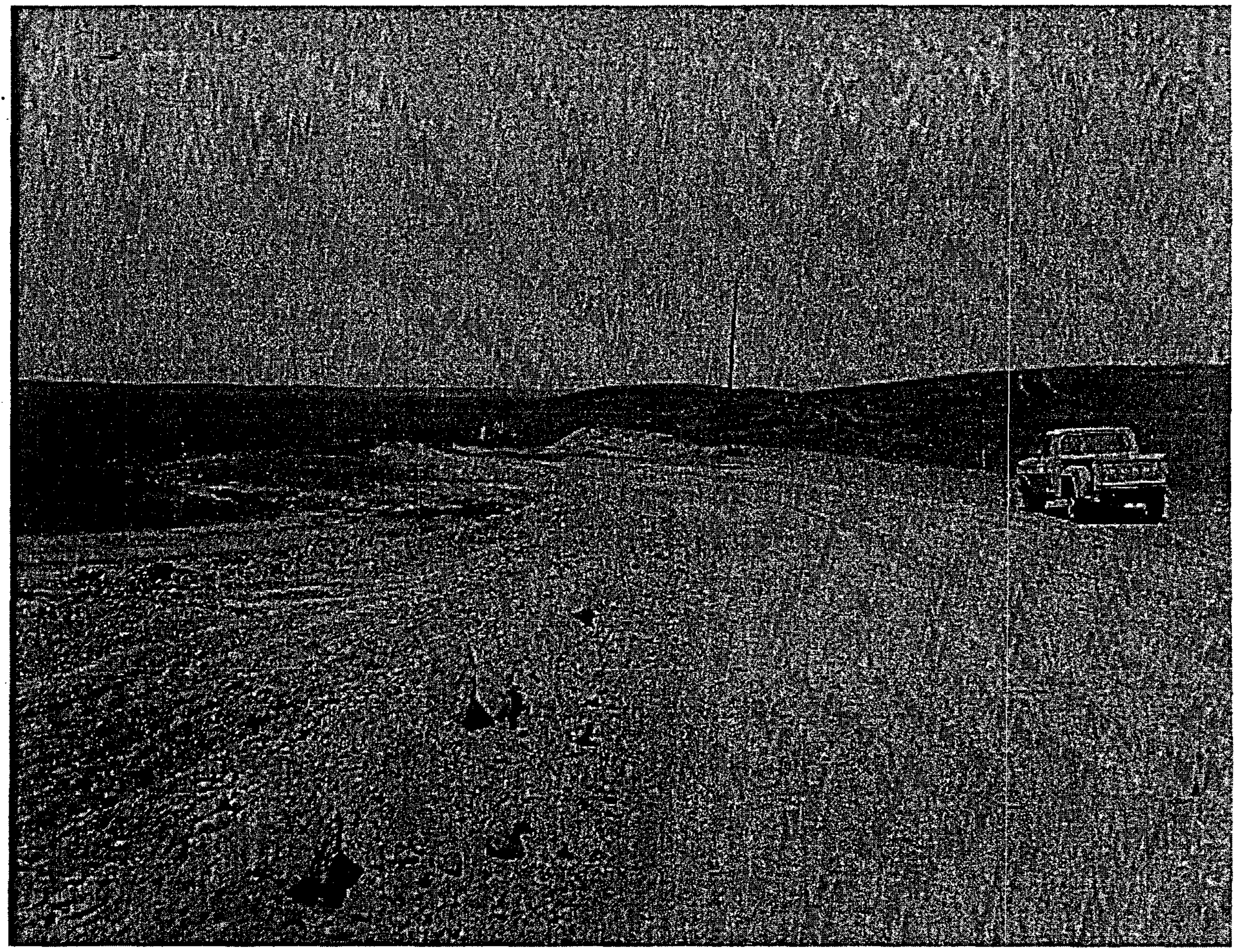
December 27, 1968 – Clay access road and Clay East road intersection

March 10, 1969 – Site Preparation, looking south at rough grading in turbine and reactor area



March 10, 1969 – Site Preparation, looking north at first rough grading cut into the turbine area

April 4, 1969 – Site Preparation, looking north along the east edge of the reactor containment structure

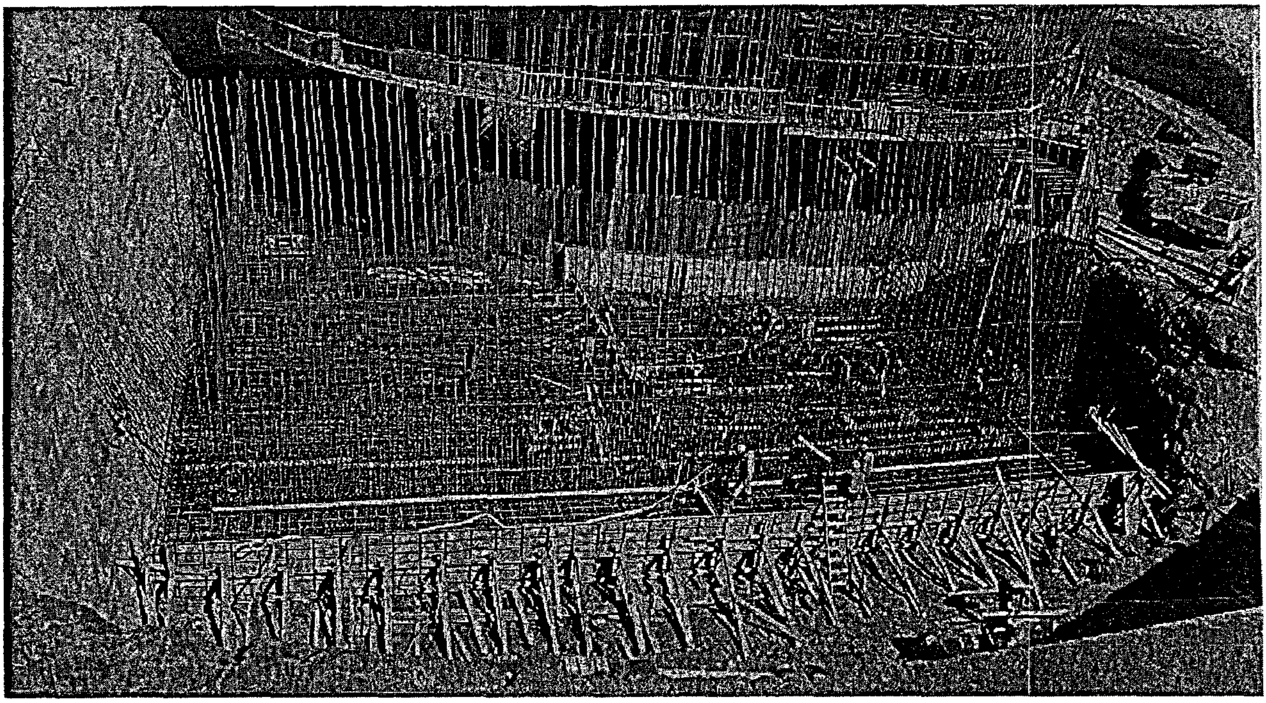


May 1, 1969 – Site Preparation, Twin Cities access road looking north

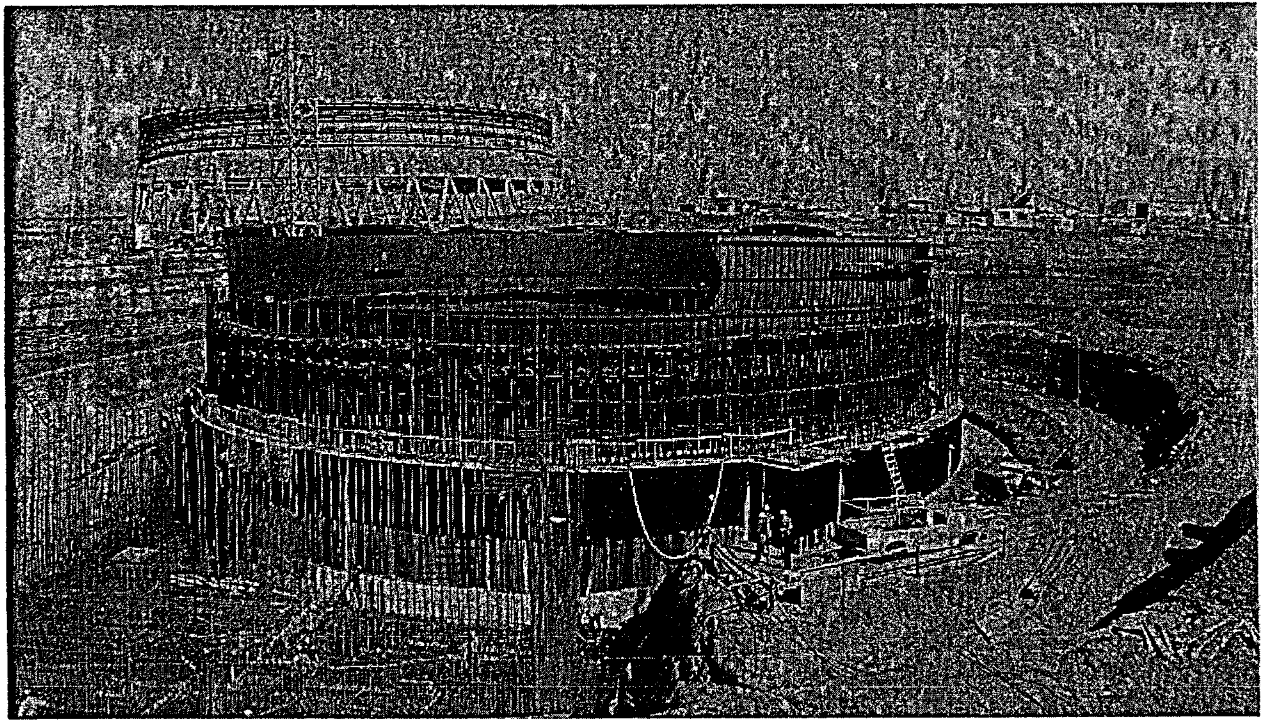
September 9, 1969 – forming for the fill concrete under the tendon access gallery floor slab

November 11, 1969 – a view of the shop and warehouse construction taken from the tower crane

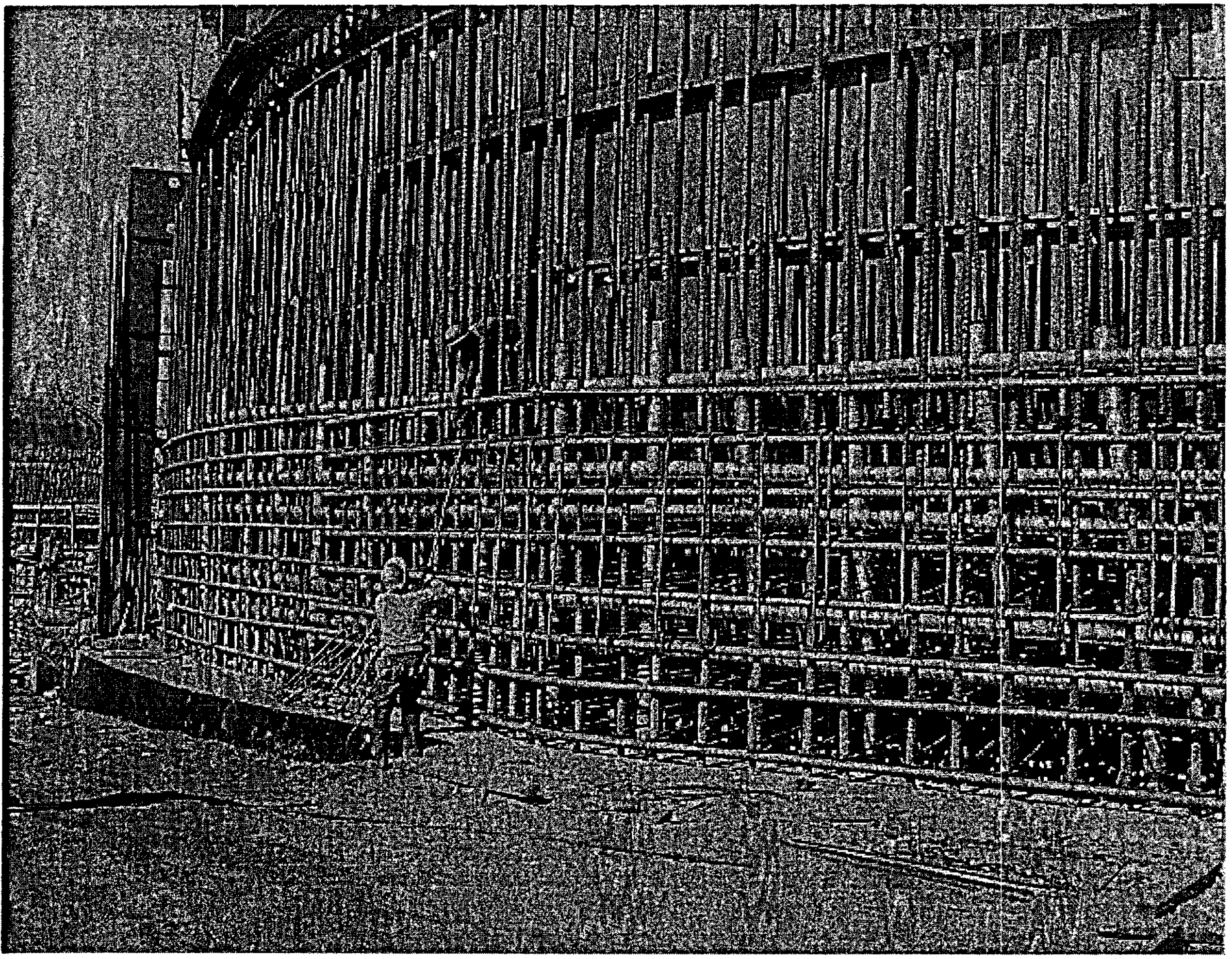
February 9, 1970 – installing floor liner plate on reactor building floor slab



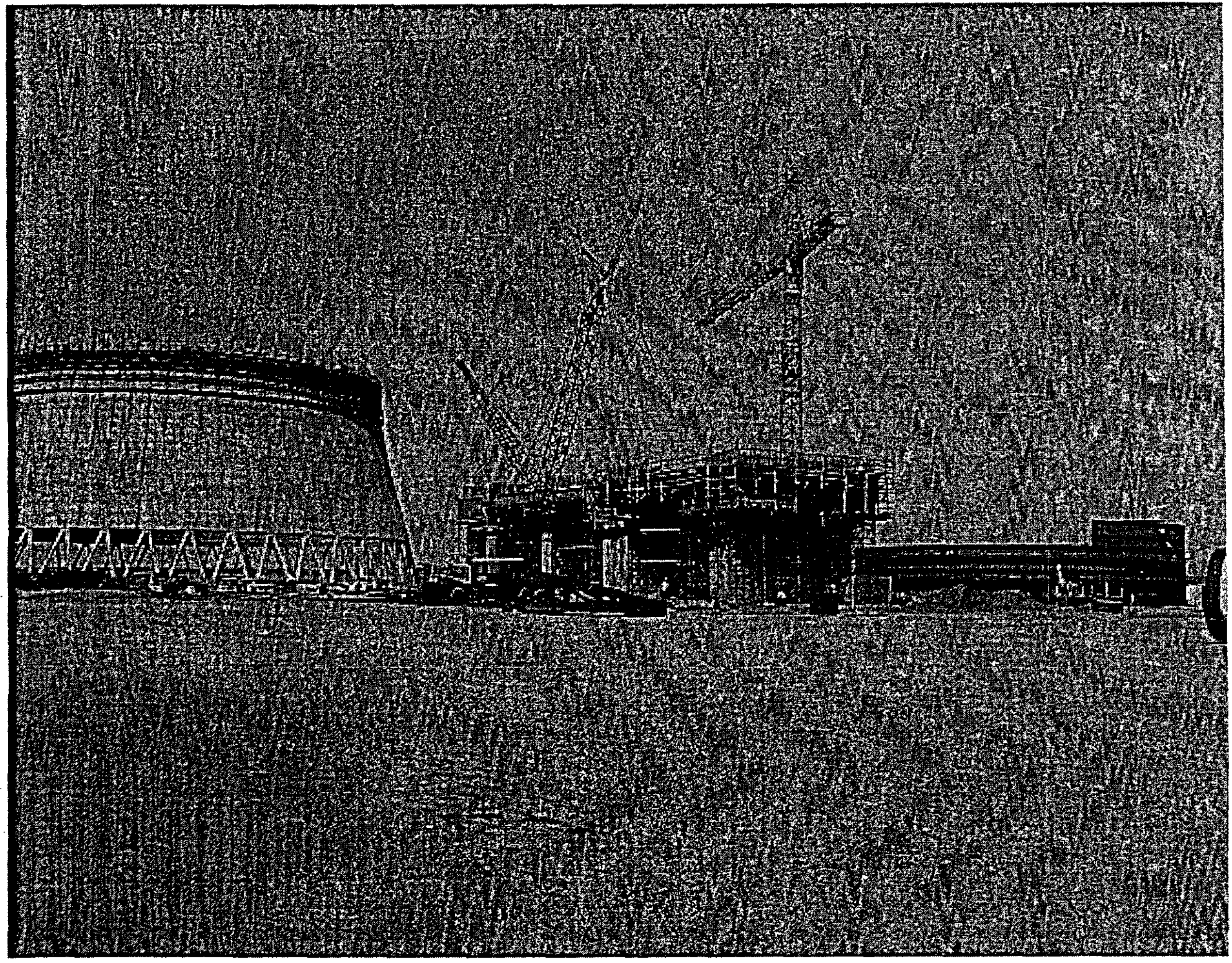
March 20, 1970 – a view looking north of the auxiliary building sub-basement slab



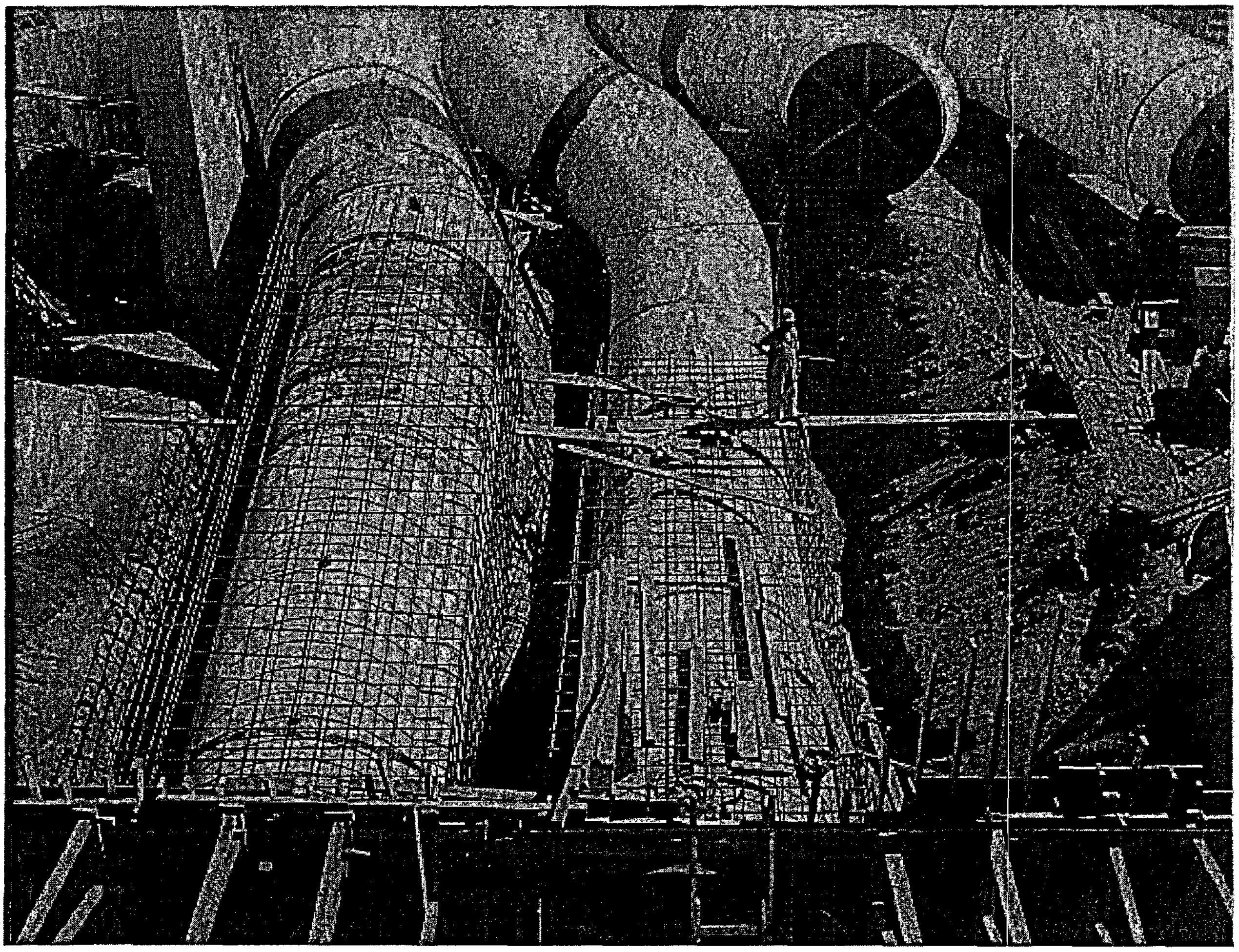
March 20, 1970 – overall view of the containment structure with cooling tower No. 1 in background



April 23, 1970 – reinforcing steel and tendon sheathing for the first lift of the containment wall



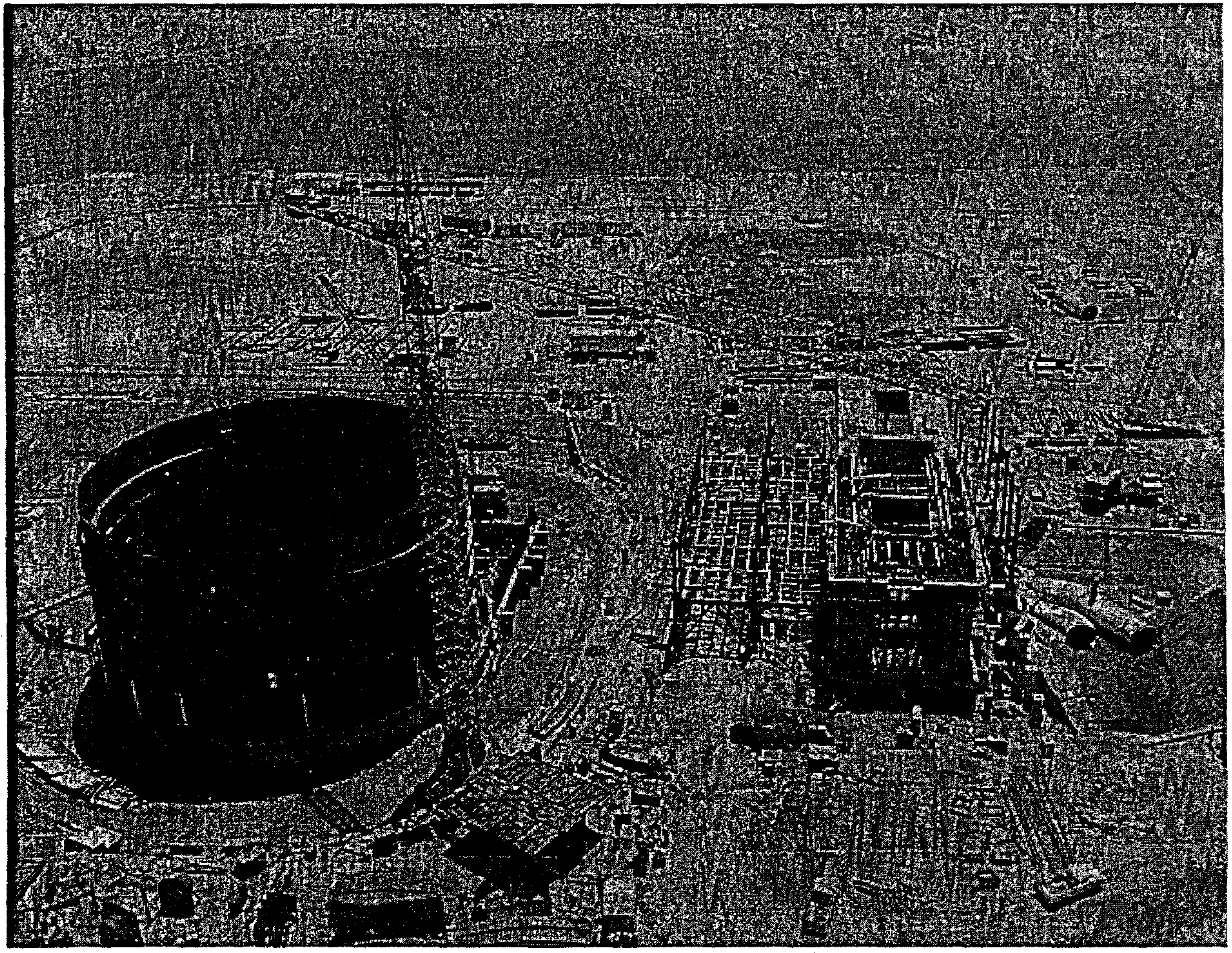
April 23, 1970 – overall view looking northeast



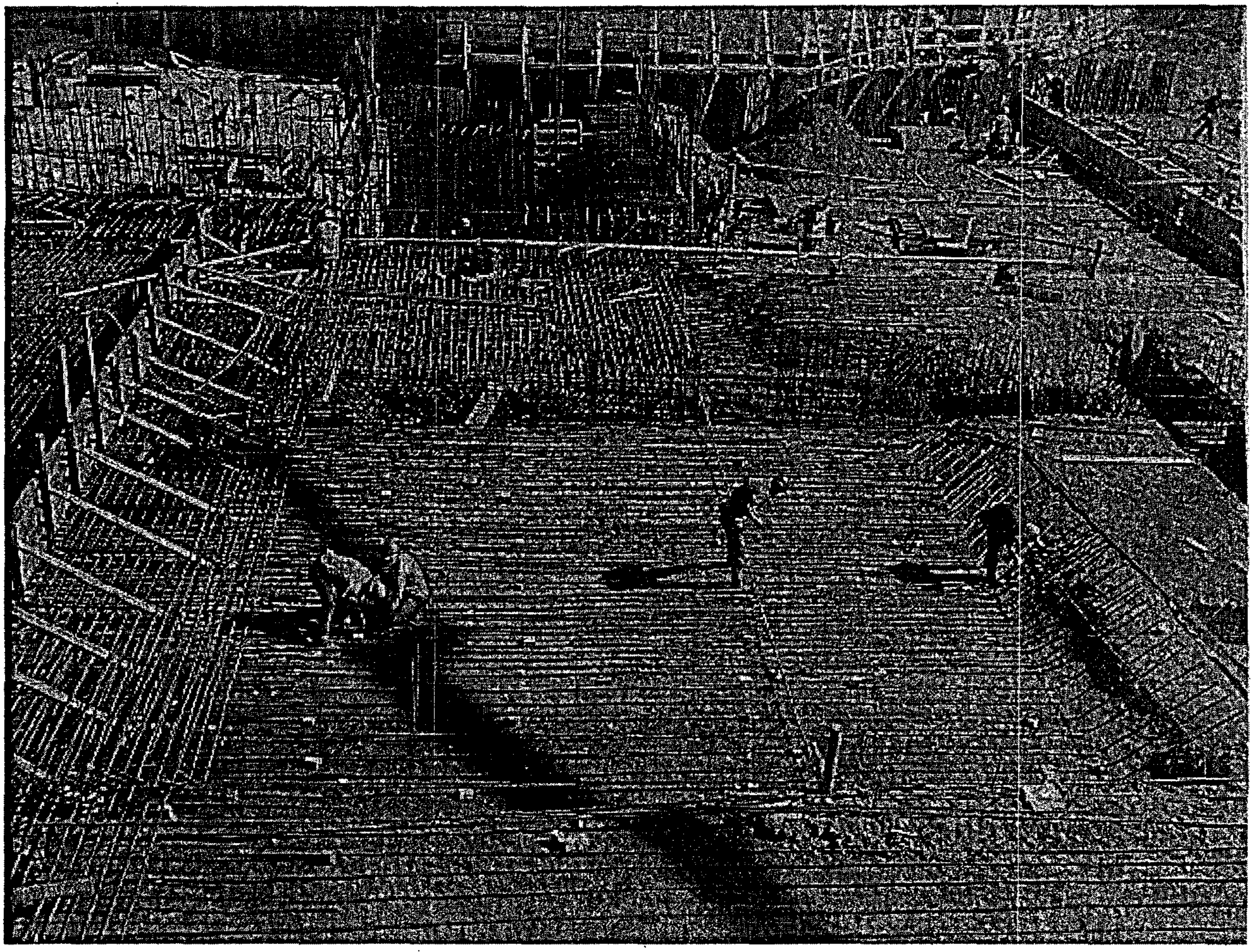
April 24, 1970 – 108" diameter circulating water pipe forming and reinforcement for concrete encasements

May 28, 1970 – looking west at the auxiliary building subbasement wall with the turbine building in the background

May 28, 1970 – view looking south at the turbine building with the reactor and auxiliary buildings on the extreme left



May 28, 1970 – view looking south at the turbine and reactor buildings



July 17, 1970 – auxiliary building showing reinforcing being placed

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⬚	IA BOUNDARY FENCE
▩	UNIQUE SUID SURVEY UNITS
▨	ROADS & RAILS
▨	PAVEMENT AREA
▨	LAND & GRASS AREA

LEGENDS

b	BETA DIRECT
g	GAMMA DIRECT
(XX)→	BETA SCAN
→[XX]	GAMMA SCAN
01	FLOOR
[XX]	LOWER WALL
⬠XX	UPPER WALL
$\underline{01}$	CEILING
$\overline{XX}$	EXTERIOR WALL
(XX)	SAMPLES
△XX	EQUIPMENT/FIXTURES

SMUD
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UTILITY DISTRICT

Figure 2–32
Legends & Notes

DWG: legends notes

C28

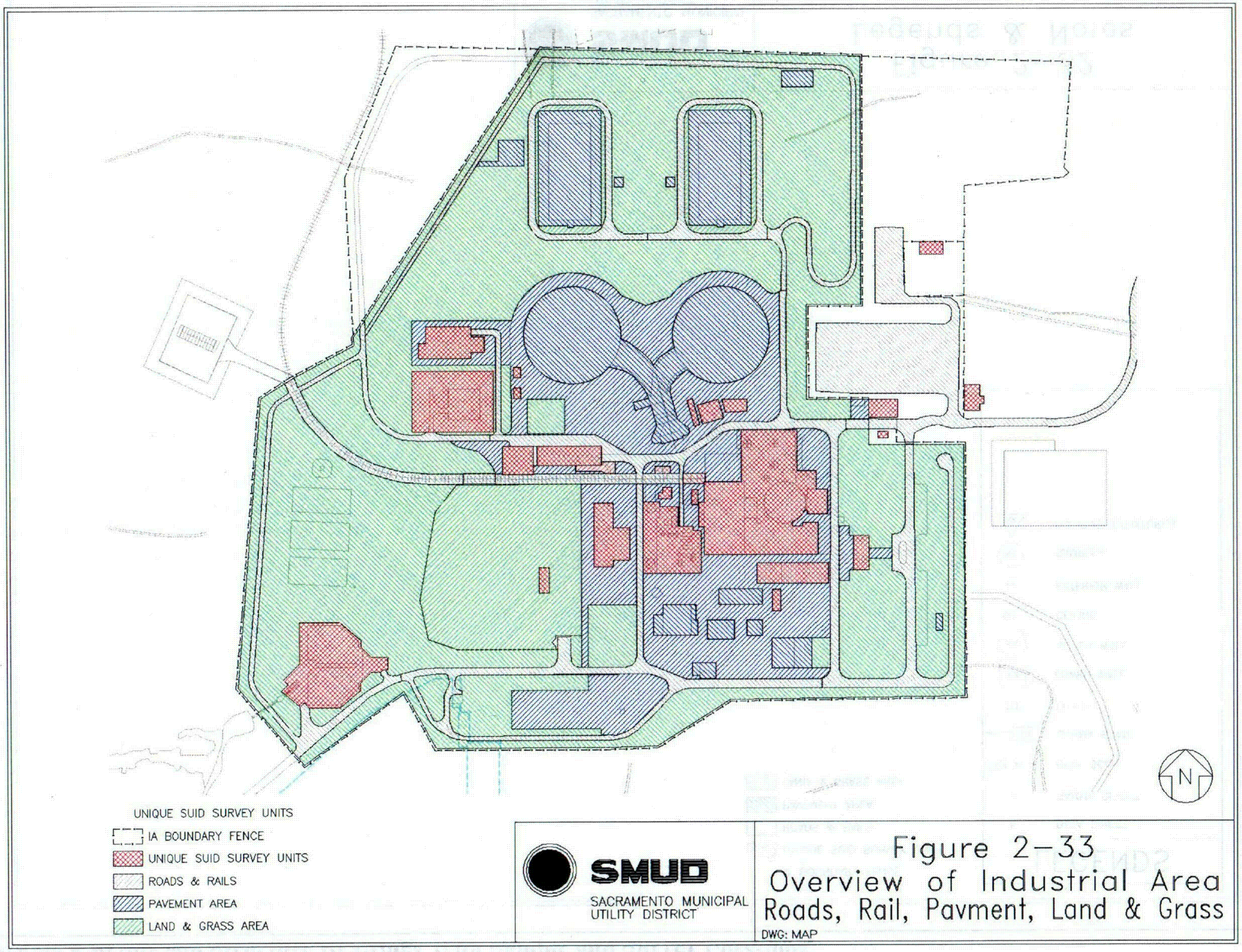

Figure 2-33
Overview of Industrial Area
Roads, Rail, Pavment, Land & Grass

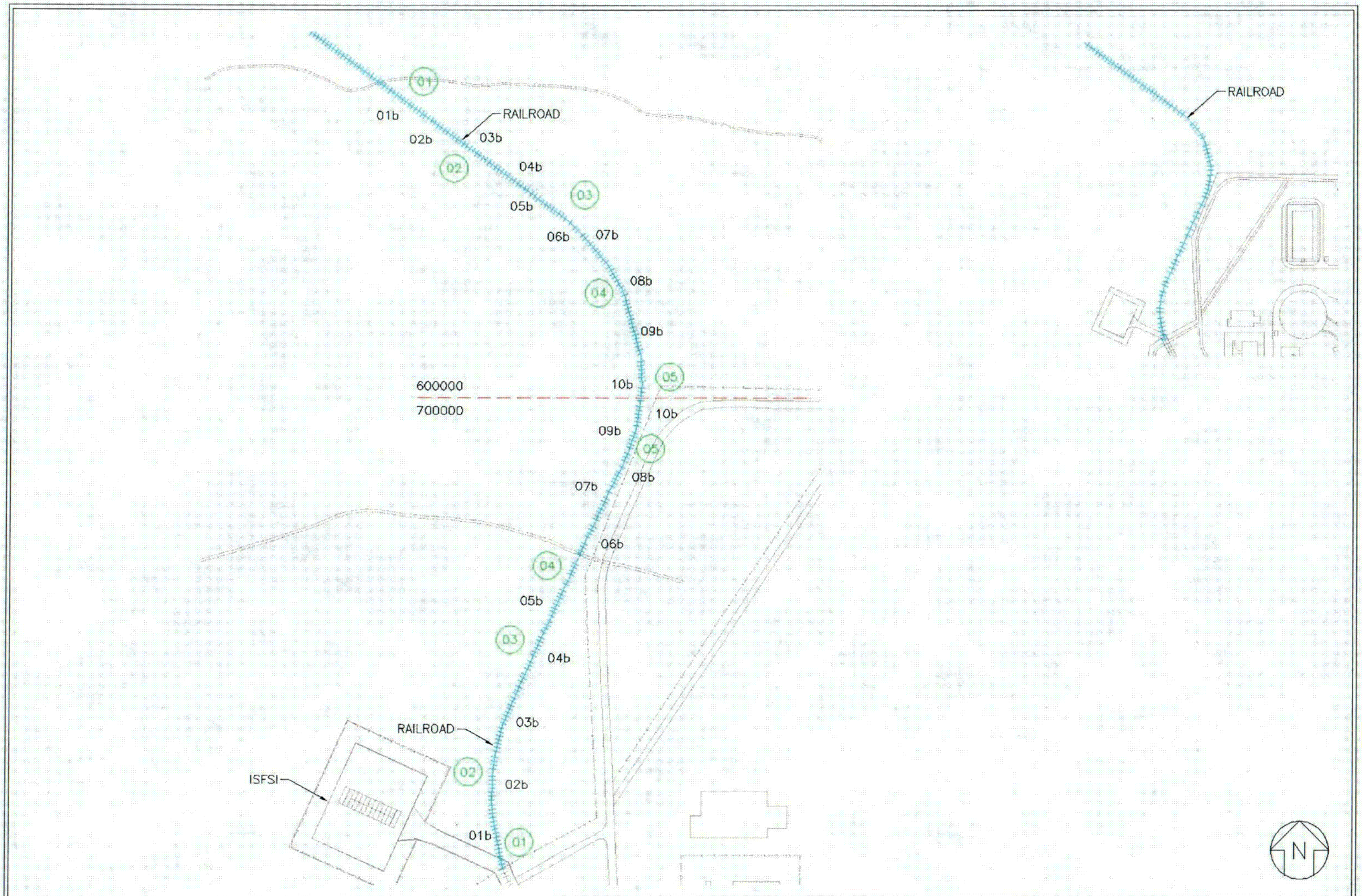

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Figure 2-34
Railroad
601000 & 701000

DWG: 601000

C30

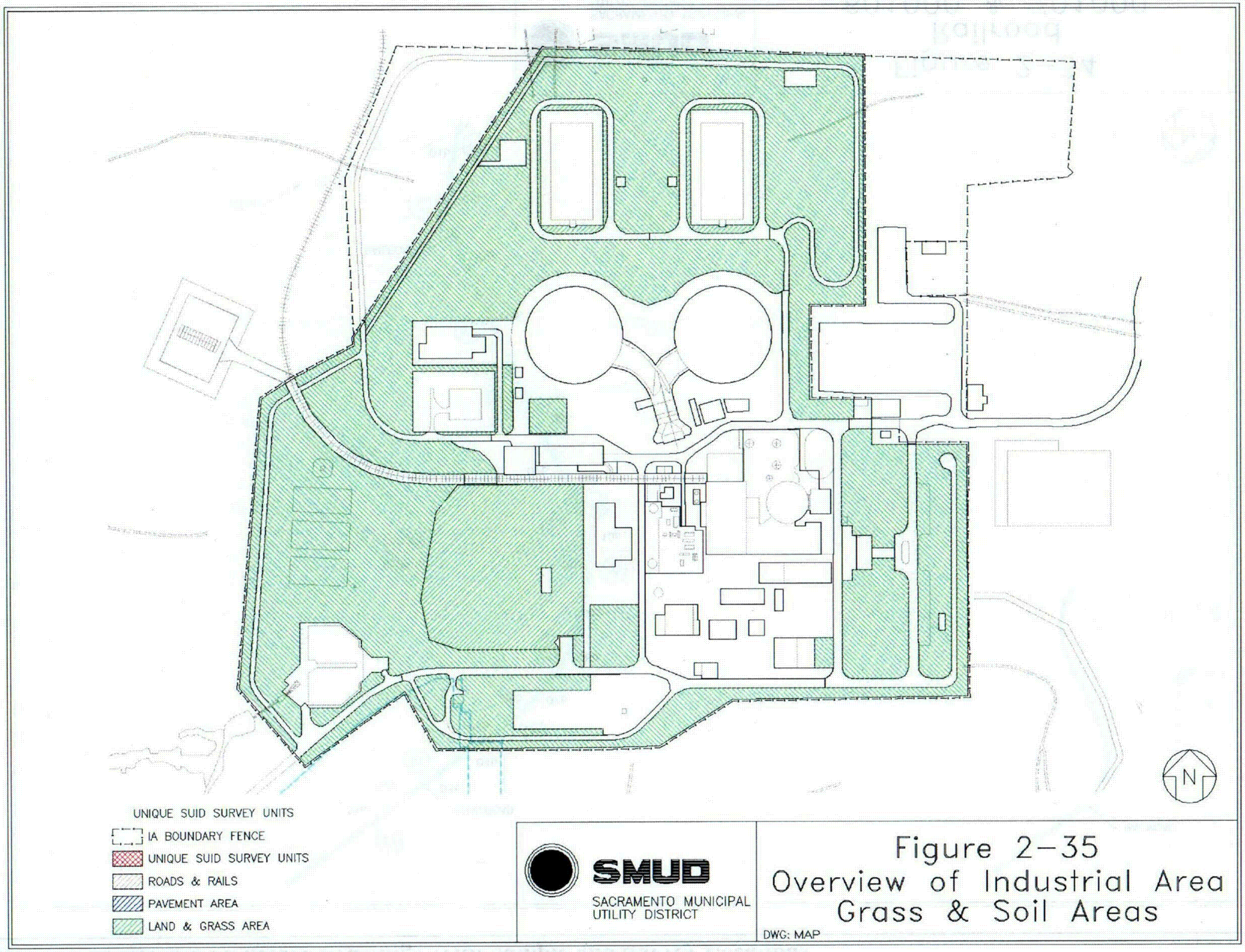

Figure 2-35
Overview of Industrial Area
Grass & Soil Areas

C31

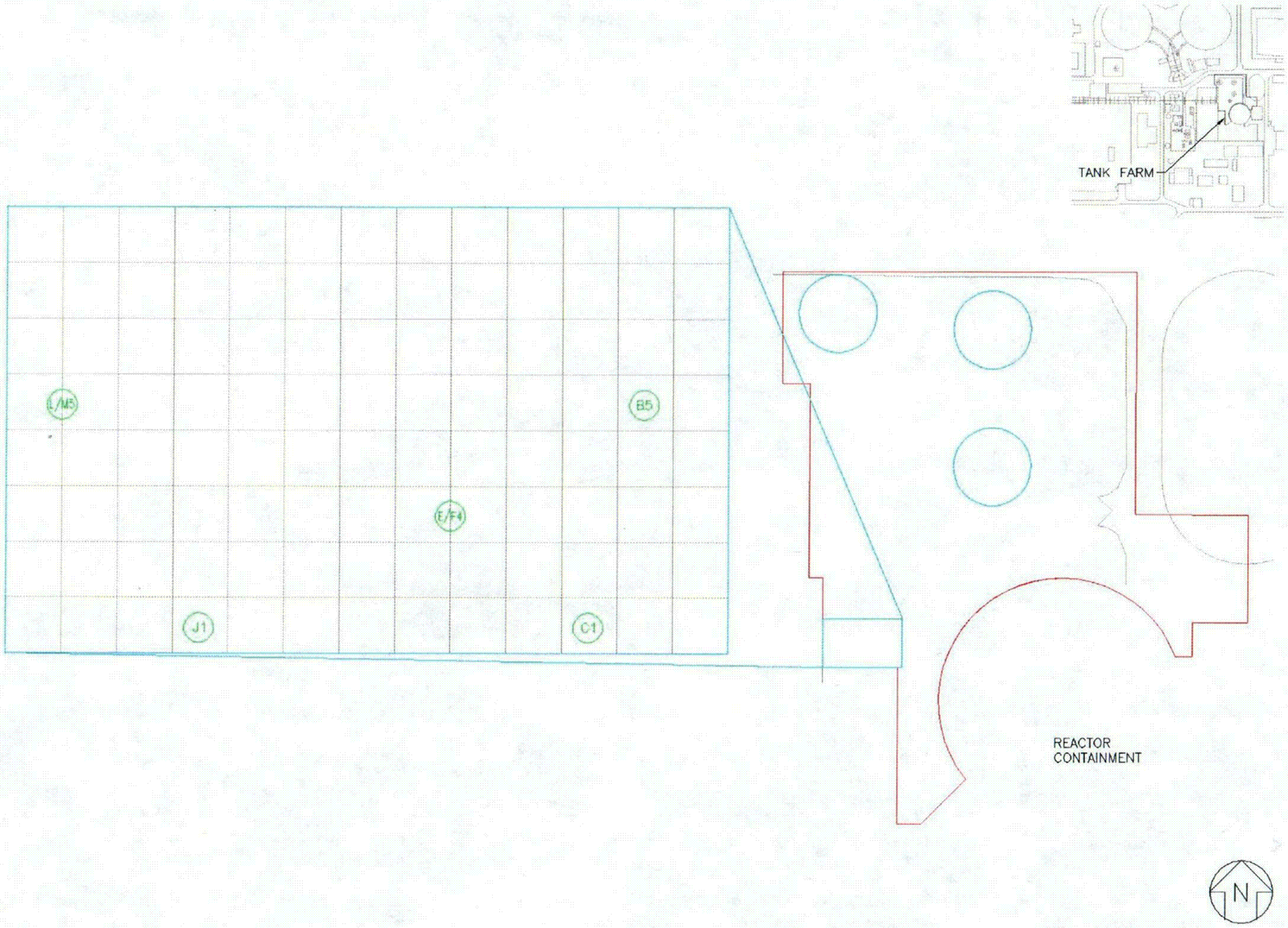

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Figure 2-36
Tank Farm
810001 & 2

DWG: 810001.01

C32

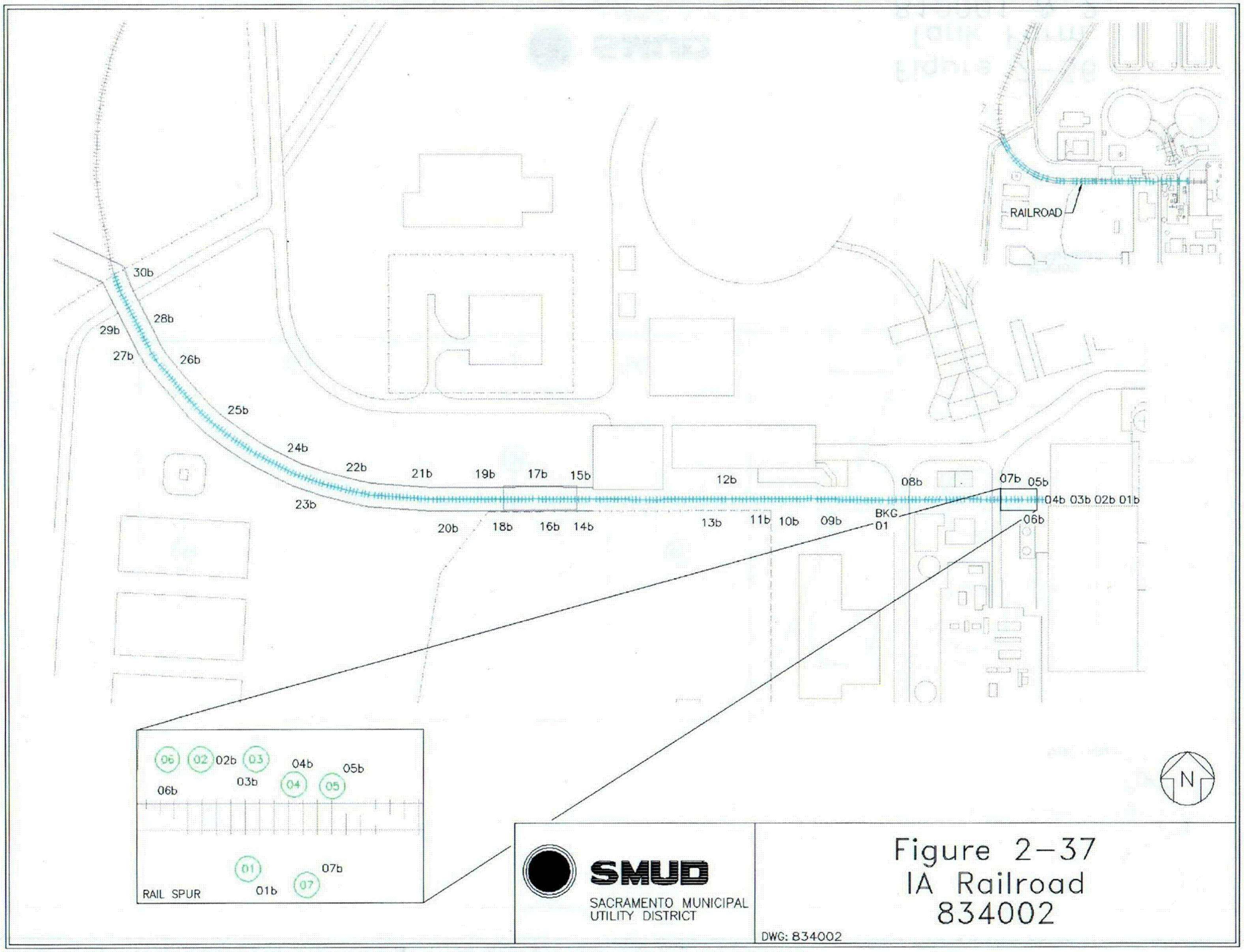
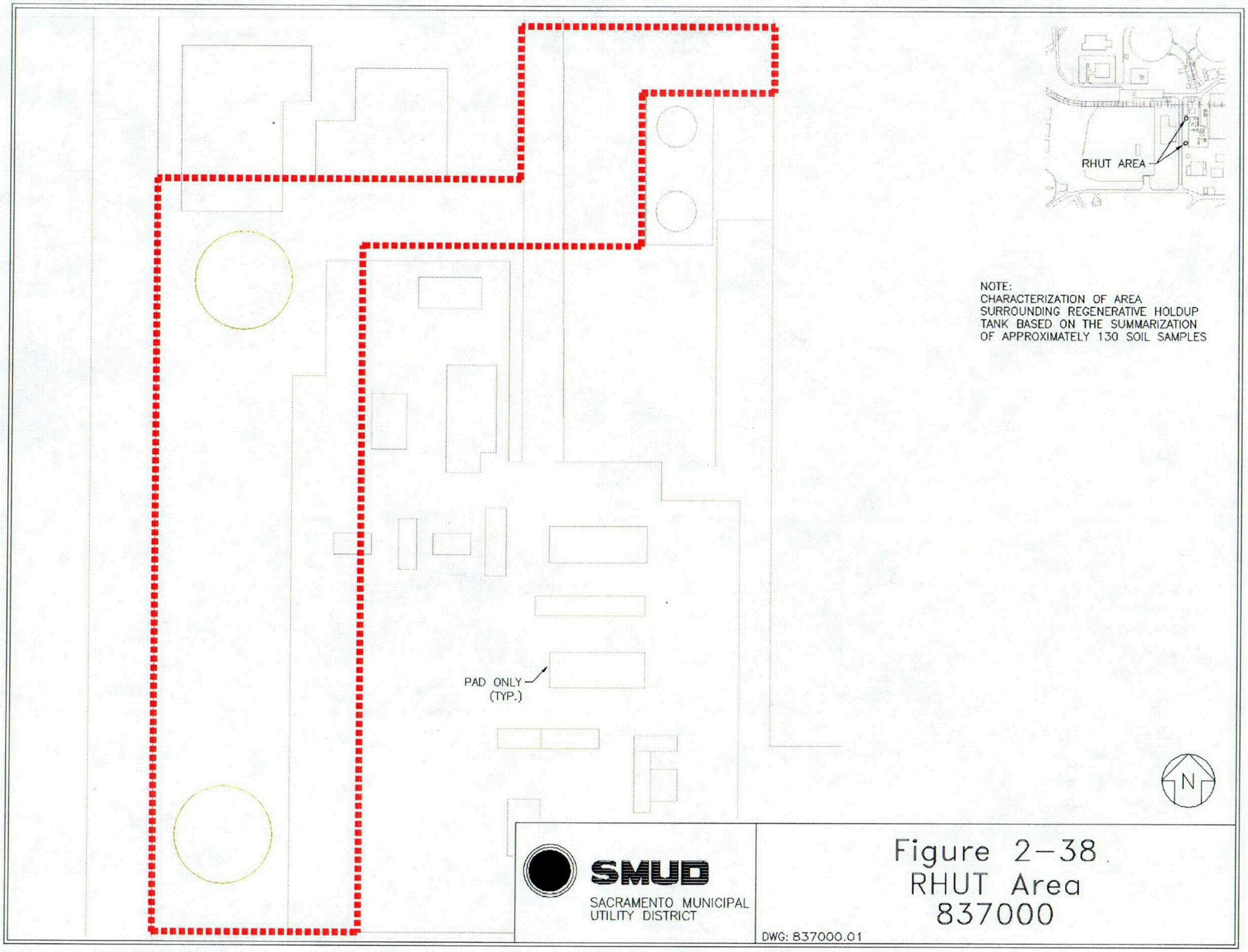


Figure 2-37
IA Railroad
834002

C33



NOTE:
CHARACTERIZATION OF AREA SURROUNDING REGENERATIVE HOLDUP TANK BASED ON THE SUMMARIZATION OF APPROXIMATELY 130 SOIL SAMPLES

Figure 2-38
RHUT Area
837000

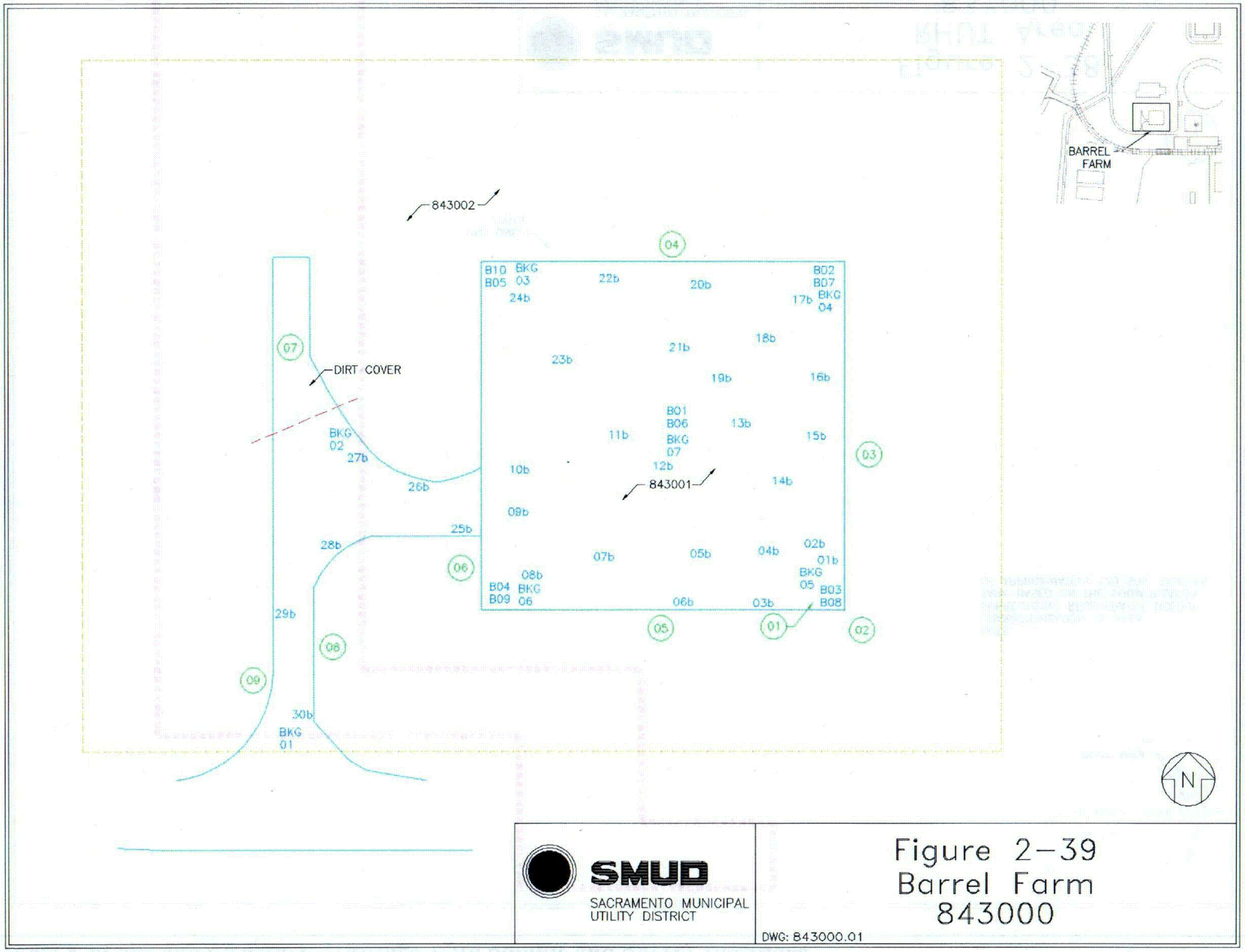


Figure 2-39
Barrel Farm
843000

C35

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Figure 2-40
Retention Basin Buffer
848002

DWG: 848002.01

C36

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Figure 2–41
Switch Yard
851000

DWG: 851000.01

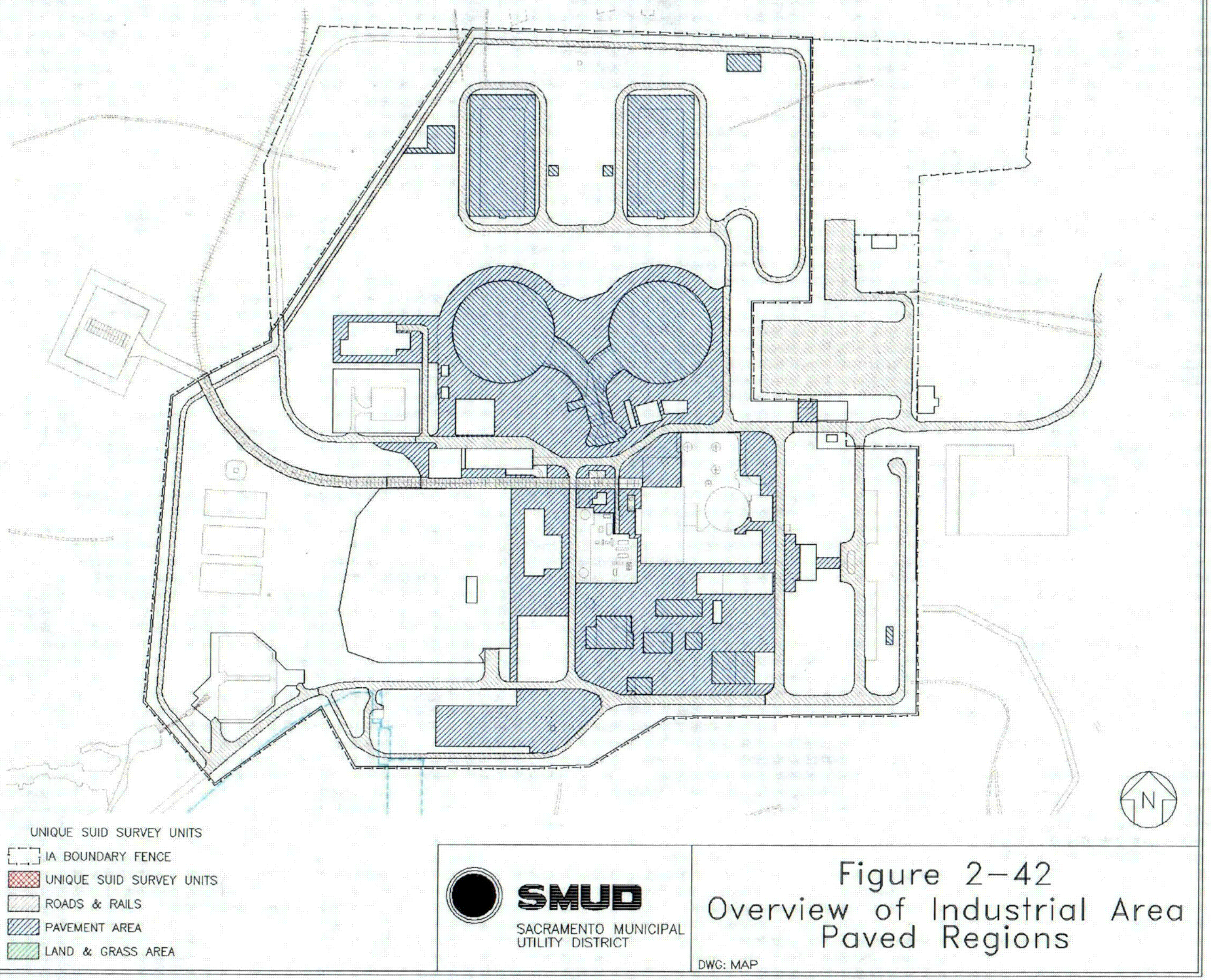

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Figure 2-42
Overview of Industrial Area Paved Regions

DWG: MAP

C38

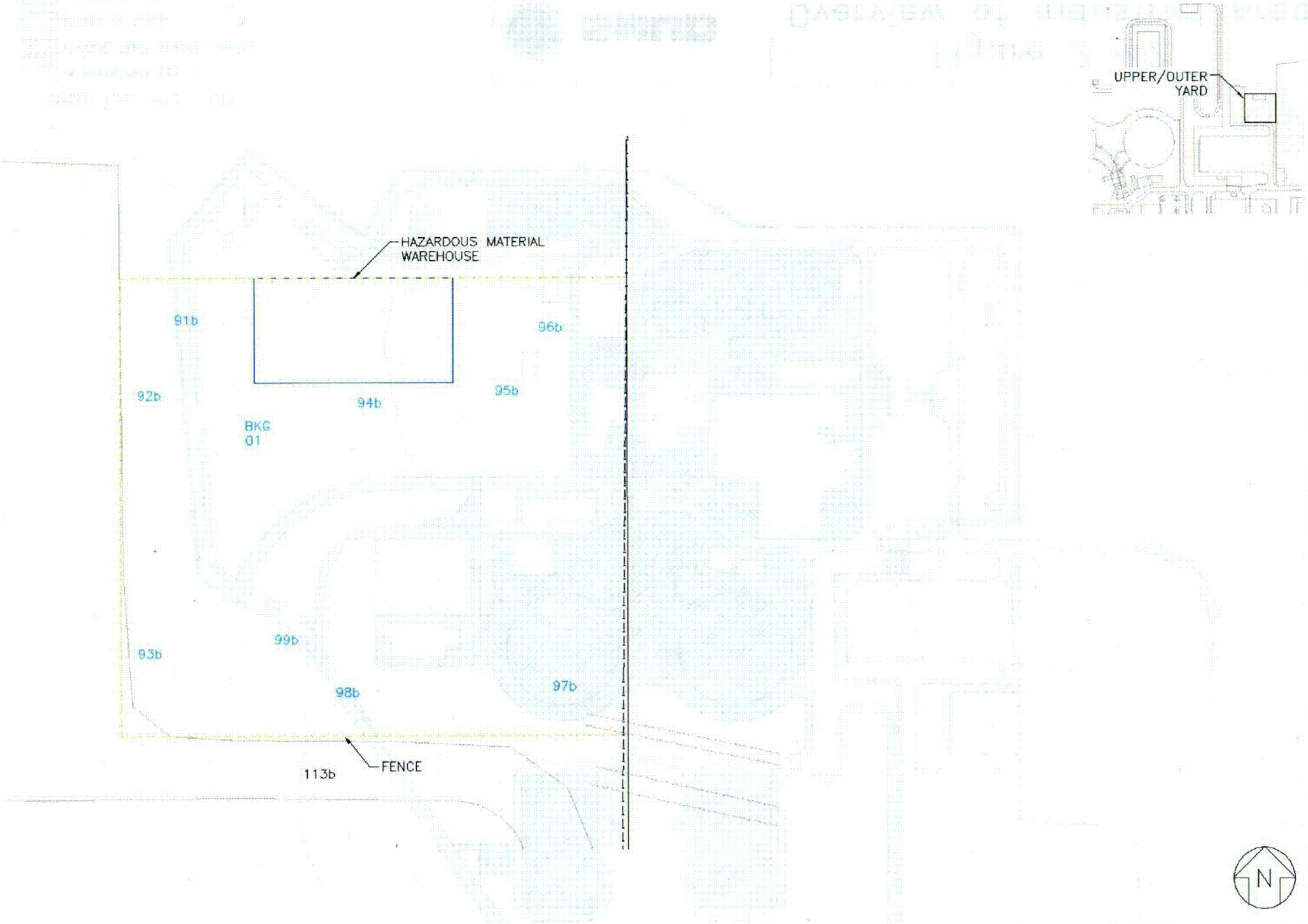

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Figure 2-43
Upper/Outer Yard
501003

DWG: 501003

C39

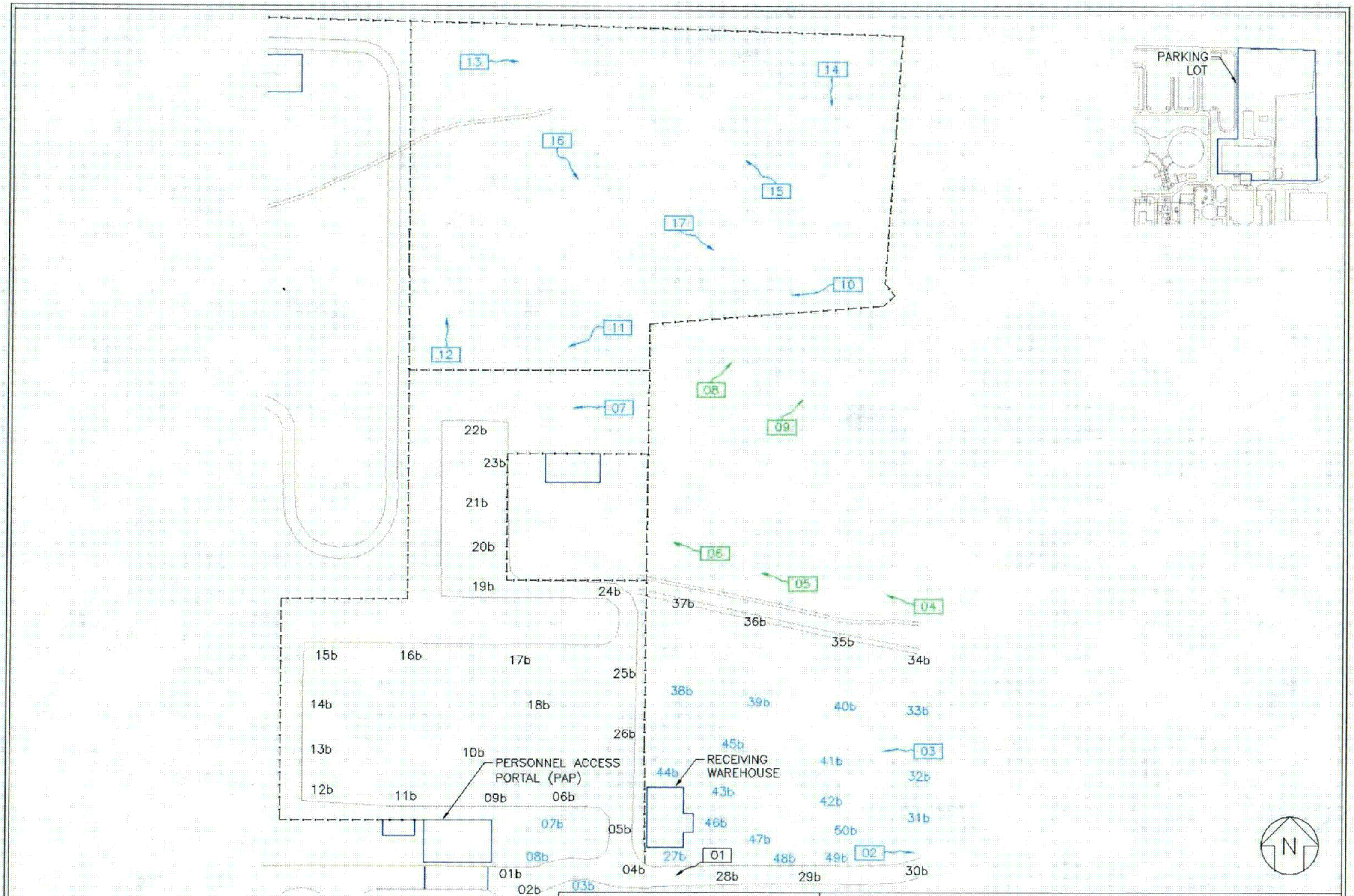

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Figure 2-44
Parking Lot/Lay Down Areas
501004

DWG: 501004

C40

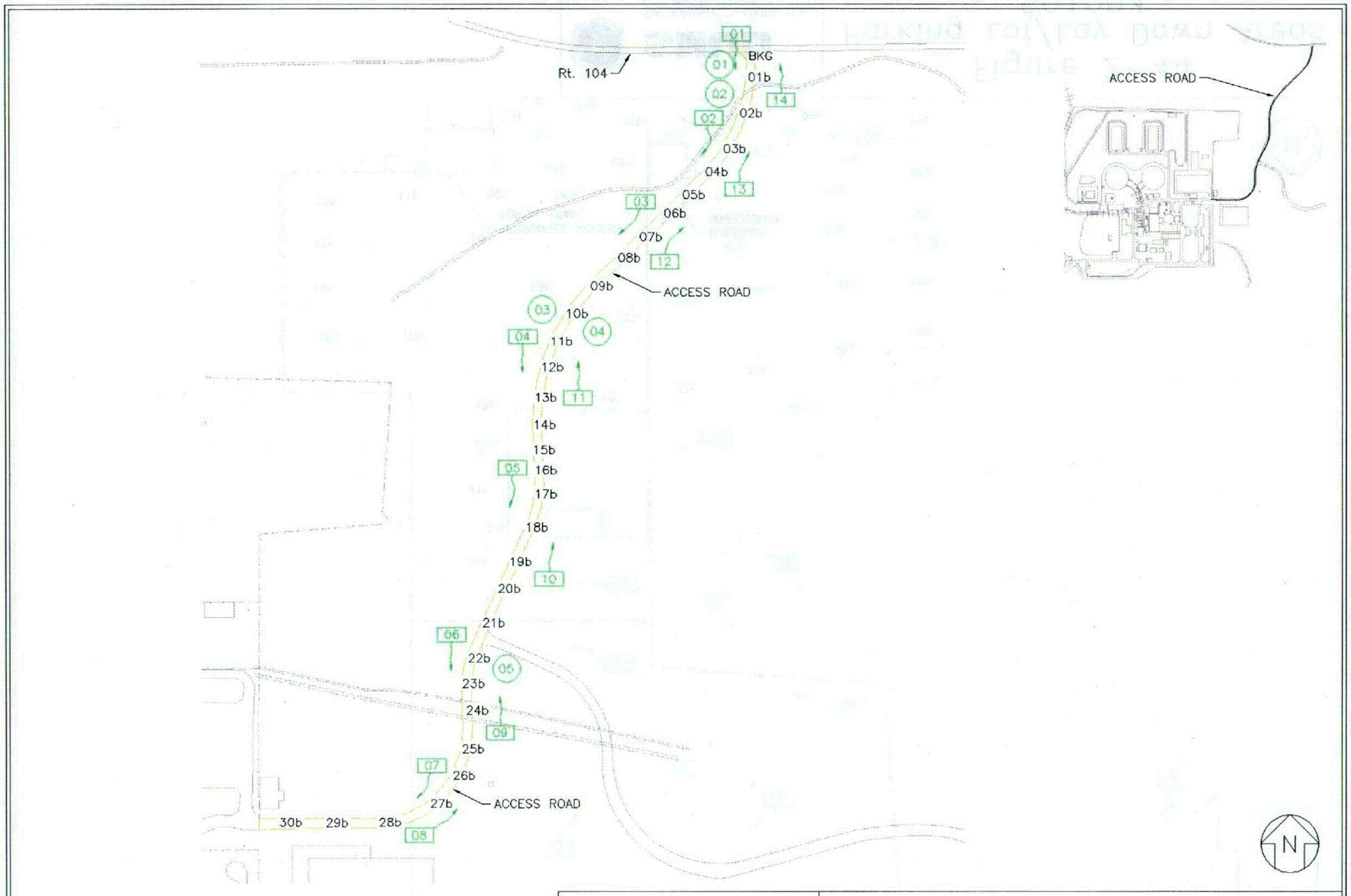

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Figure 2-45
Access Road
501005

DWG: 501005

C41

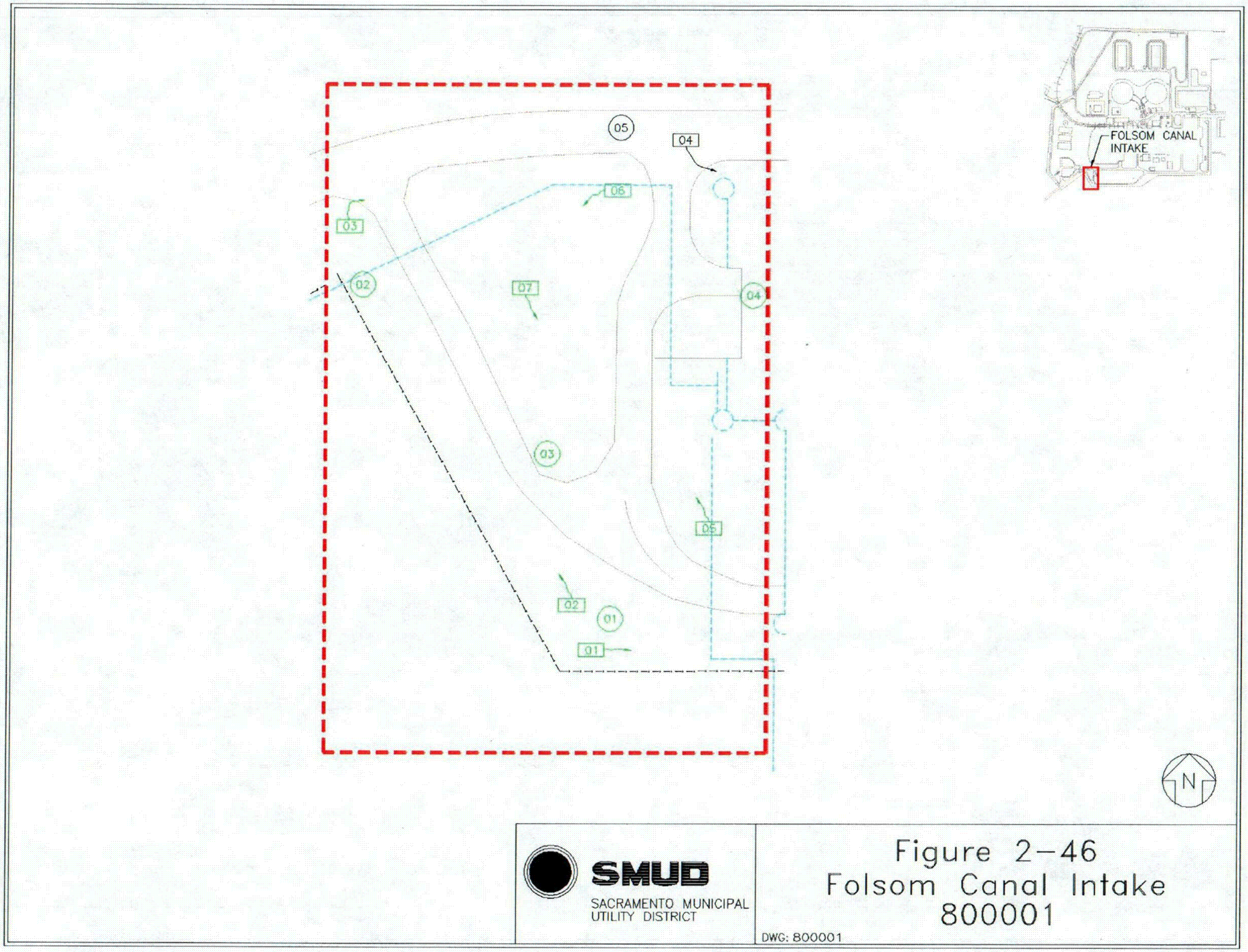

Figure 2-46
Folsom Canal Intake
800001

C42

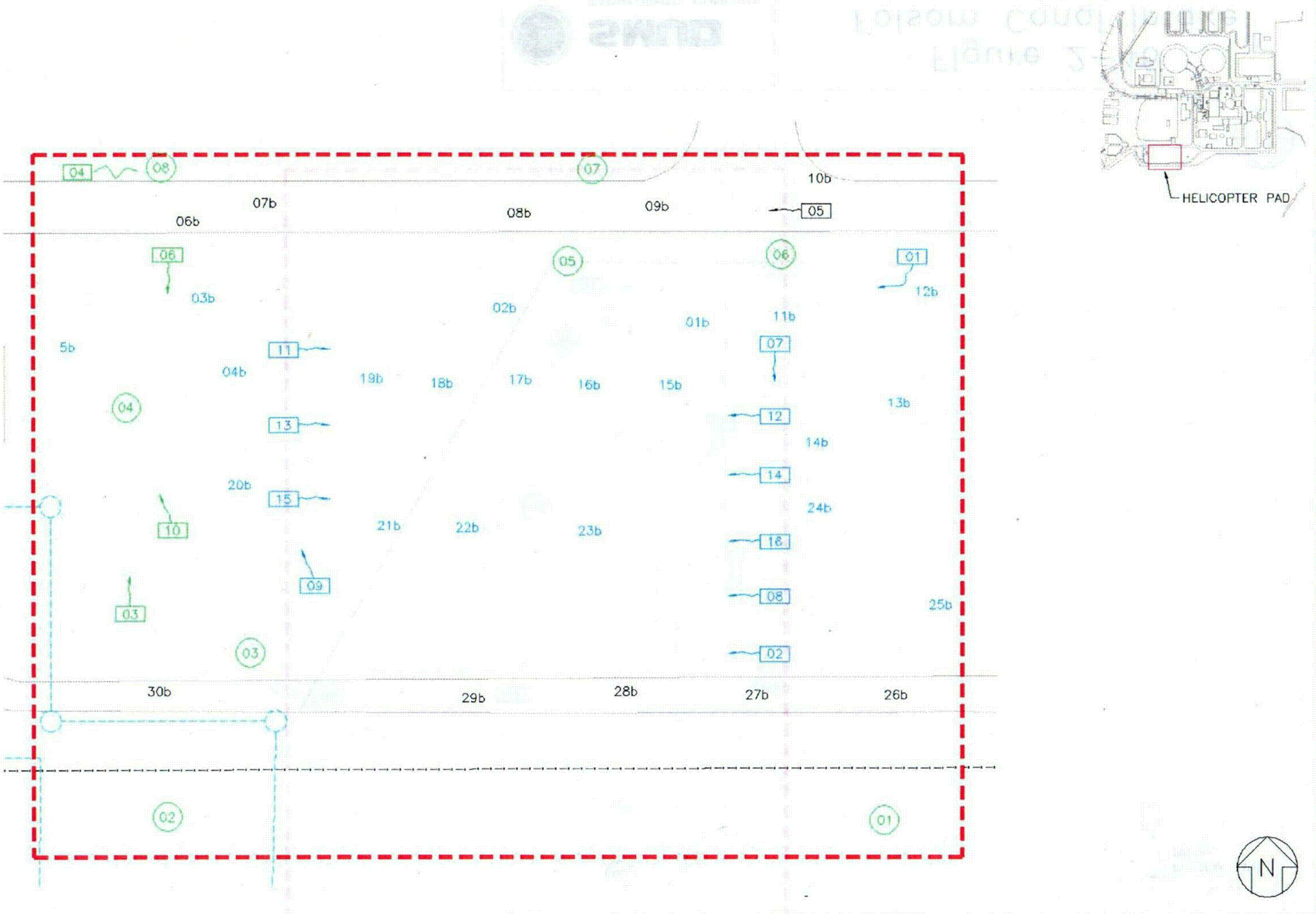

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Figure 2-47
Helicopter Pad
800002

DWG: 800002

C43

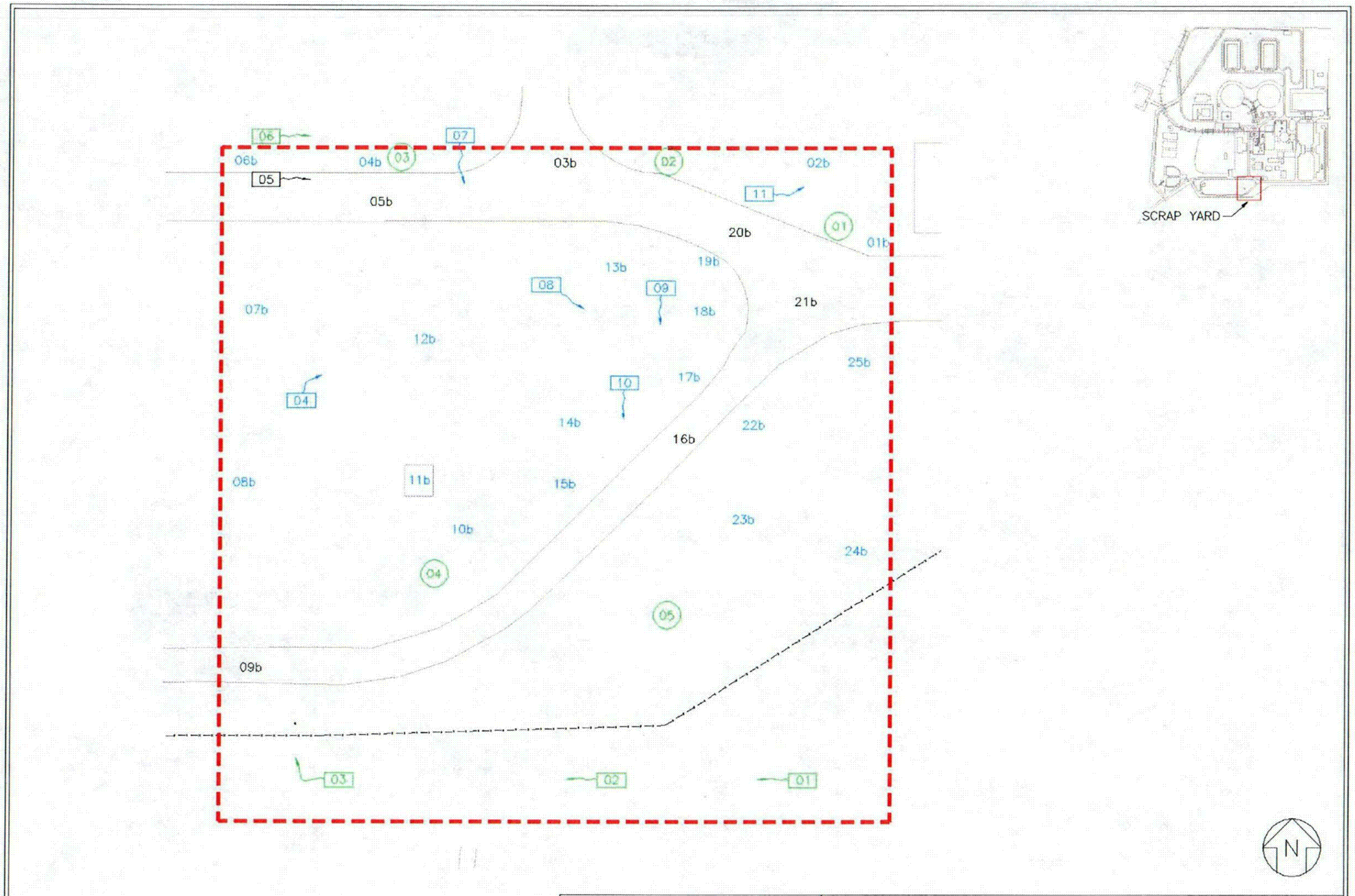

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Figure 2-48
South Scrap Yard
800003

DWG: 800003

C44

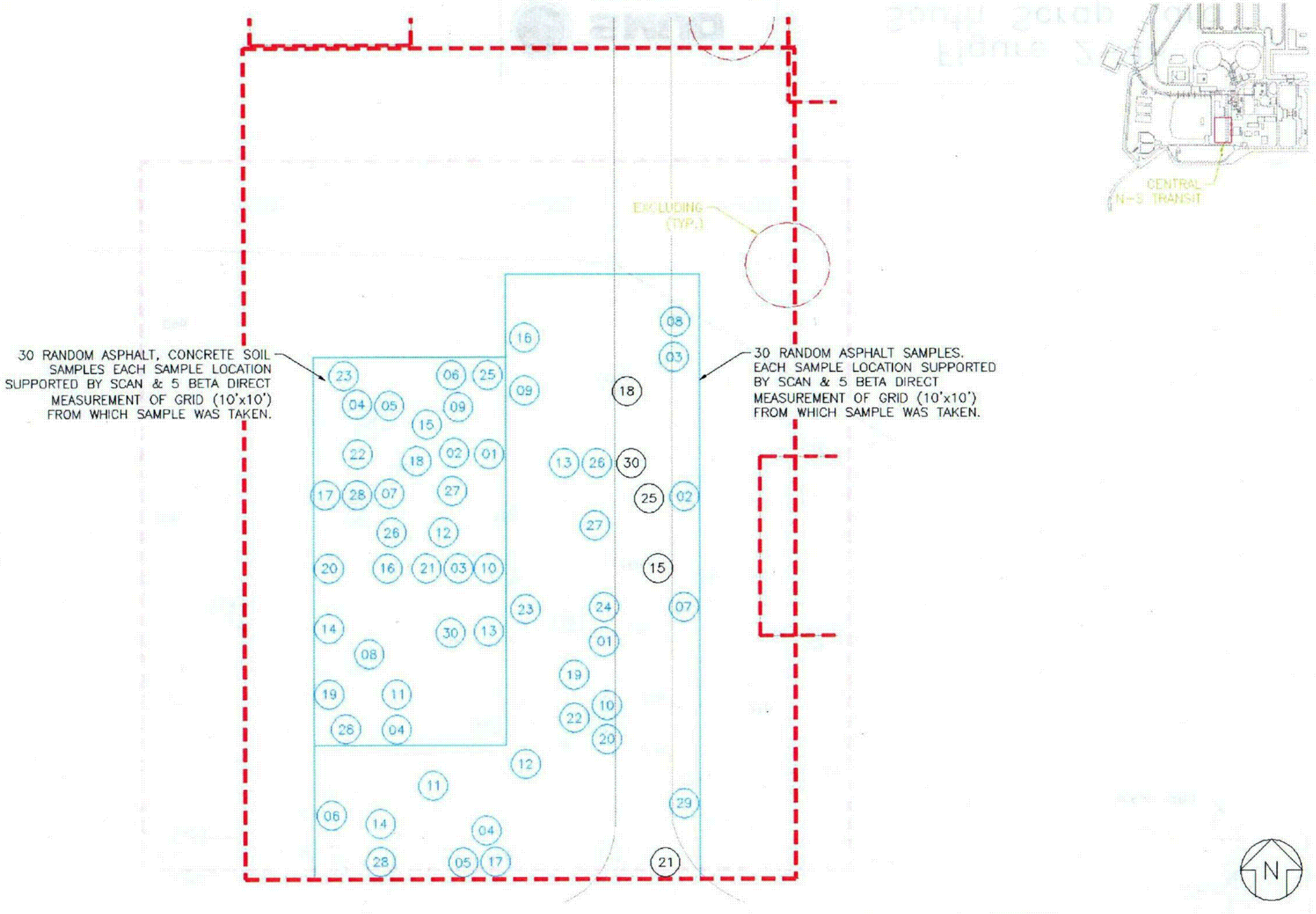

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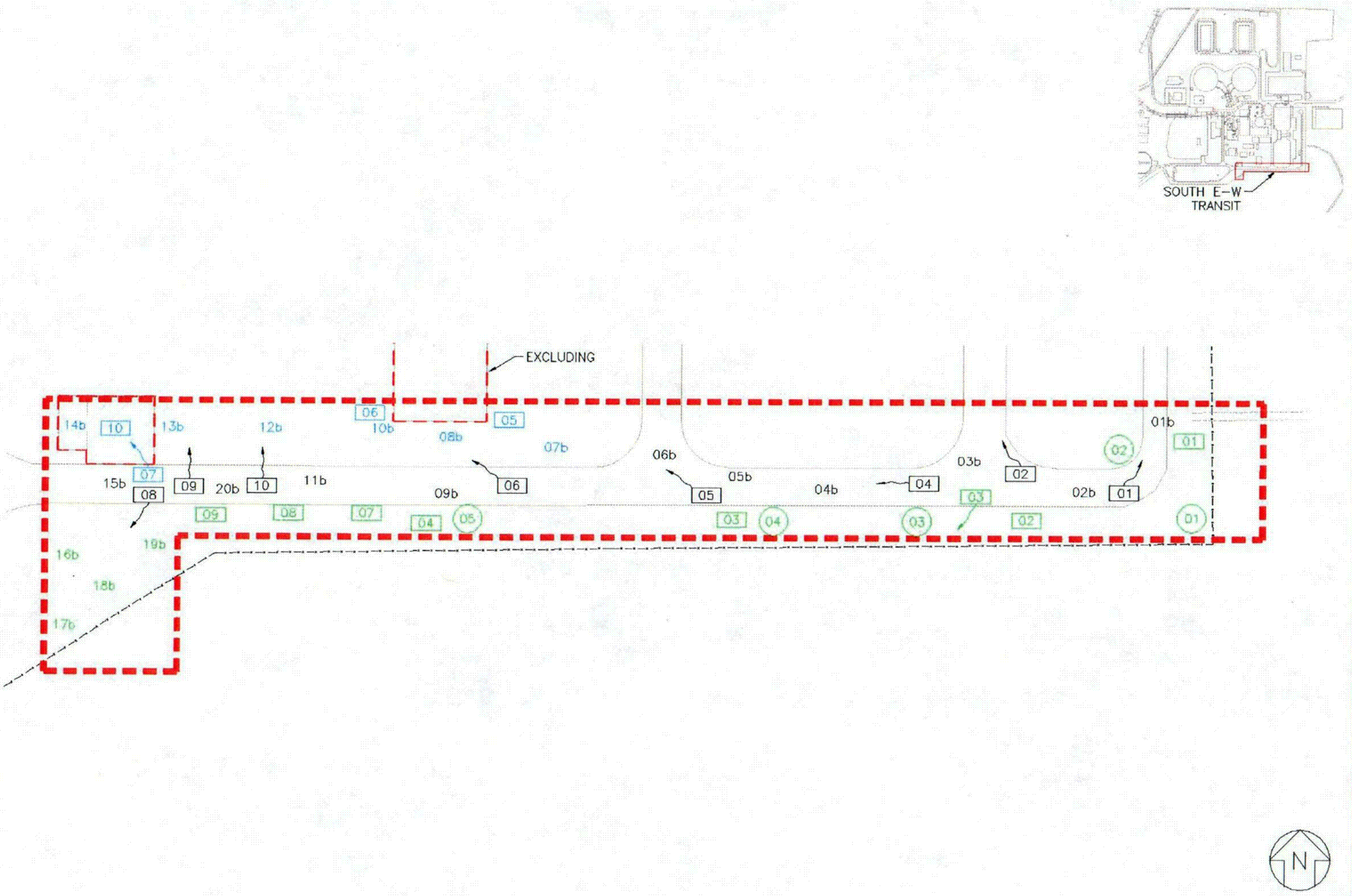
Figure 2-49
Central N-S Transit
800004

DWG: 800004

C45



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UTILITY DISTRICT

Figure 2-50
South E-W Transit
800005

DWG: 800005

C46

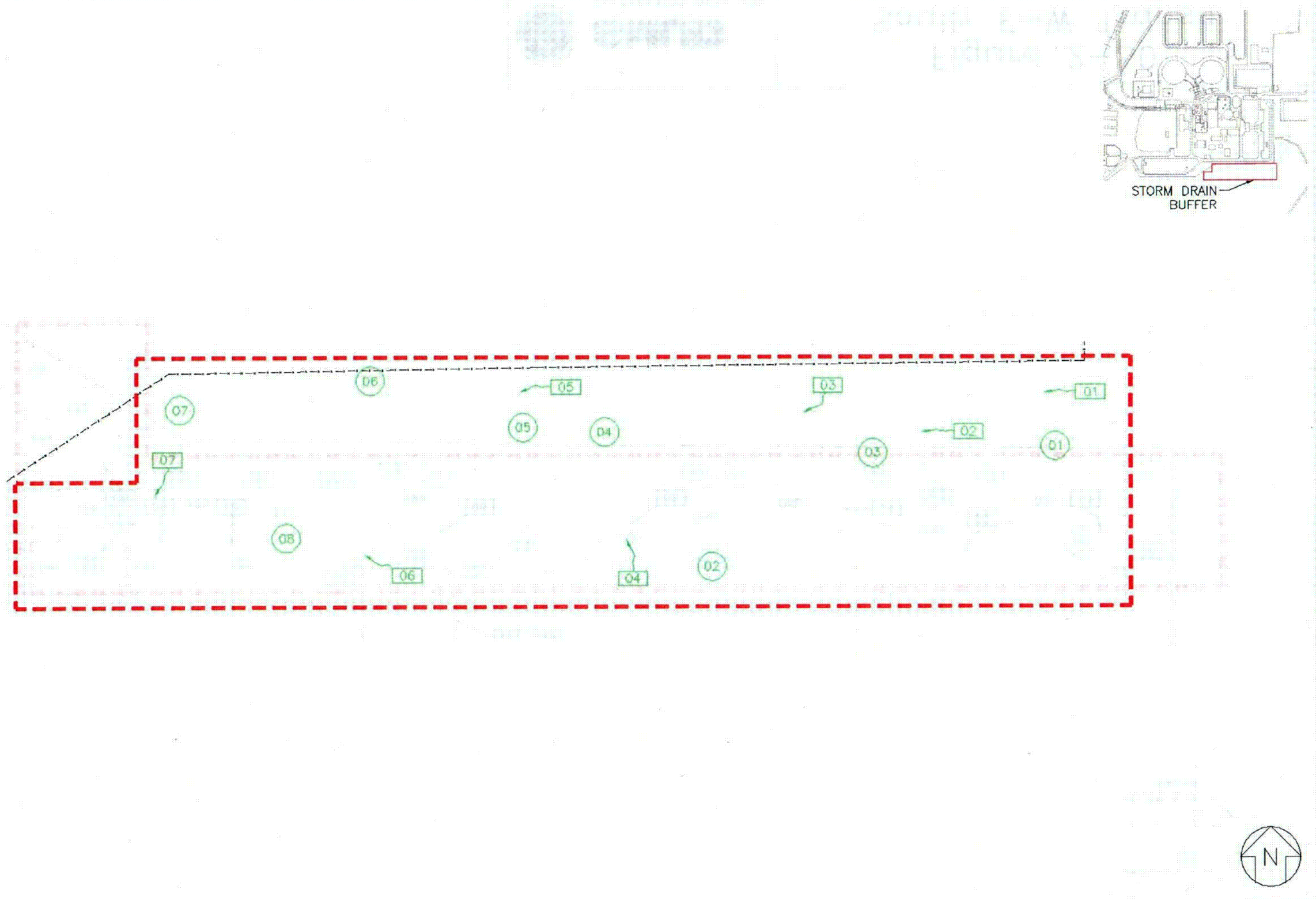

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Figure 2-50a
Storm Drain Buffer
800006

DWG: 800006

C47

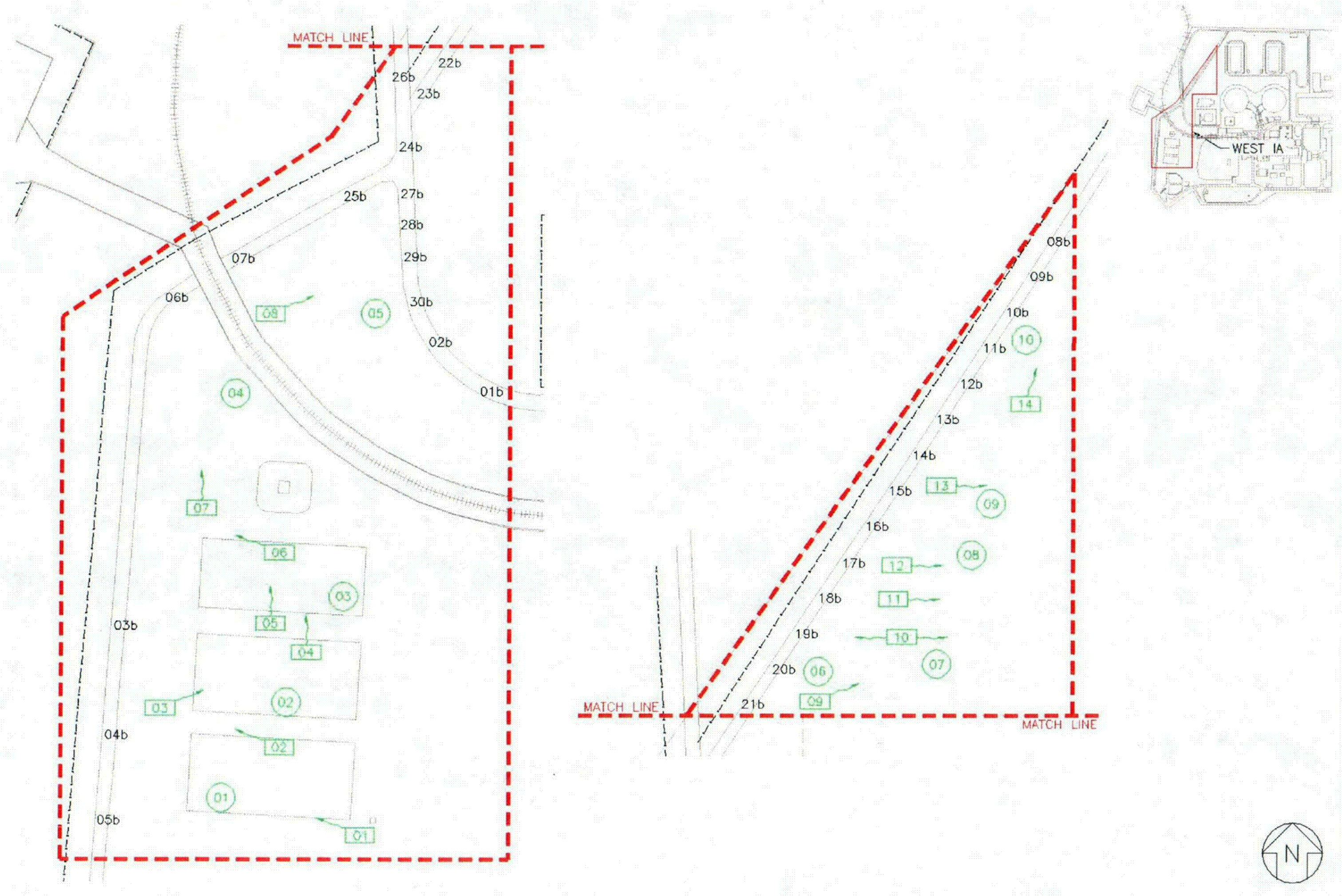

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Figure 2-51
West Industrial Area
800007

DWG: 800007

C48

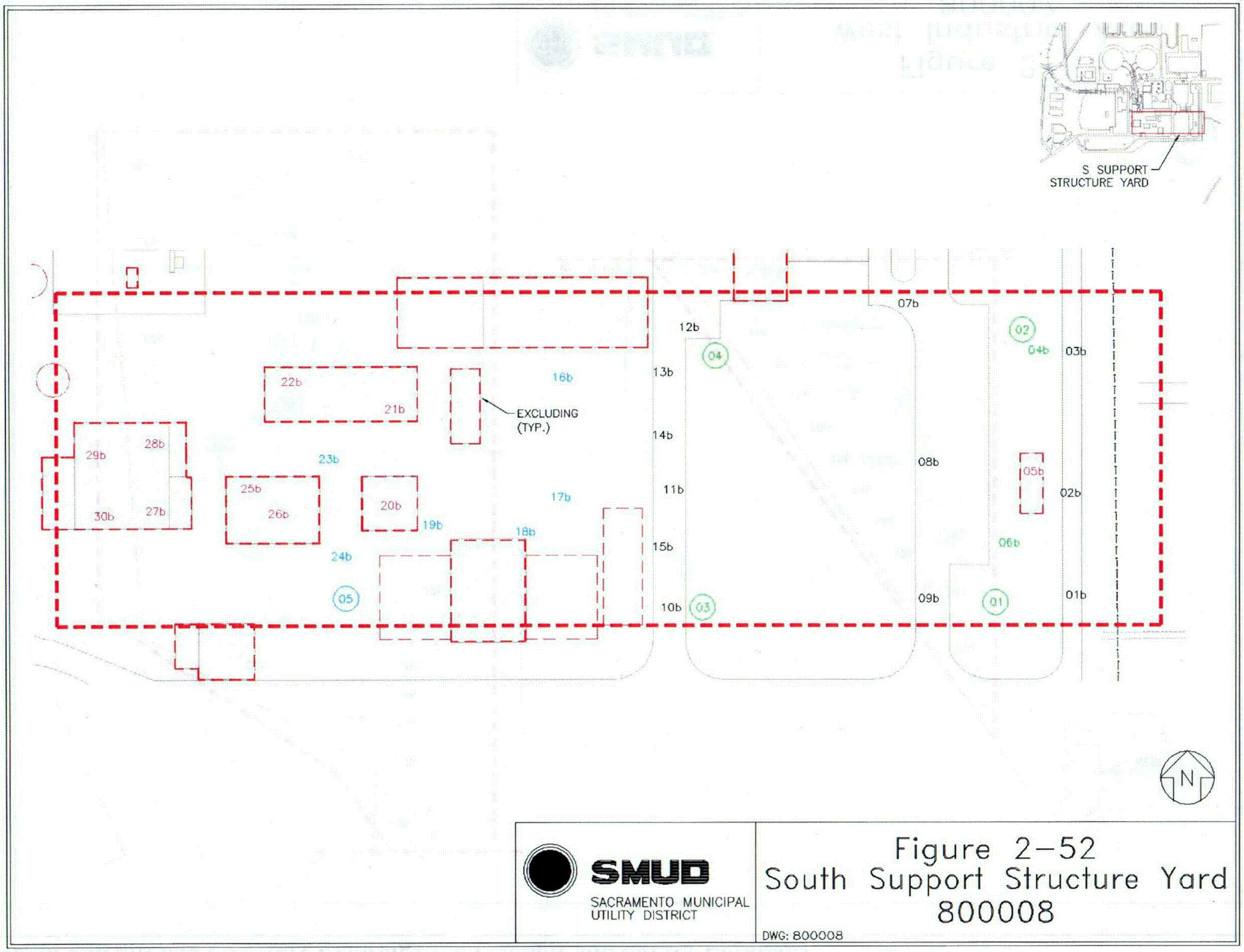

Figure 2-52
South Support Structure Yard
800008

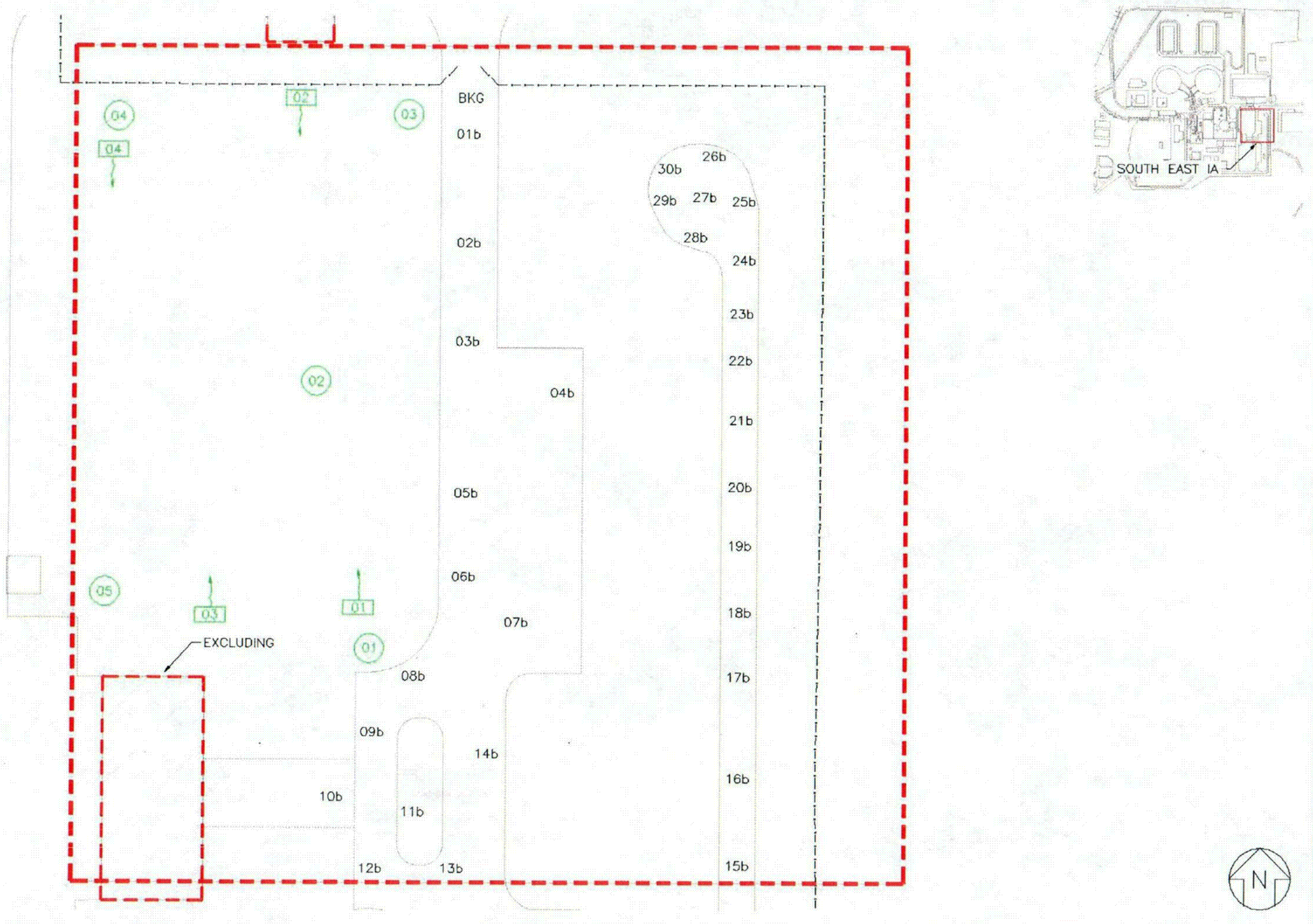

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Figure 2-53
South East IA
800009

DWG: 800009

C50

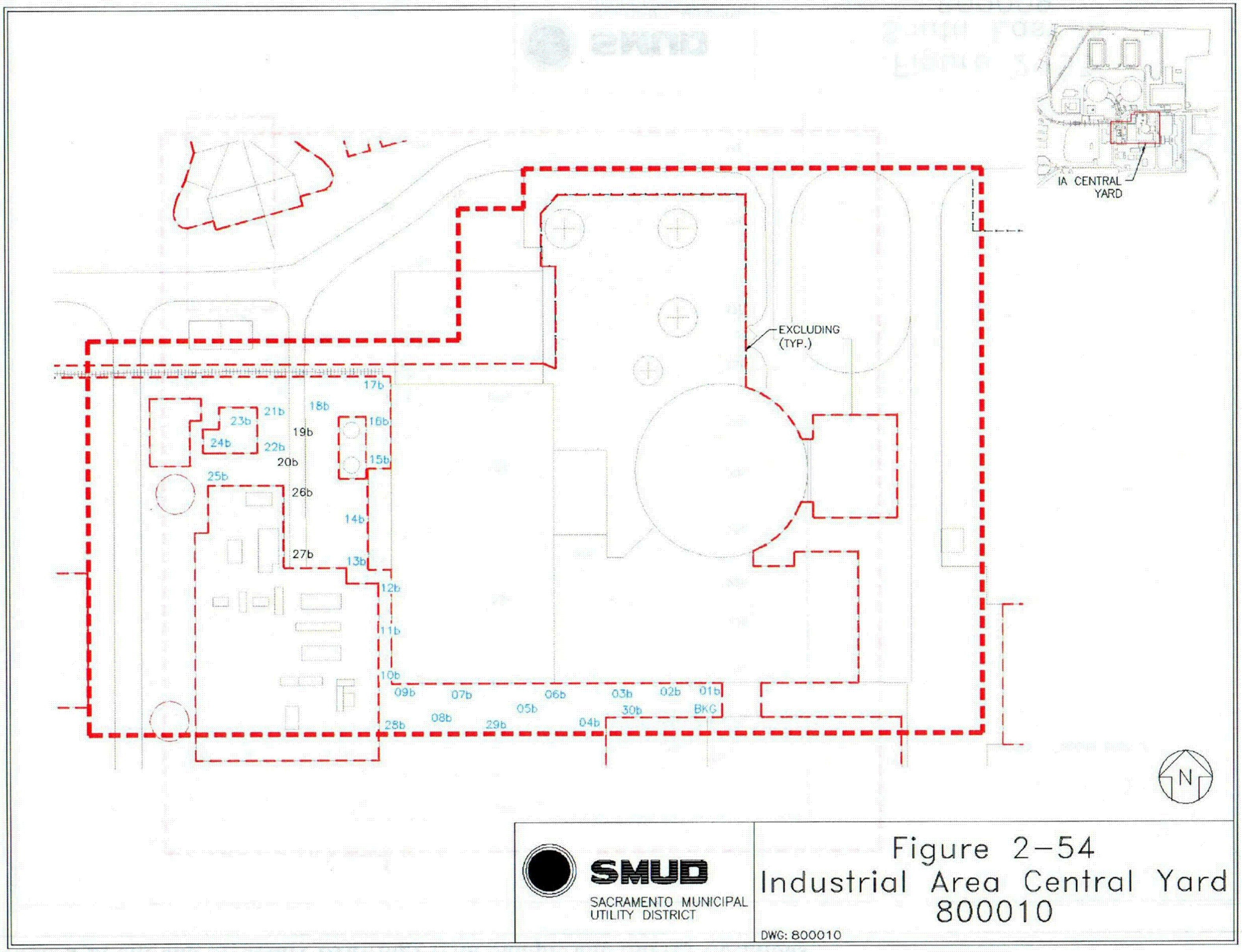
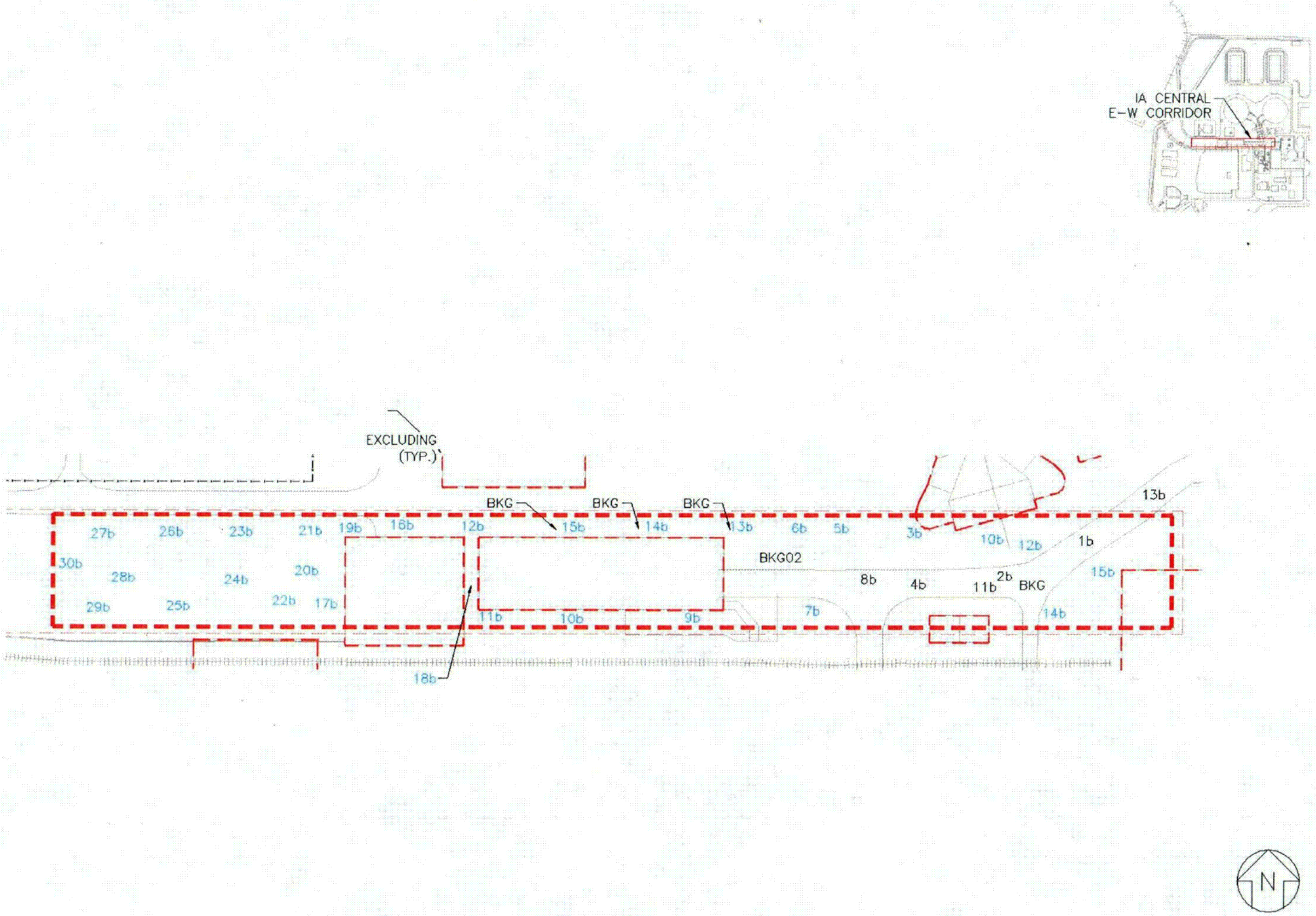


Figure 2-54
Industrial Area Central Yard
800010

C51



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UTILITY DISTRICT

Figure 2-55
Industrial Area E-W Corridor
800011

DWG: 800011

C52

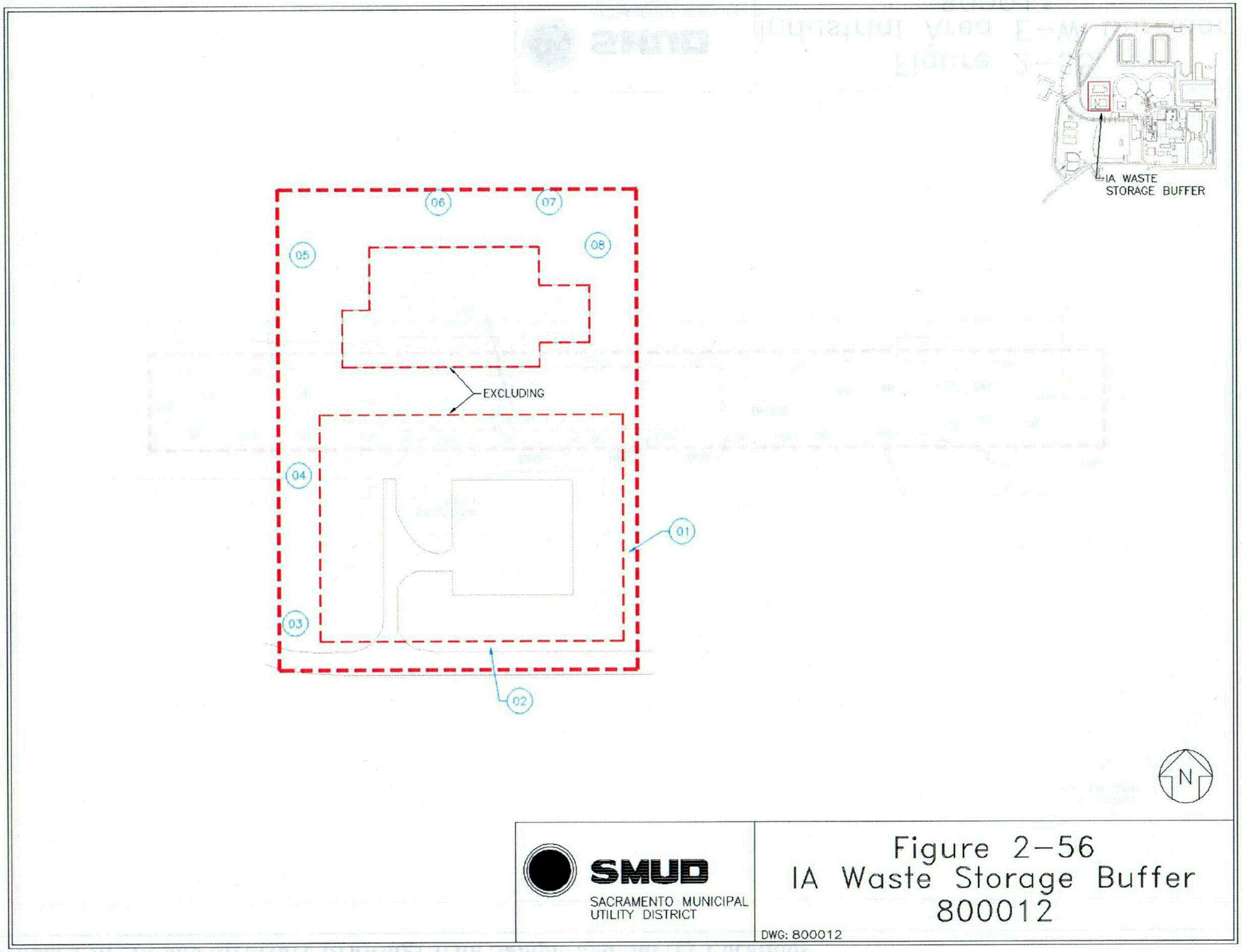
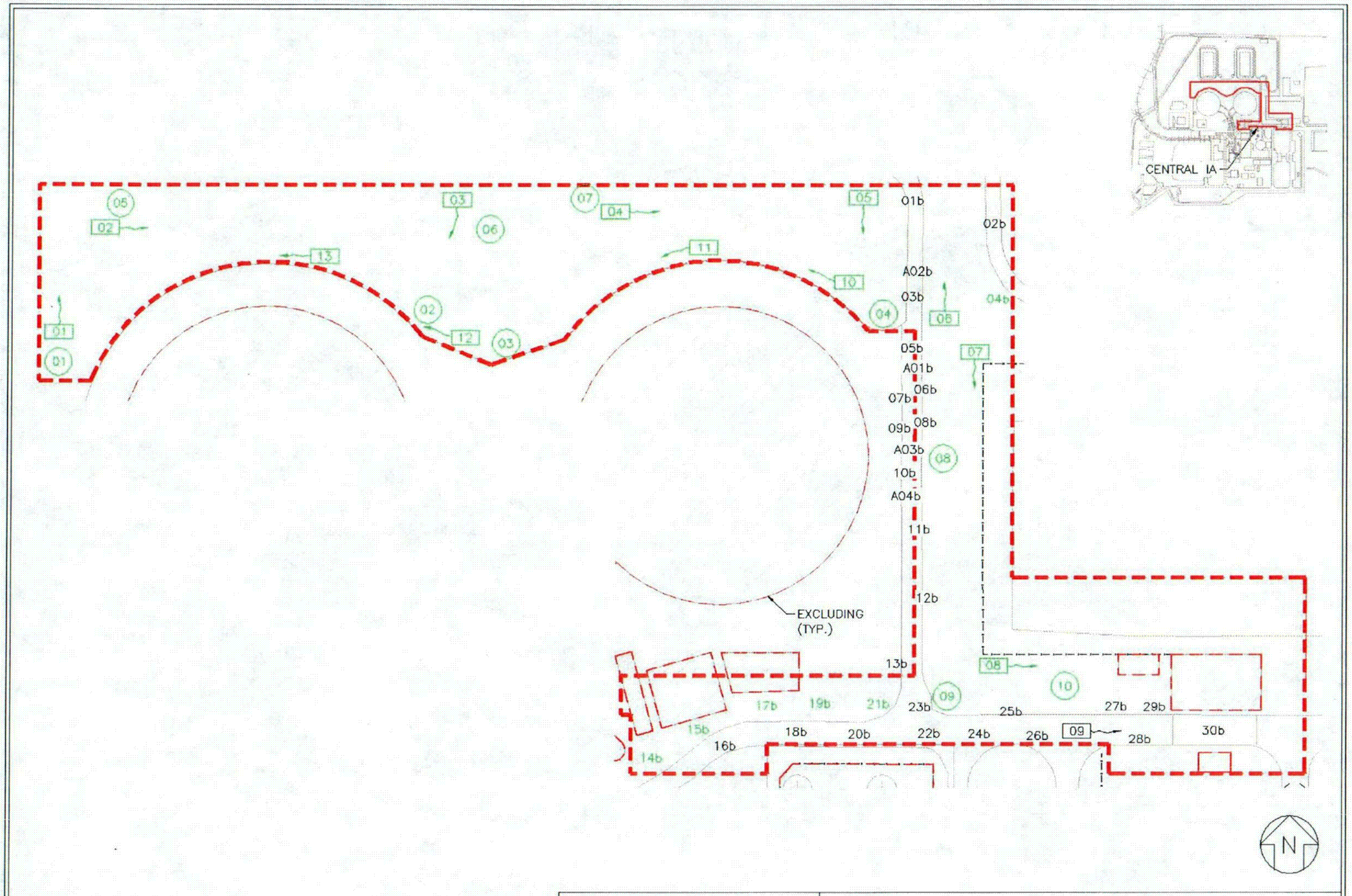


Figure 2-56
IA Waste Storage Buffer
800012

C53



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UTILITY DISTRICT

Figure 2-57
Central Industrial Area
800013

DWG: 800013

C54

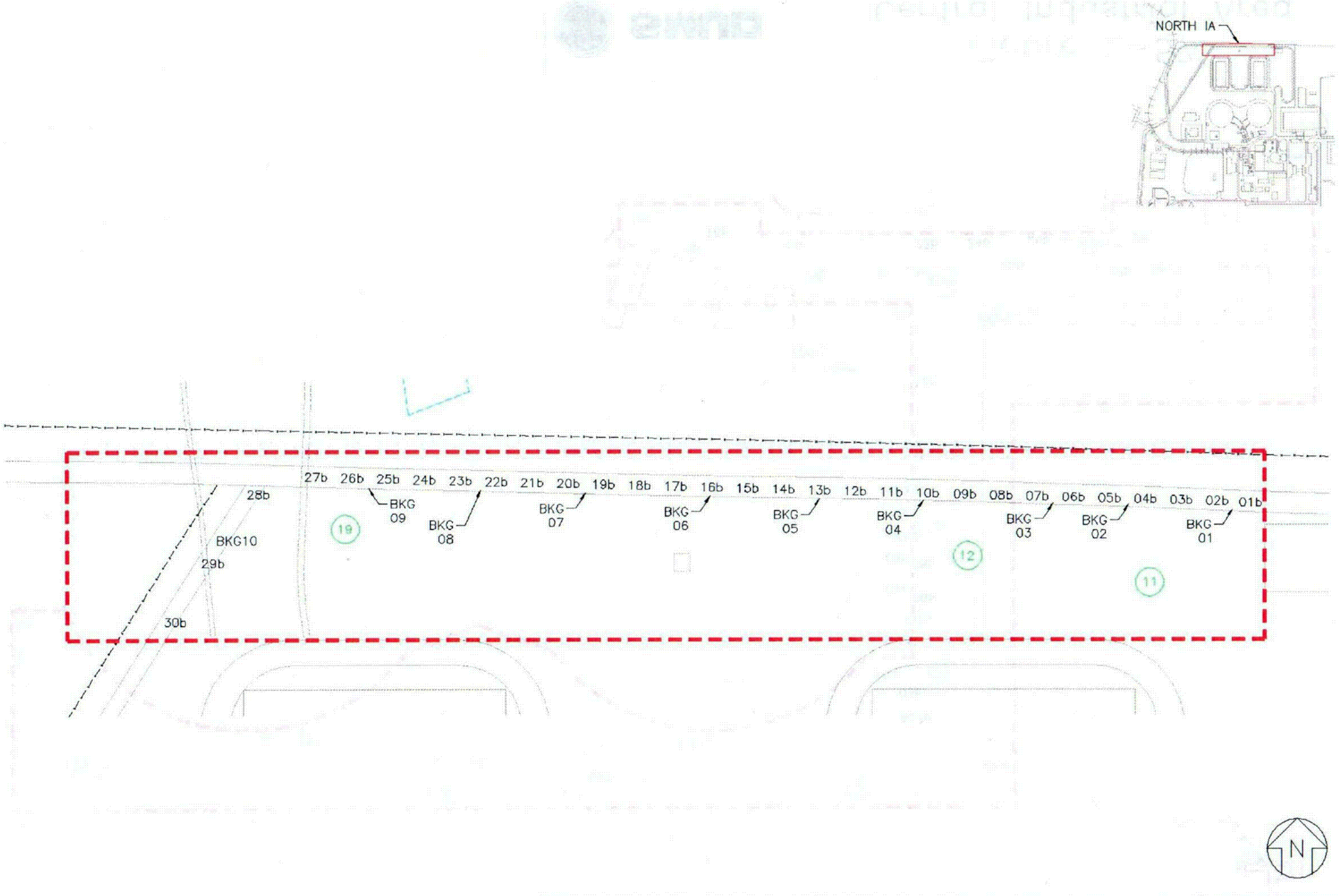

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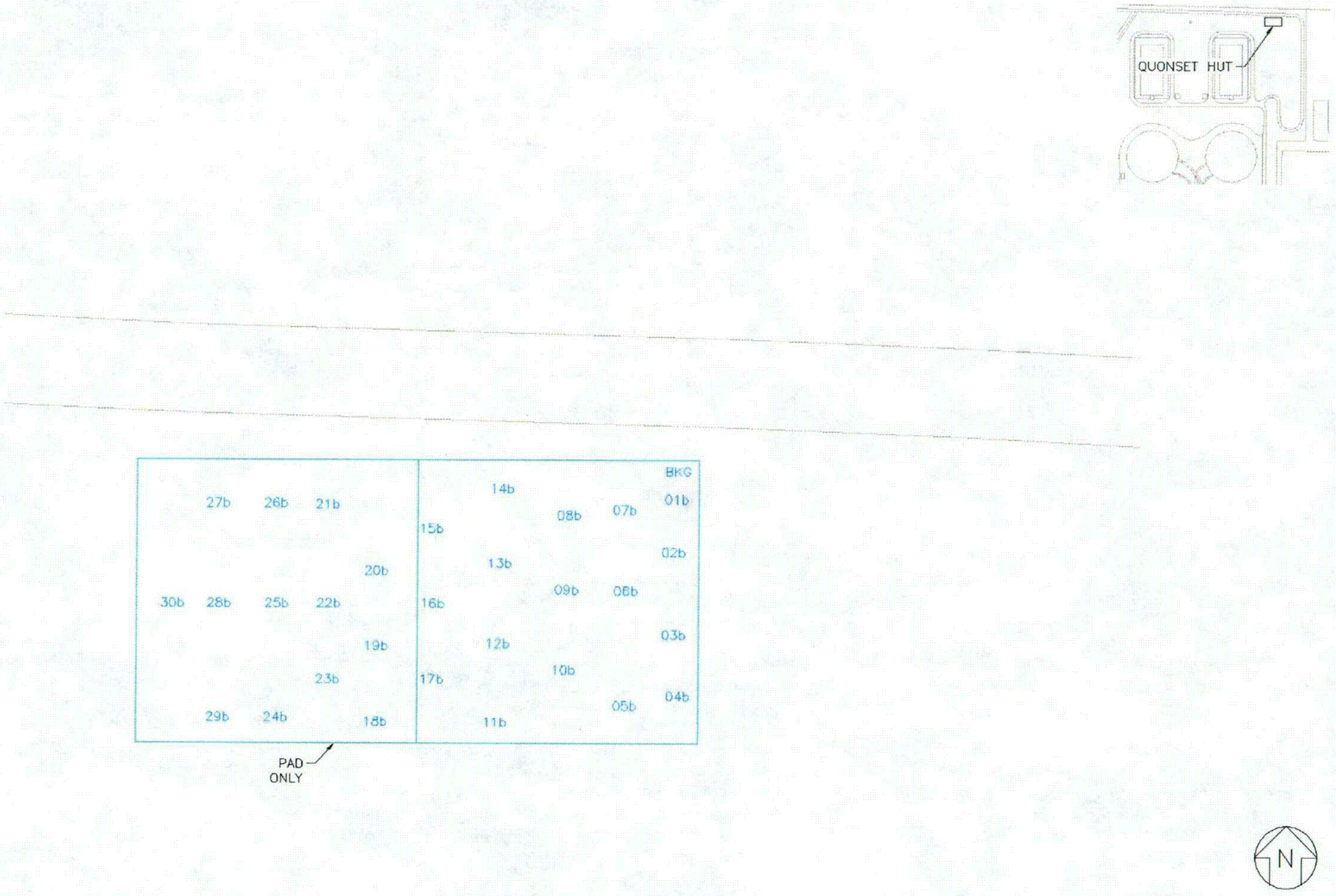
Figure 2-58
North Industrial Area
800014

DWG: 800014

C55



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UTILITY DISTRICT

Figure 2-59
Quonset Hut
803001

DWG: 803001.01

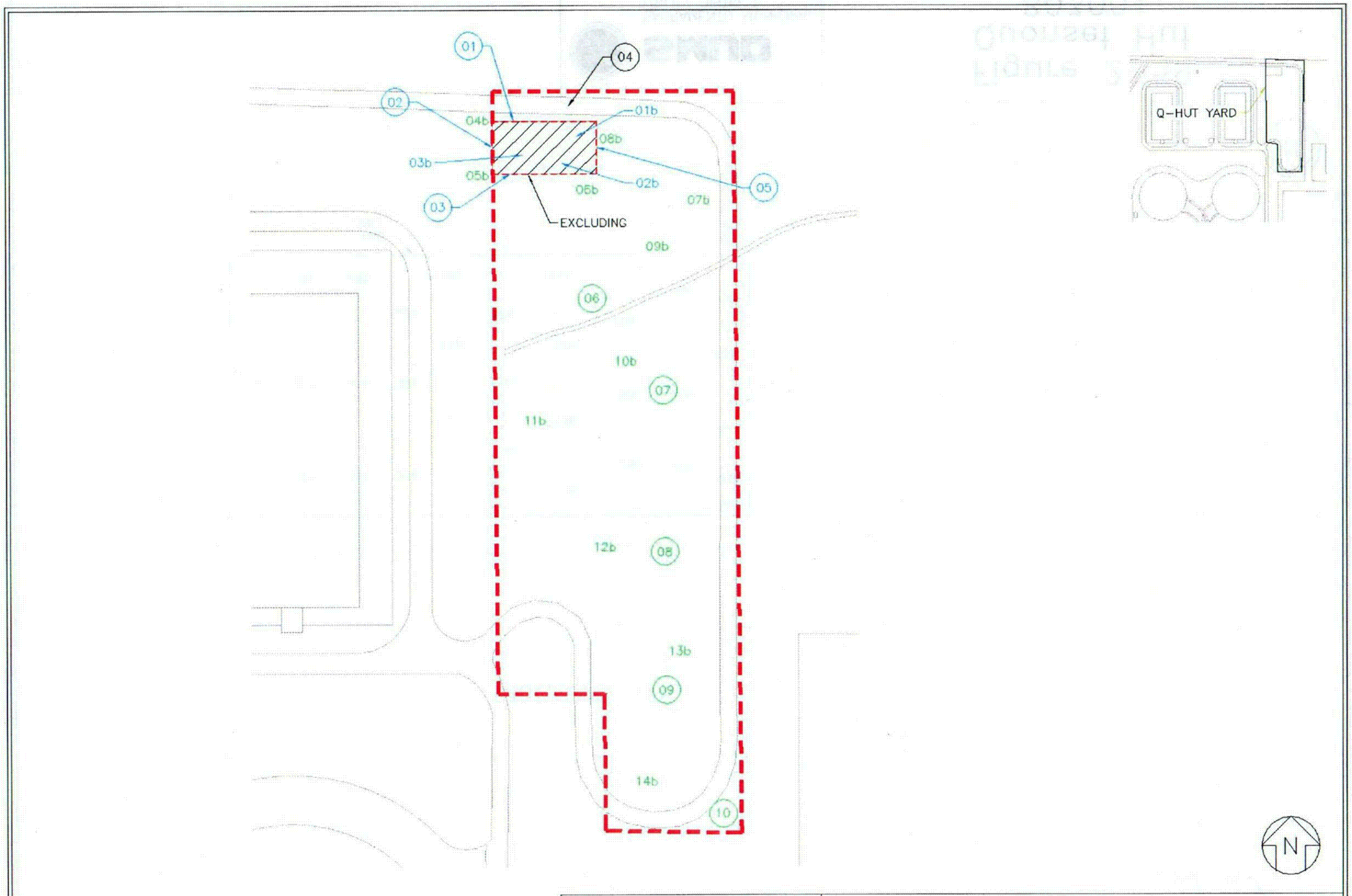

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Figure 2-60
Q-Hut Scrap Yard Area
803002

DWG: 803002.01

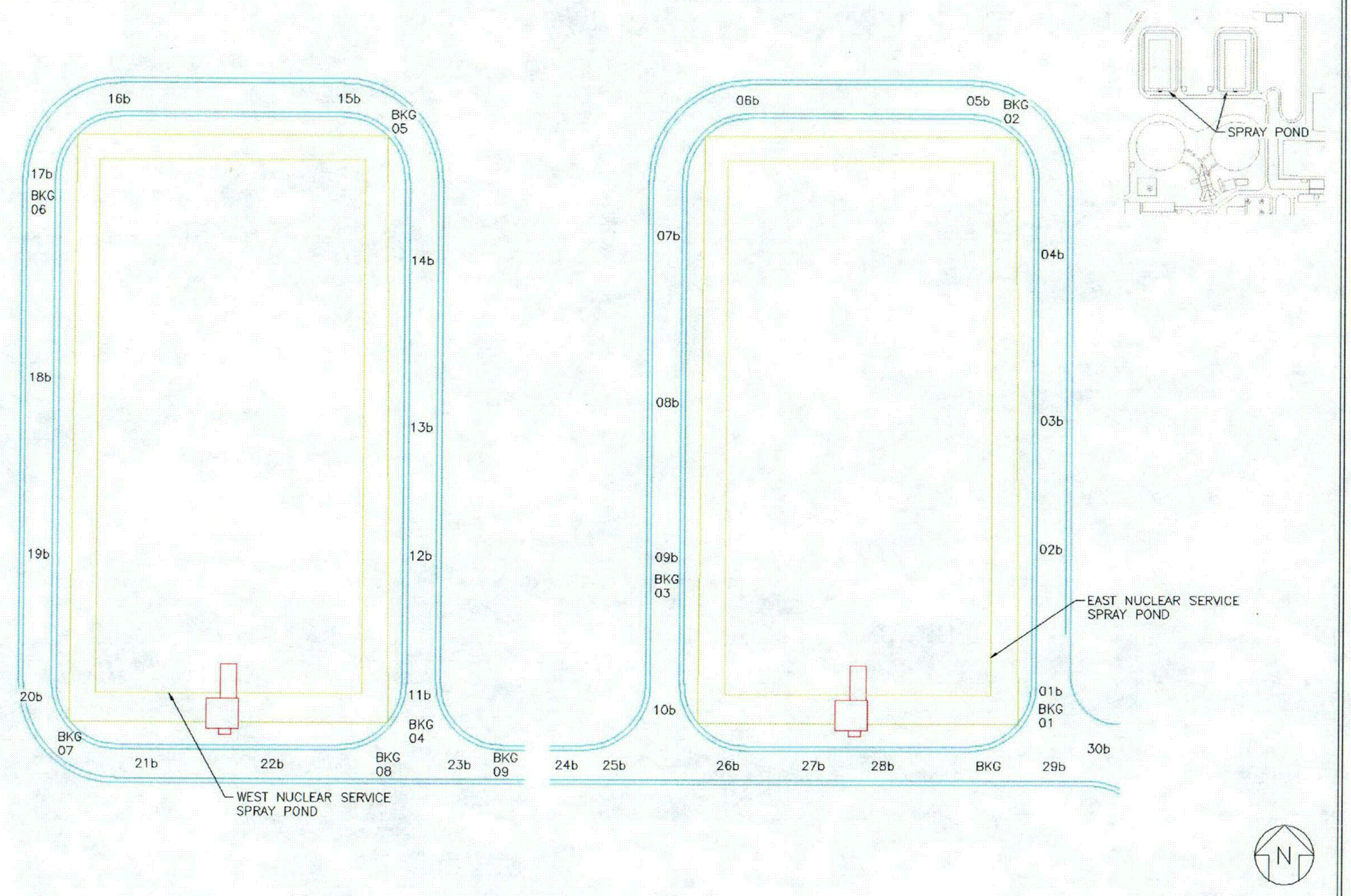

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Figure 2-61
West & East Spray Ponds
806001 & 2

DWG: 806001.01c

C58

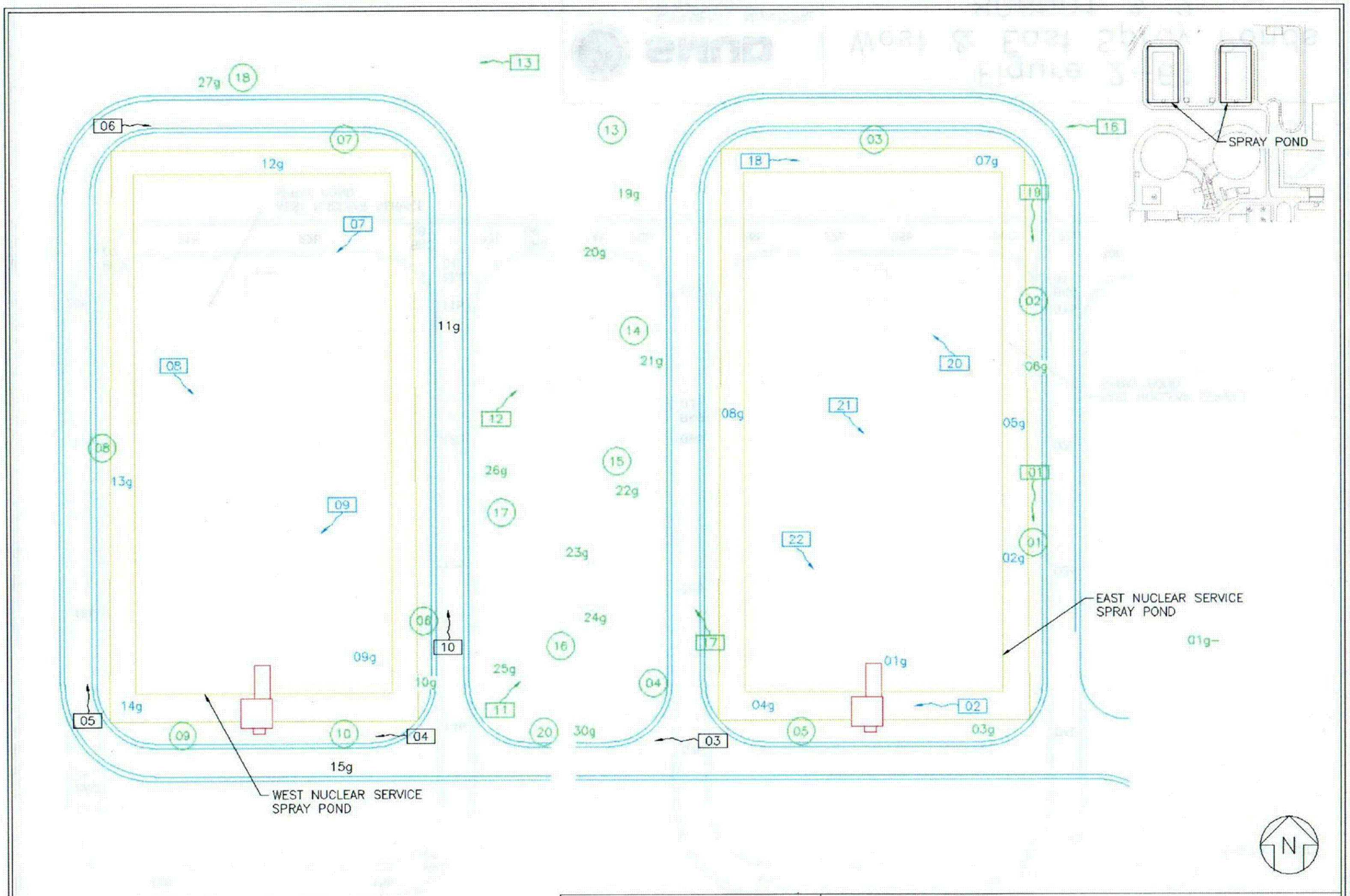

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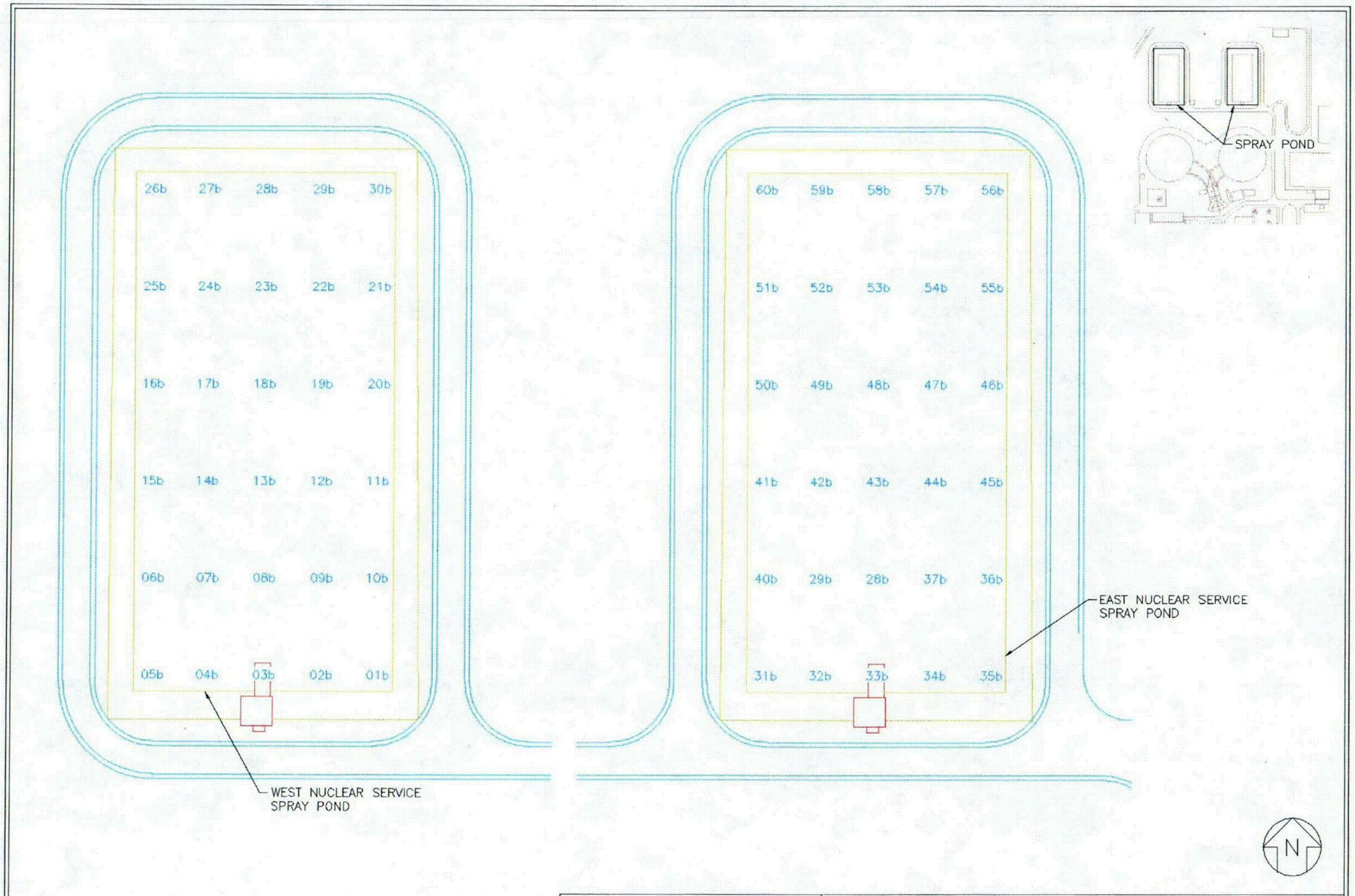
Figure 2-62
West & East Spray Ponds
806001 & 2

DWG: 806001.01b

C59



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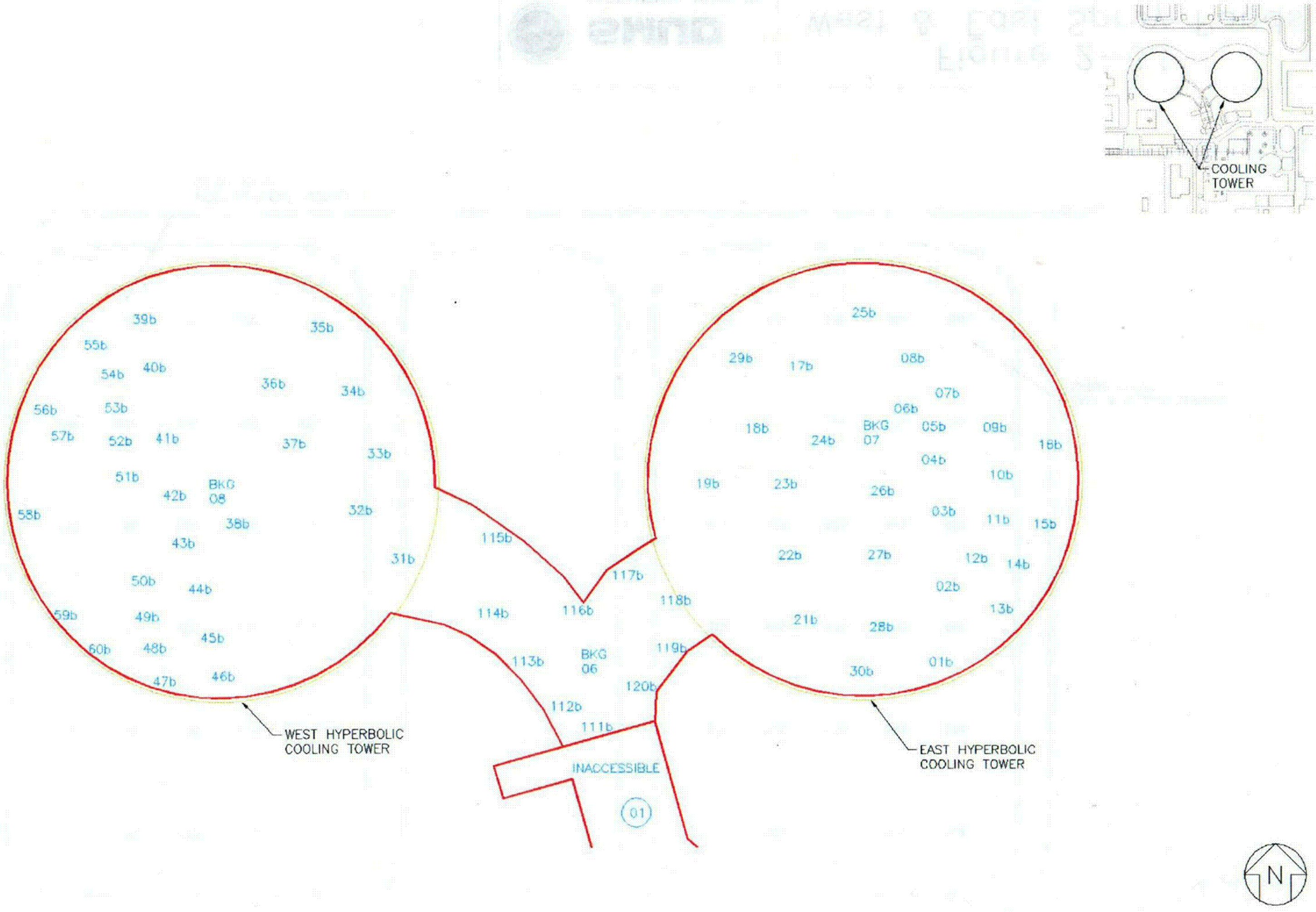
SACRAMENTO MUNICIPAL
UTILITY DISTRICT

Figure 2-63
West & East Spray Ponds
806001 & 2

DWG: 806001.01a

C60



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SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-64
West & East Cooling Towers
808001 & 2

DWG: 808001.01

C61

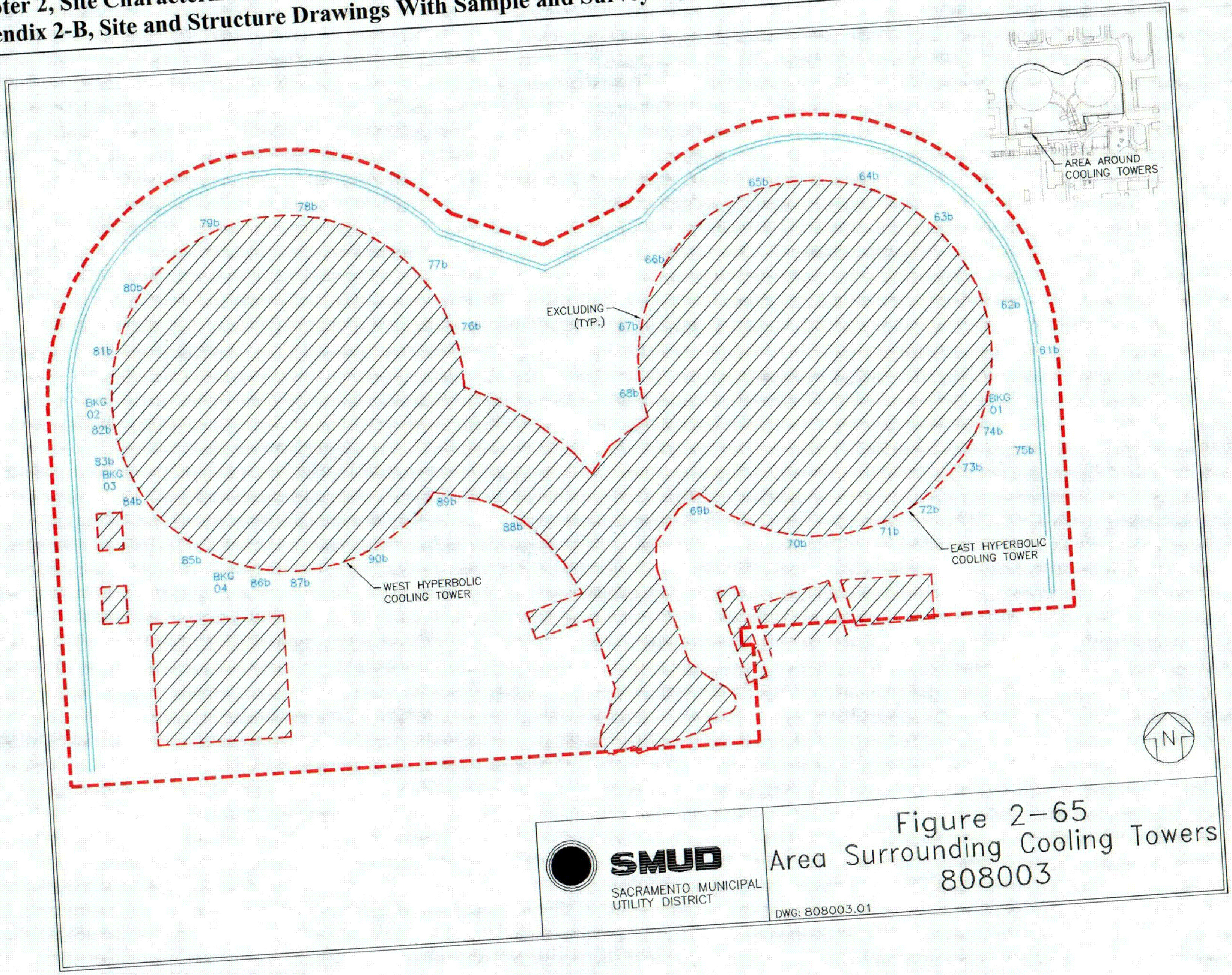
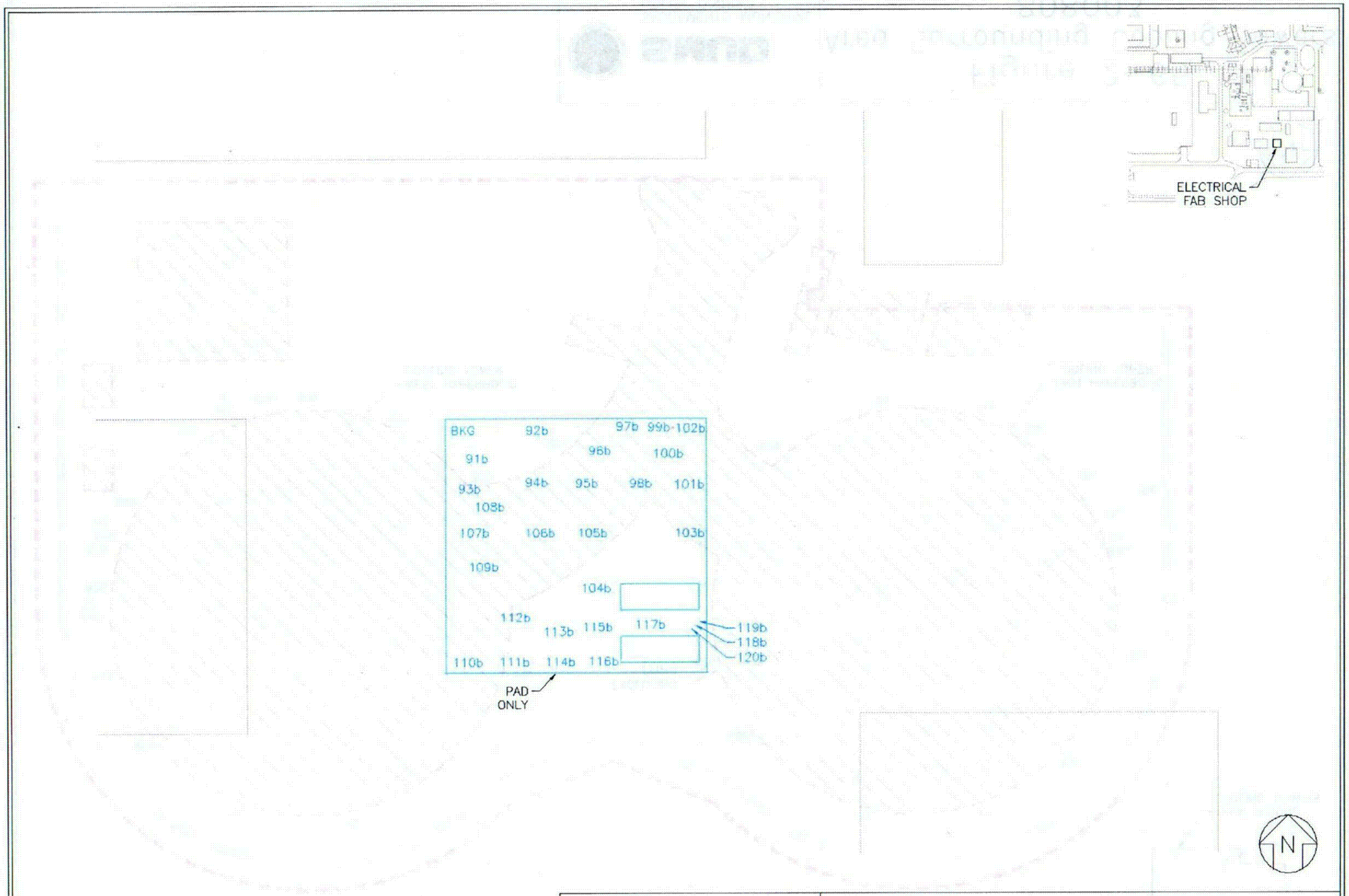


Figure 2-65
Area Surrounding Cooling Towers
808003

C62



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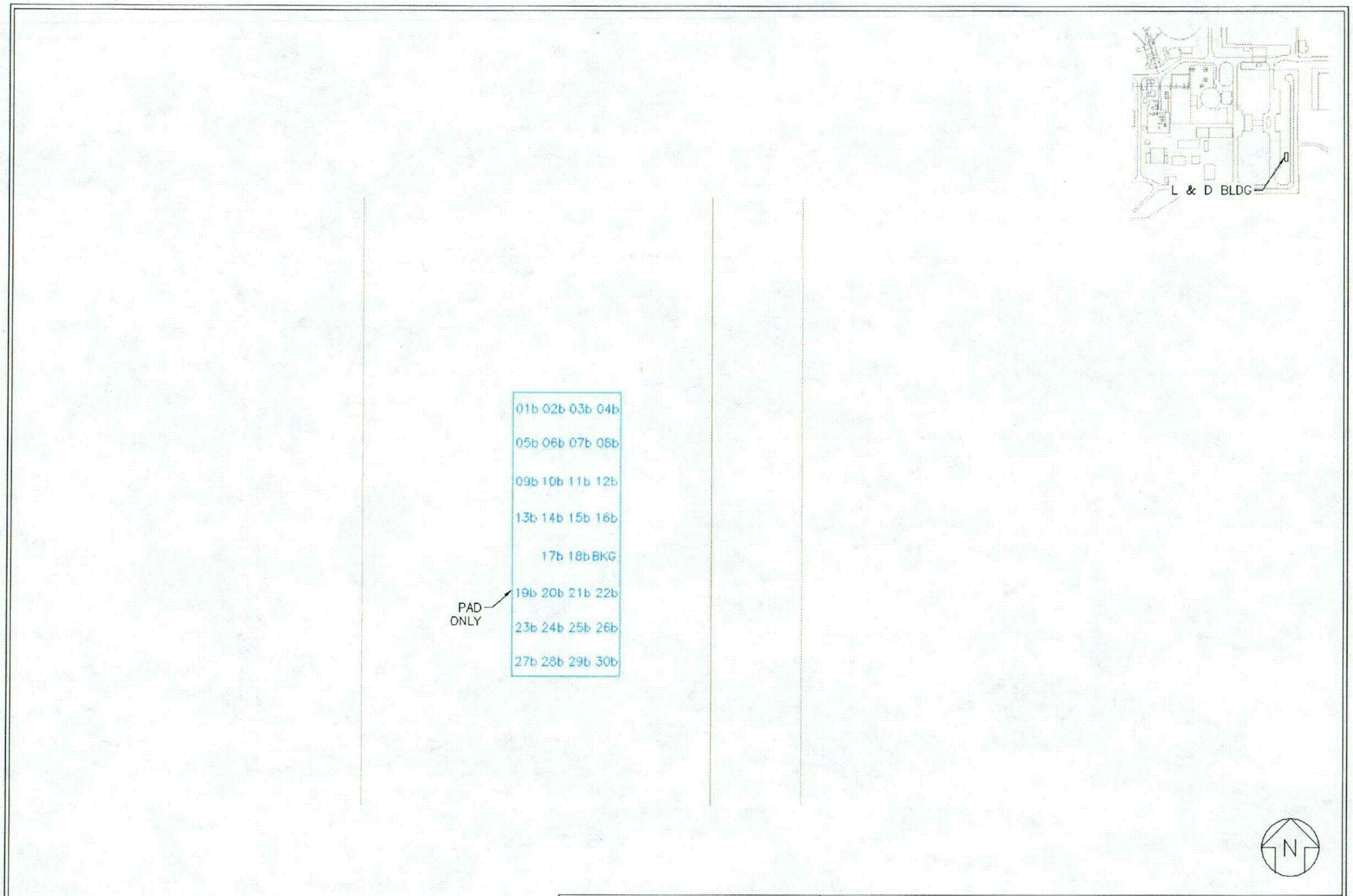
UTILITY DISTRICT

Figure 2-66
Electrical Fab Shop
818000

DWG: 818000.01

C63



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SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-67
L & D Building
820000

DWG: 820000.01

C64

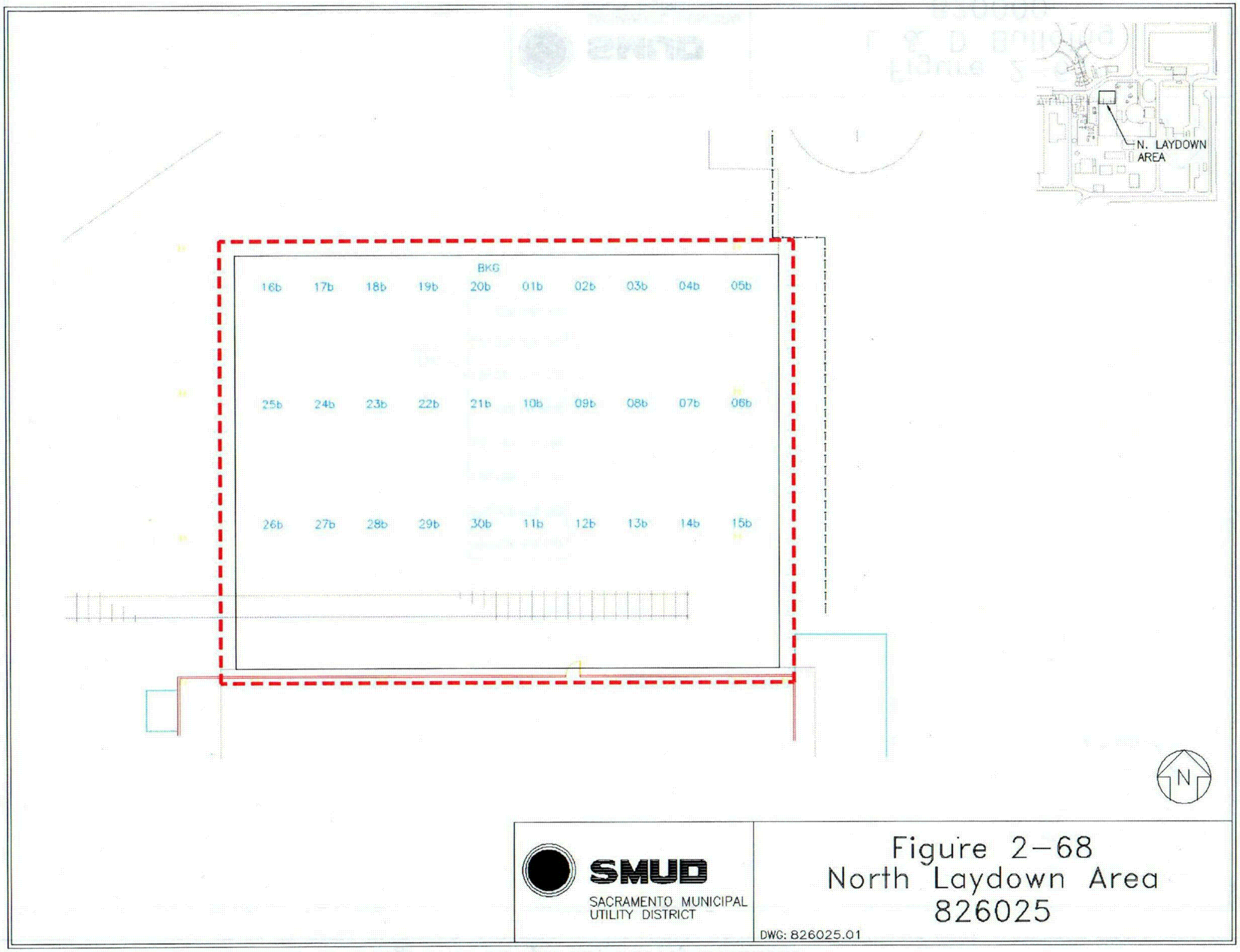

Figure 2-68
North Laydown Area
826025

CG5

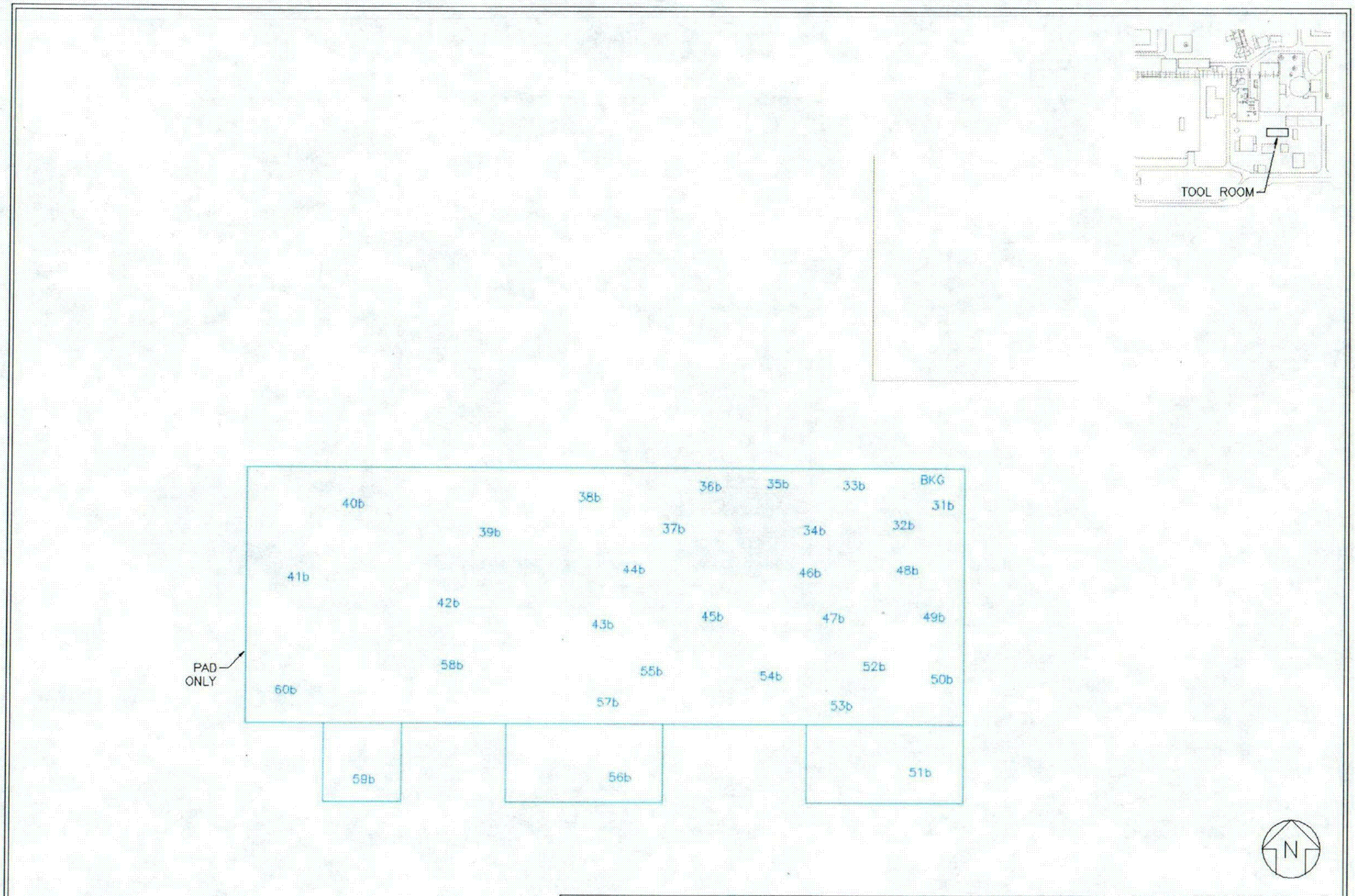

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Figure 2-69
Tool Room
827000

DWG: 827000.01

C66

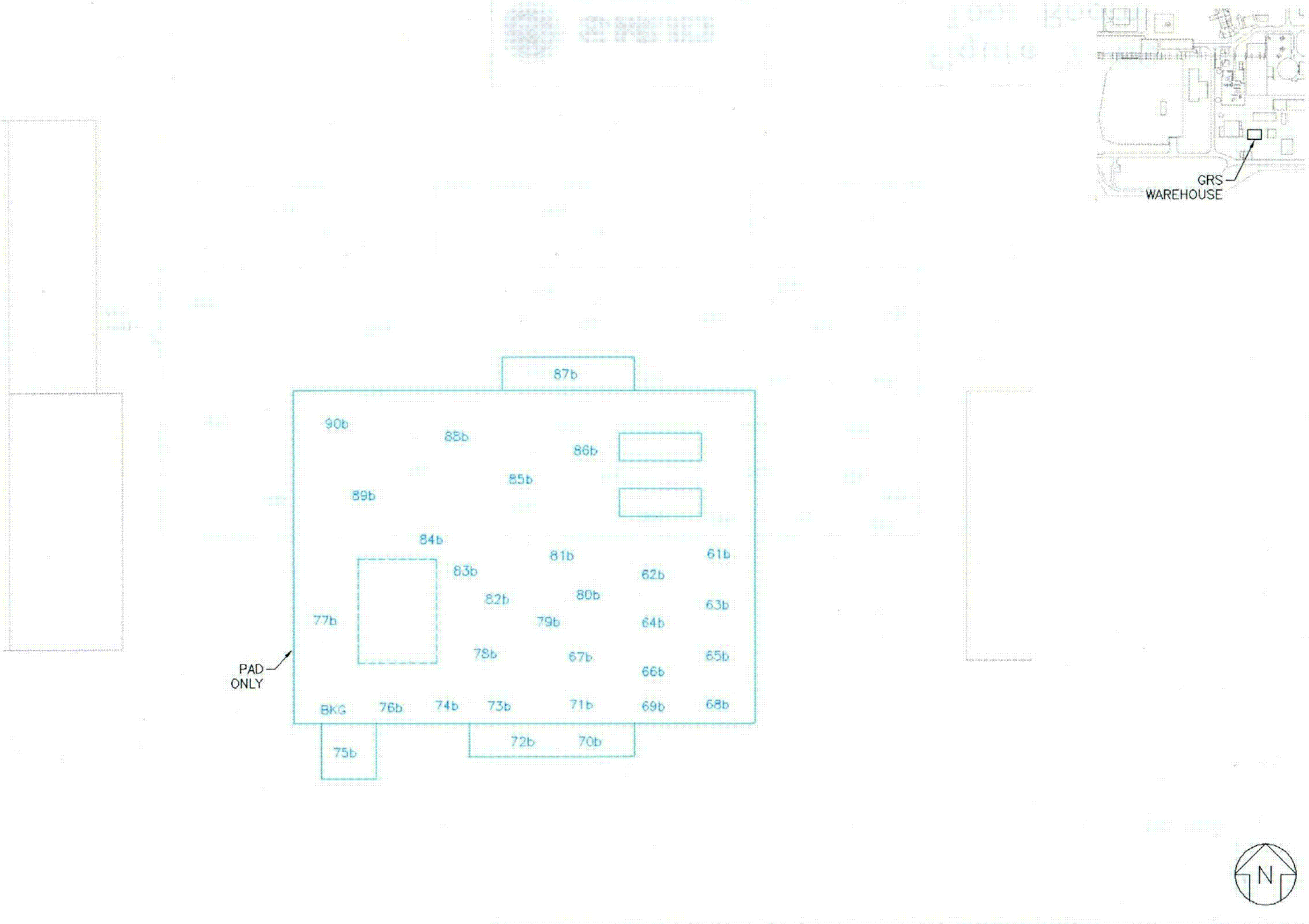

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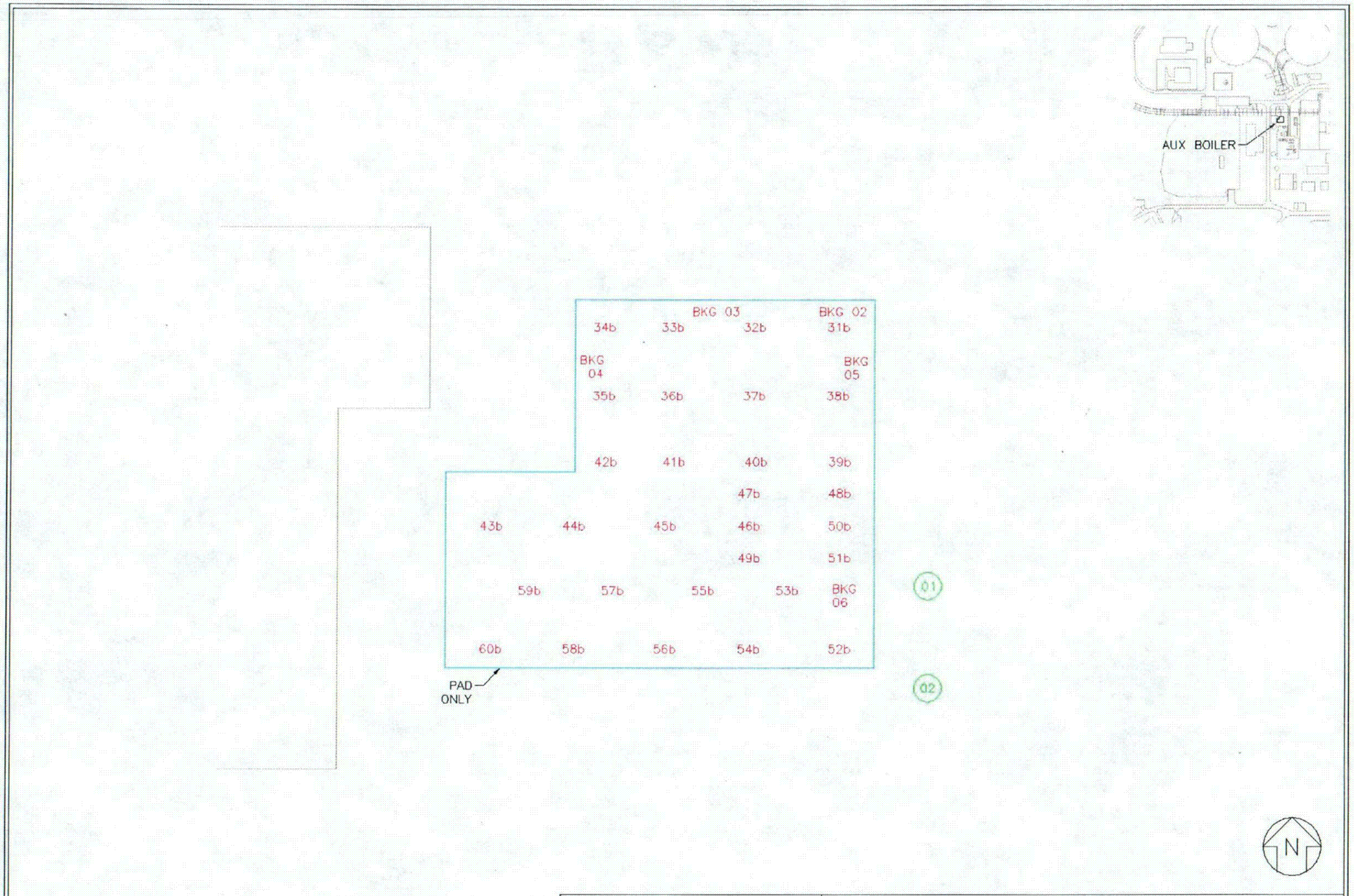
Figure 2-70
GRS Warehouse
828000

DWG: 828000.01

CC67



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DWG: 836000.01

Figure 2-71
Aux Boiler
836000

C68

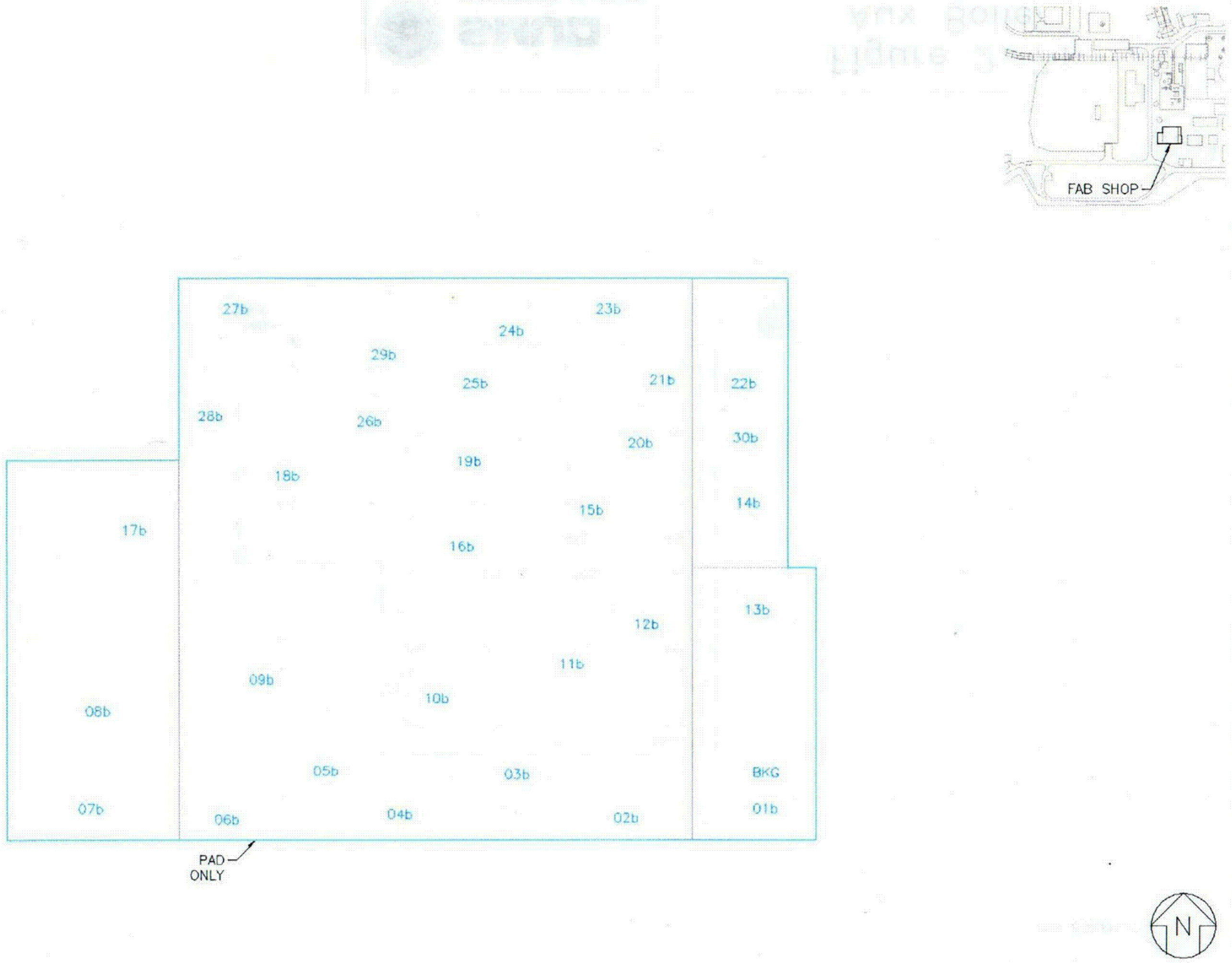

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Figure 2-72
Fab Shop
838000

DWG: 838000.01

C69

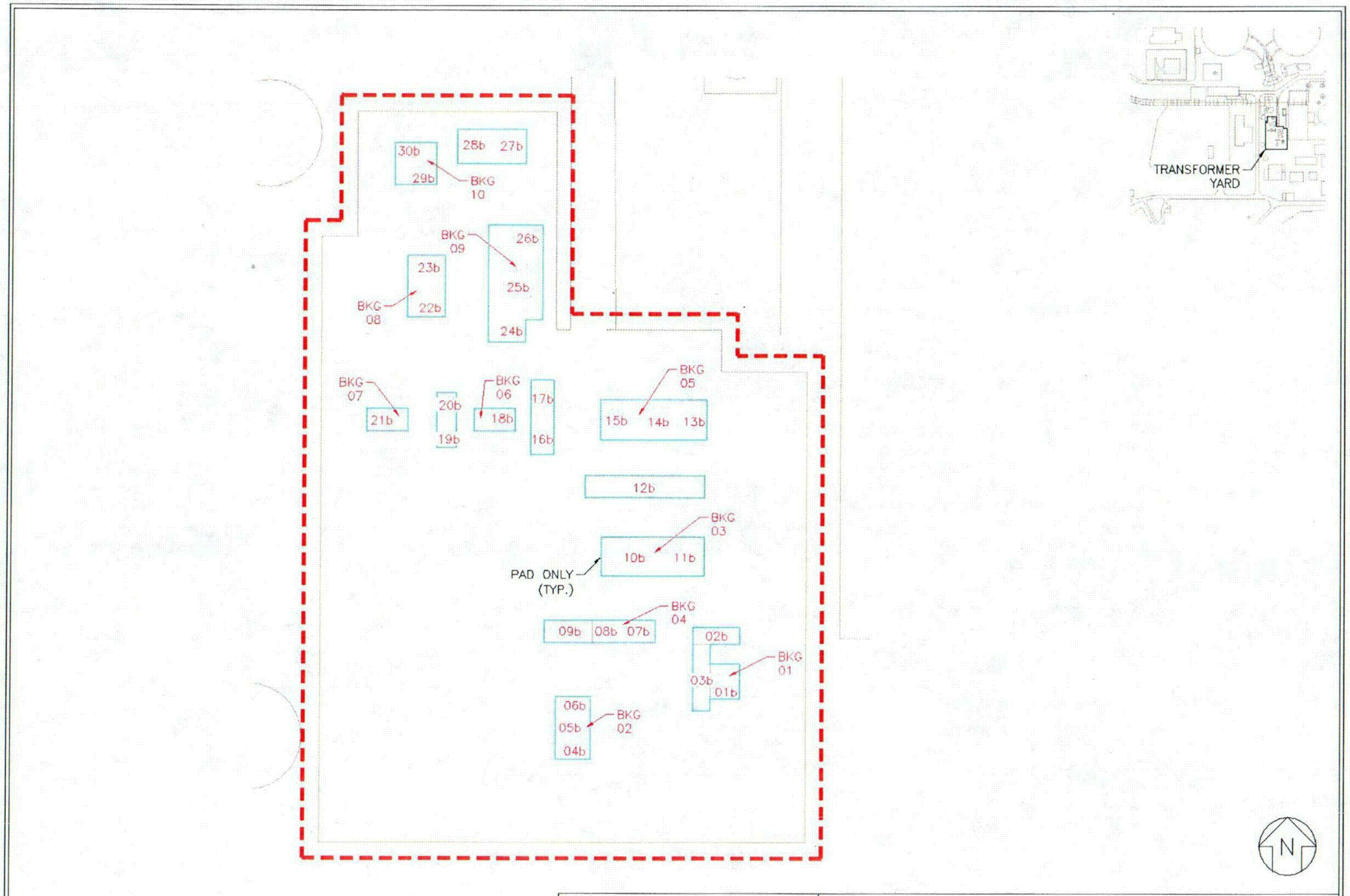

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Figure 2-73
Transformer Yard
839000

DWG: 839000.01

C70

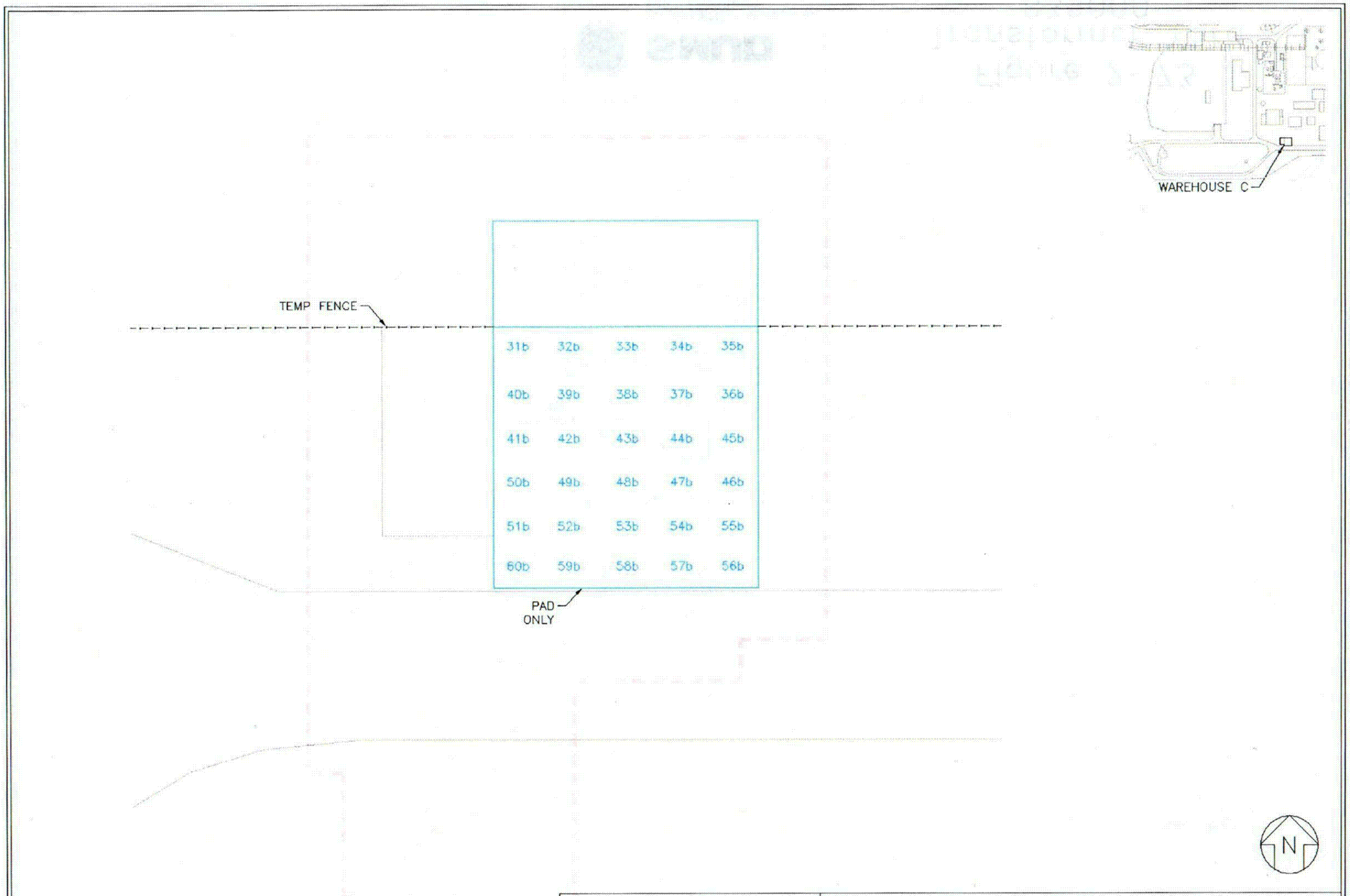

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Figure 2-74
Warehouse C
842000

DWG: 842000.01

C71

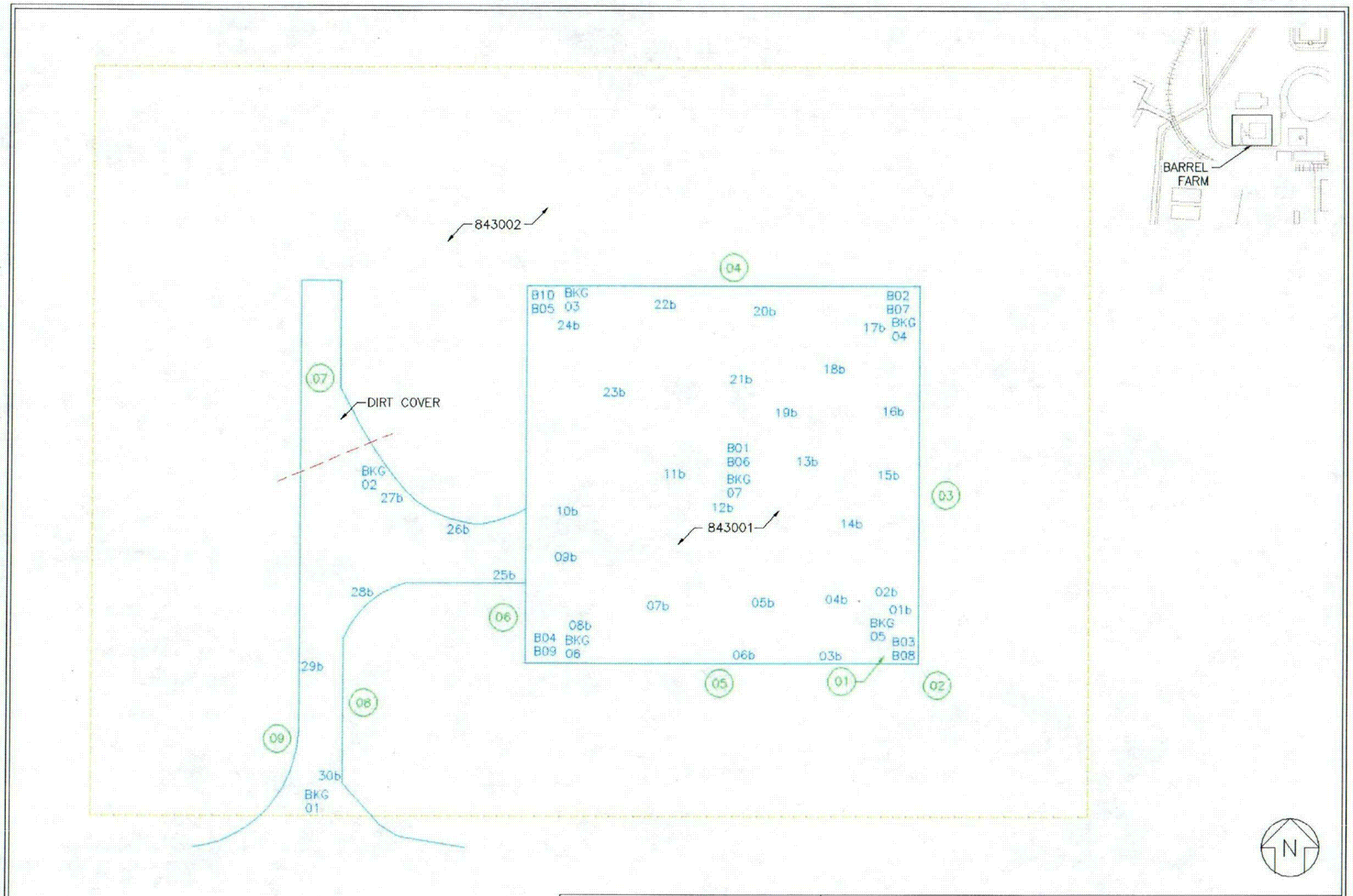

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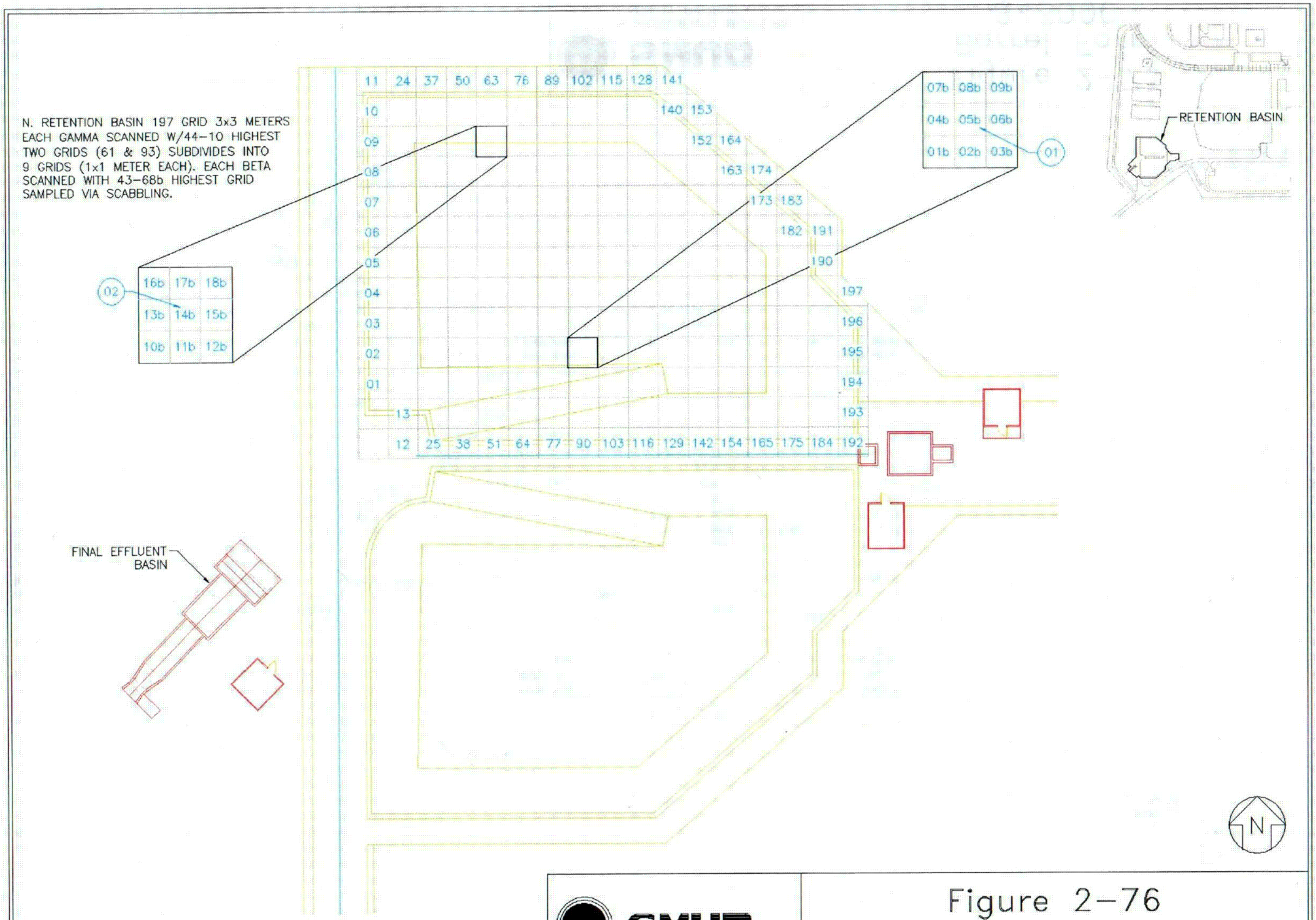
Figure 2-75
Barrel Farm
843000

DWG: 843000.01

C72



N. RETENTION BASIN 197 GRID 3x3 METERS EACH GAMMA SCANNED W/44-10 HIGHEST TWO GRIDS (61 & 93) SUBDIVIDES INTO 9 GRIDS (1x1 METER EACH). EACH BETA SCANNED WITH 43-68b HIGHEST GRID SAMPLED VIA SCABBLING.

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Figure 2–76
Retention Basin
848000

DWG: 848000.01

C73

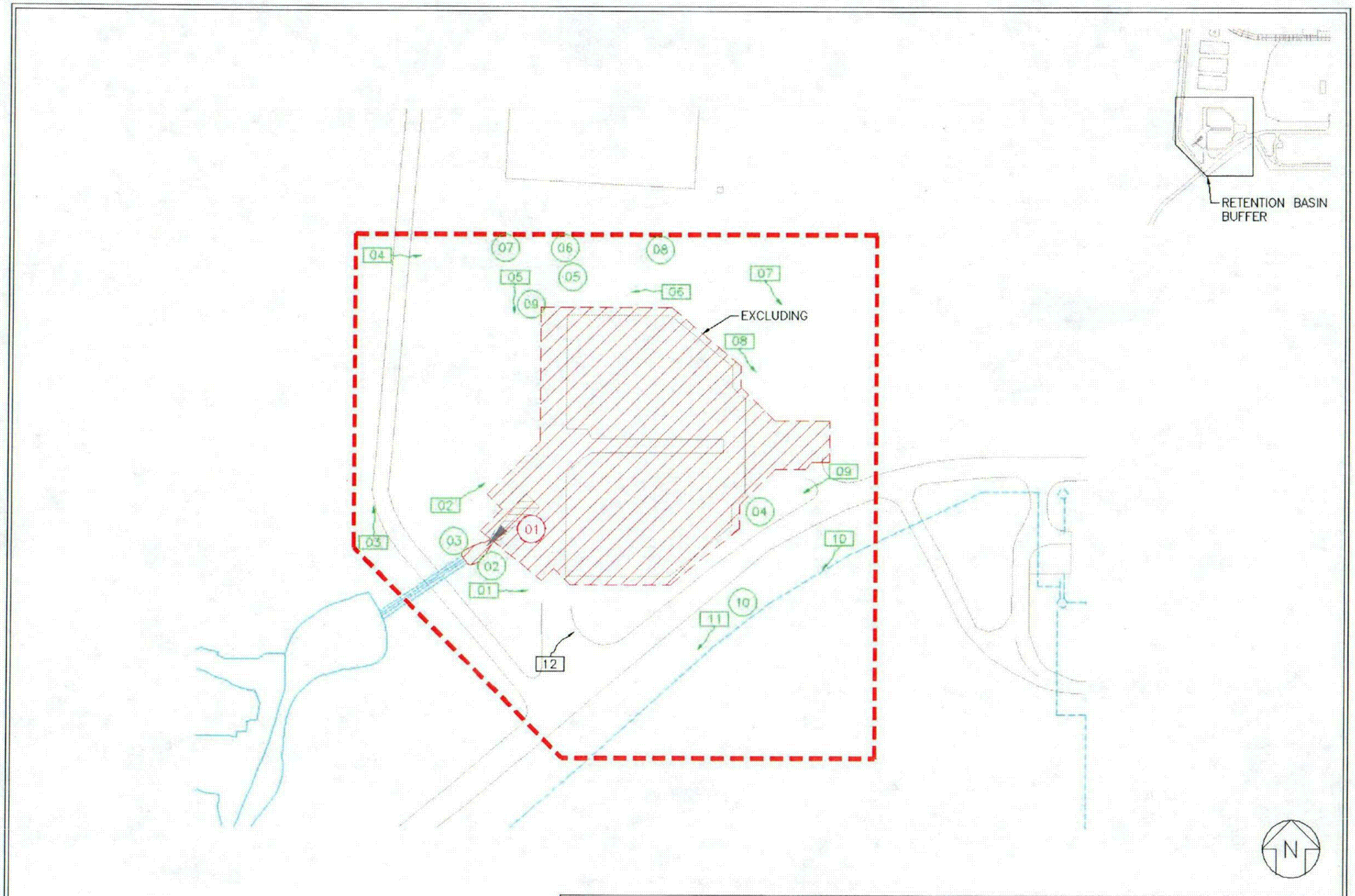

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Figure 2-77
Retention Basin Buffer
848002

DWG: 848002.01

C74

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Figure 2–78
Switch Yard
851000

DWG: 851000.01

C75

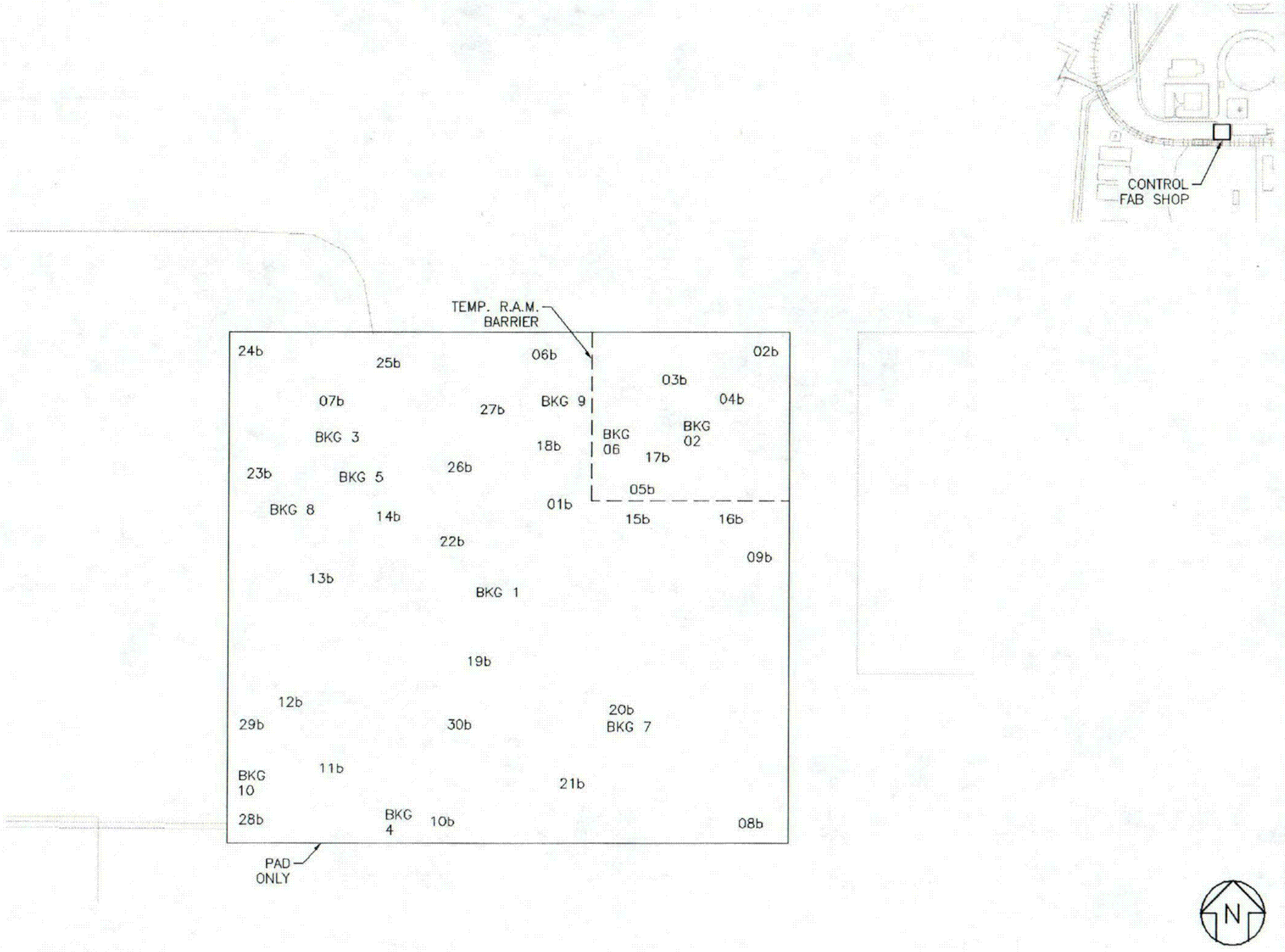

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Figure 2-79
Control Fab Shop
853000

DWG: 853000.01

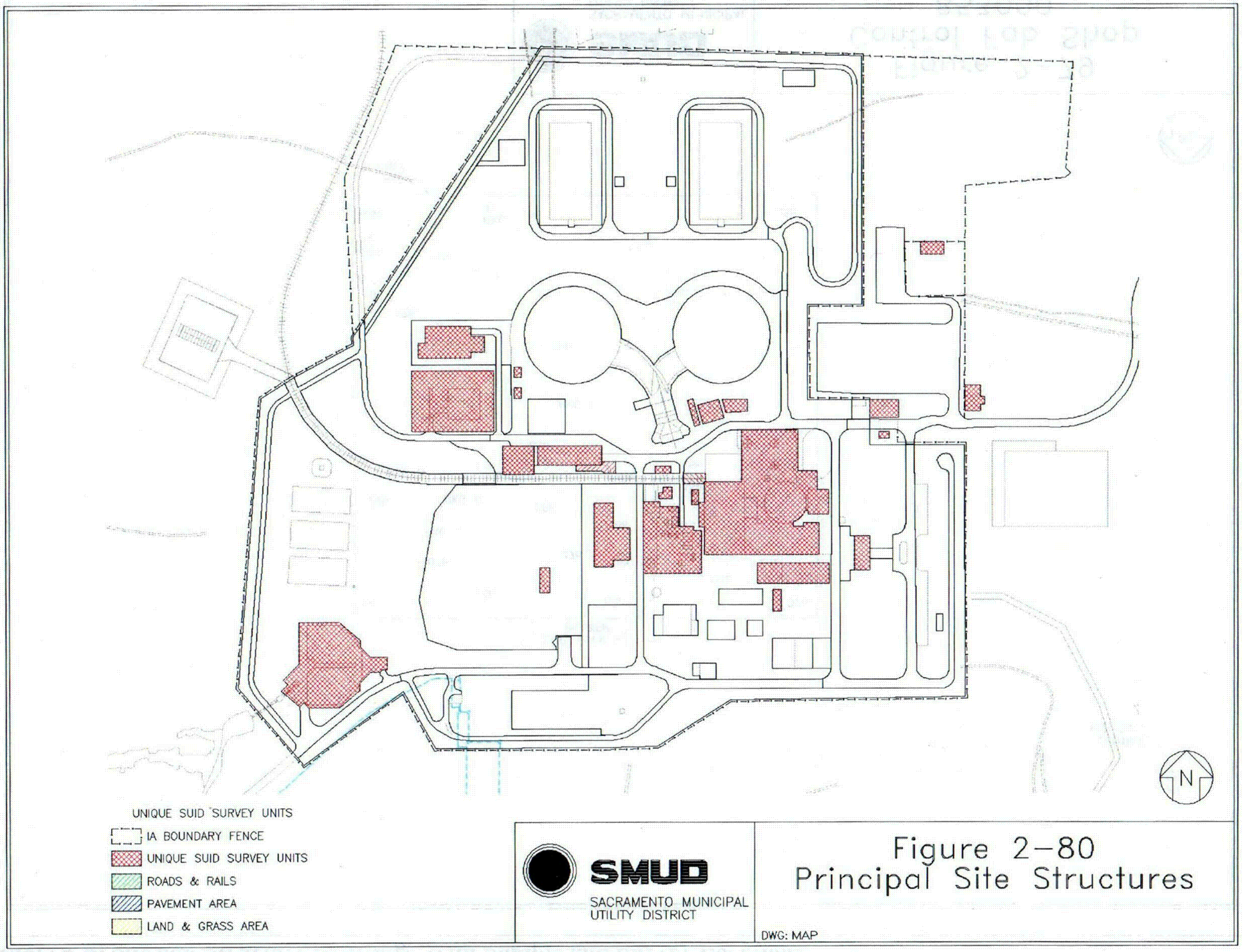

Figure 2-80
Principal Site Structures

C76

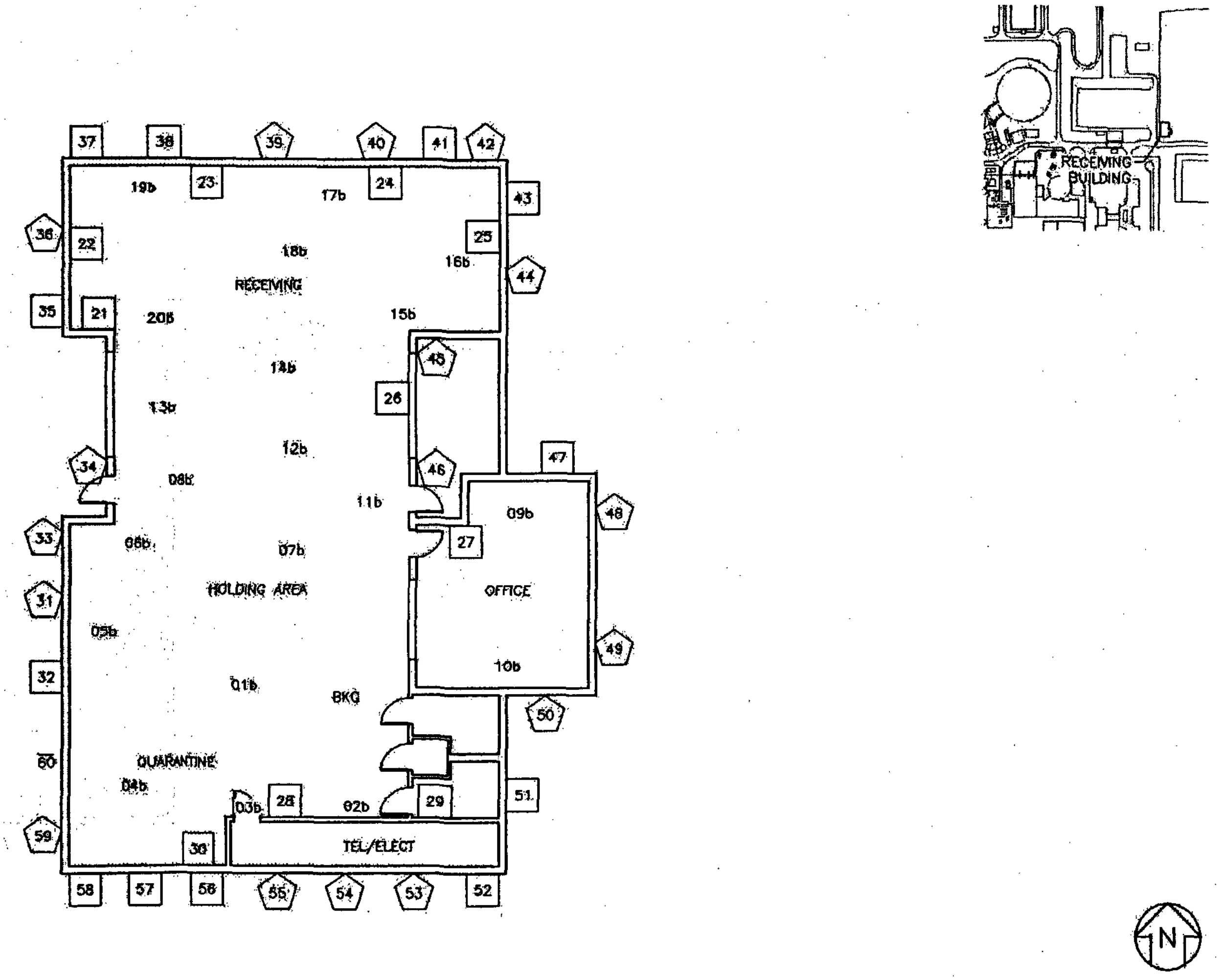

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Figure 2–81
Receiving Building
501001

DWG: 501001.01

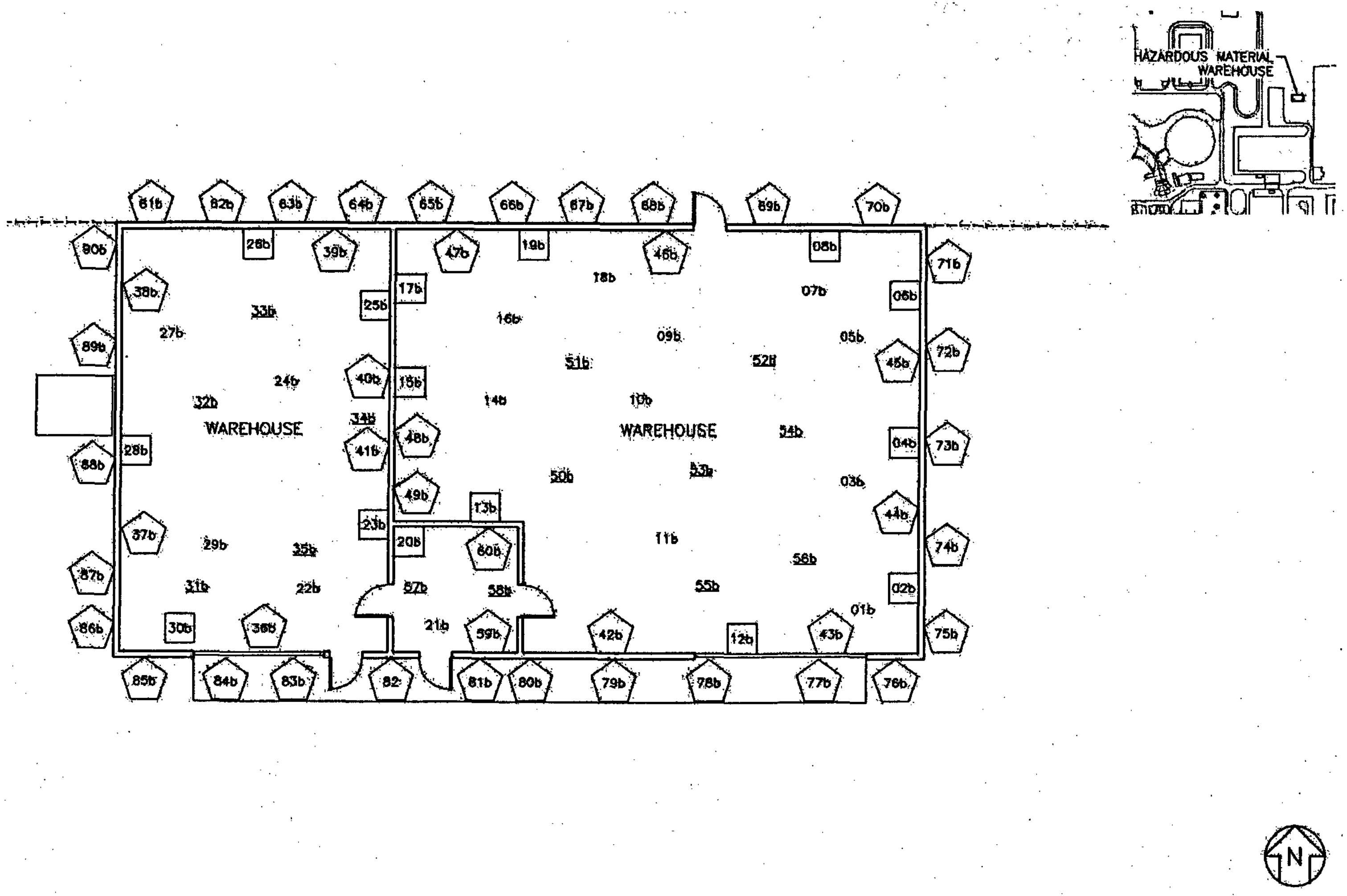

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Figure 2–82
Hazard Material Warehouse
501002

DWG: 501002.01

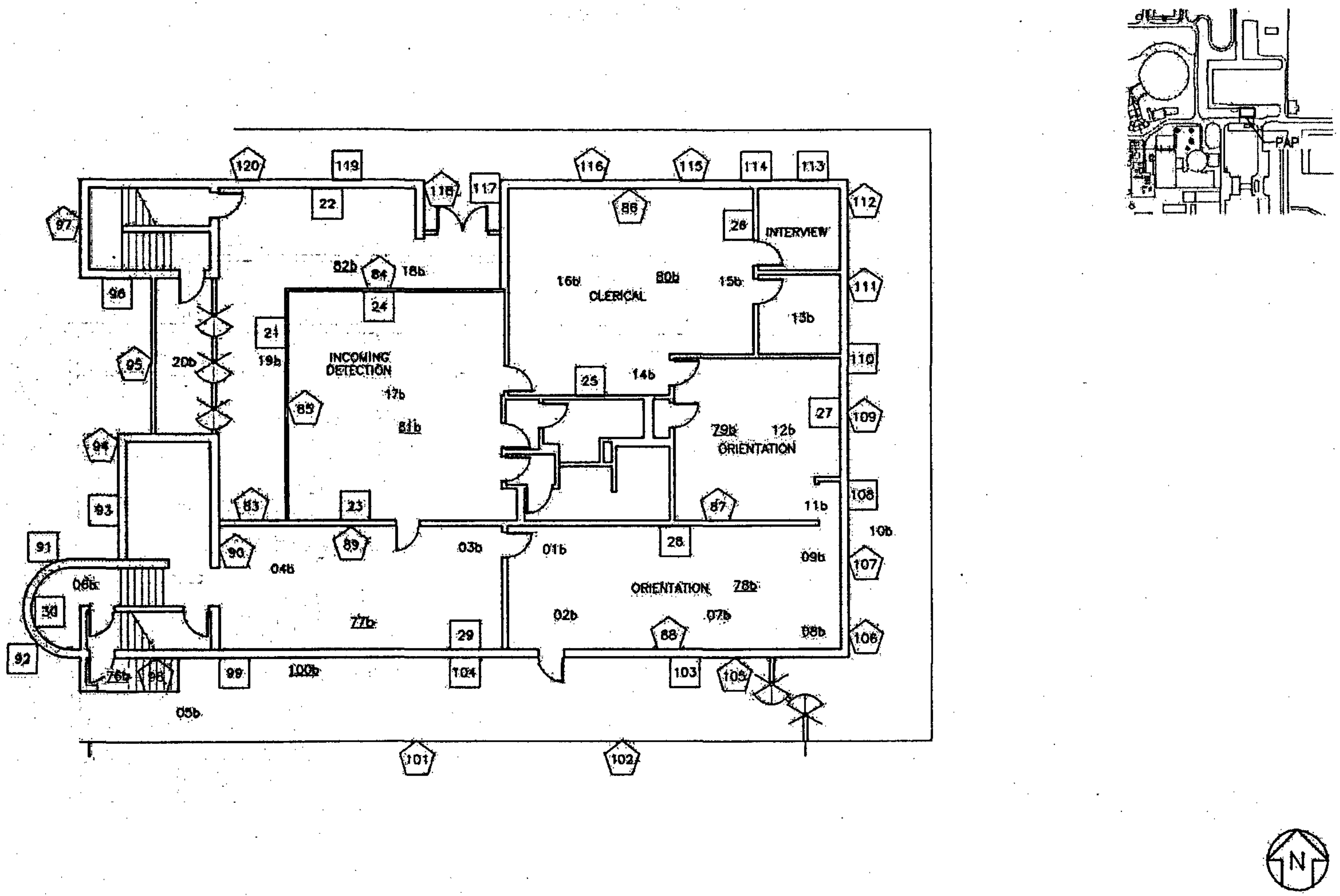

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Figure 2-83
PAP Building 1st Floor
804001

DWG: 804001.01

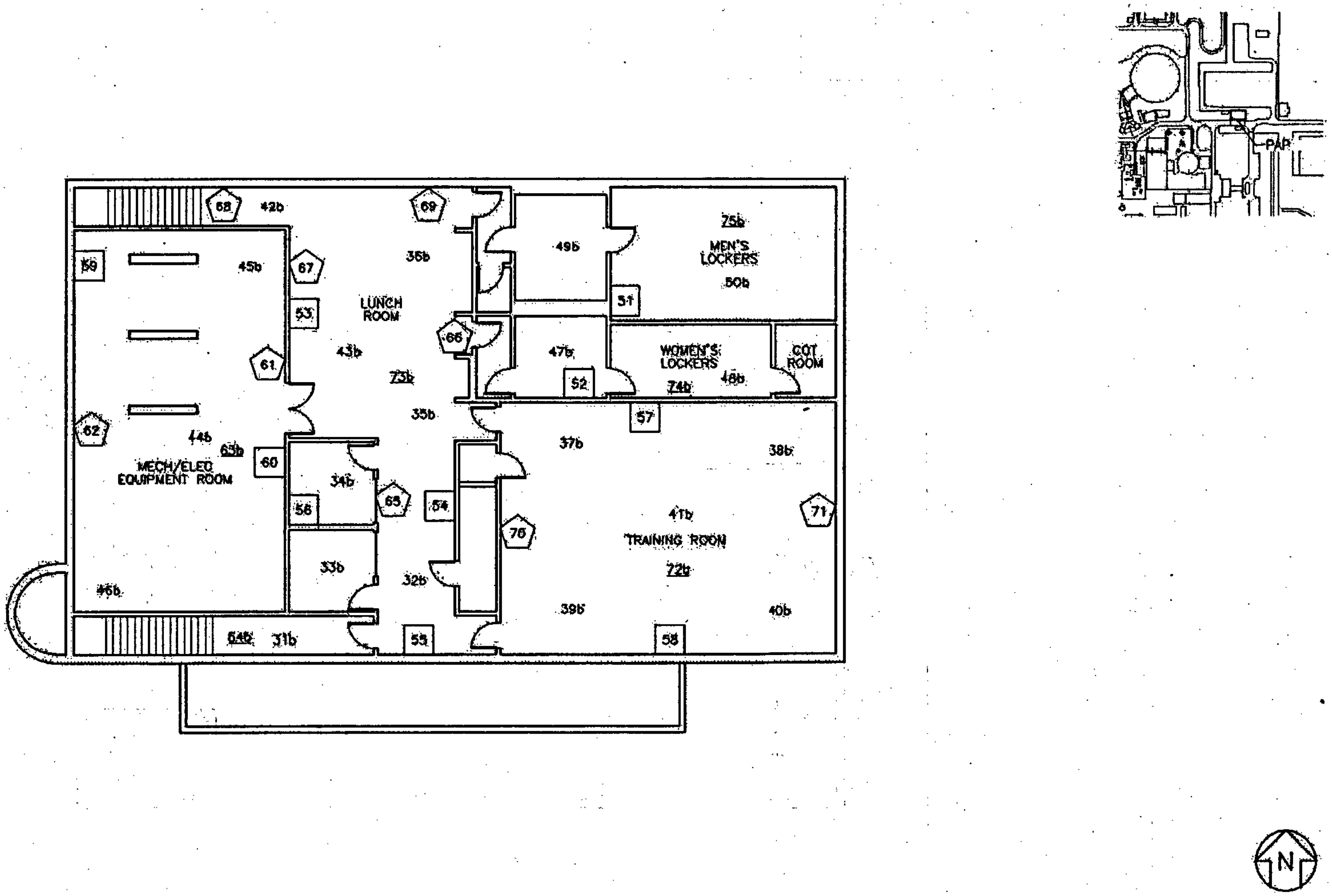

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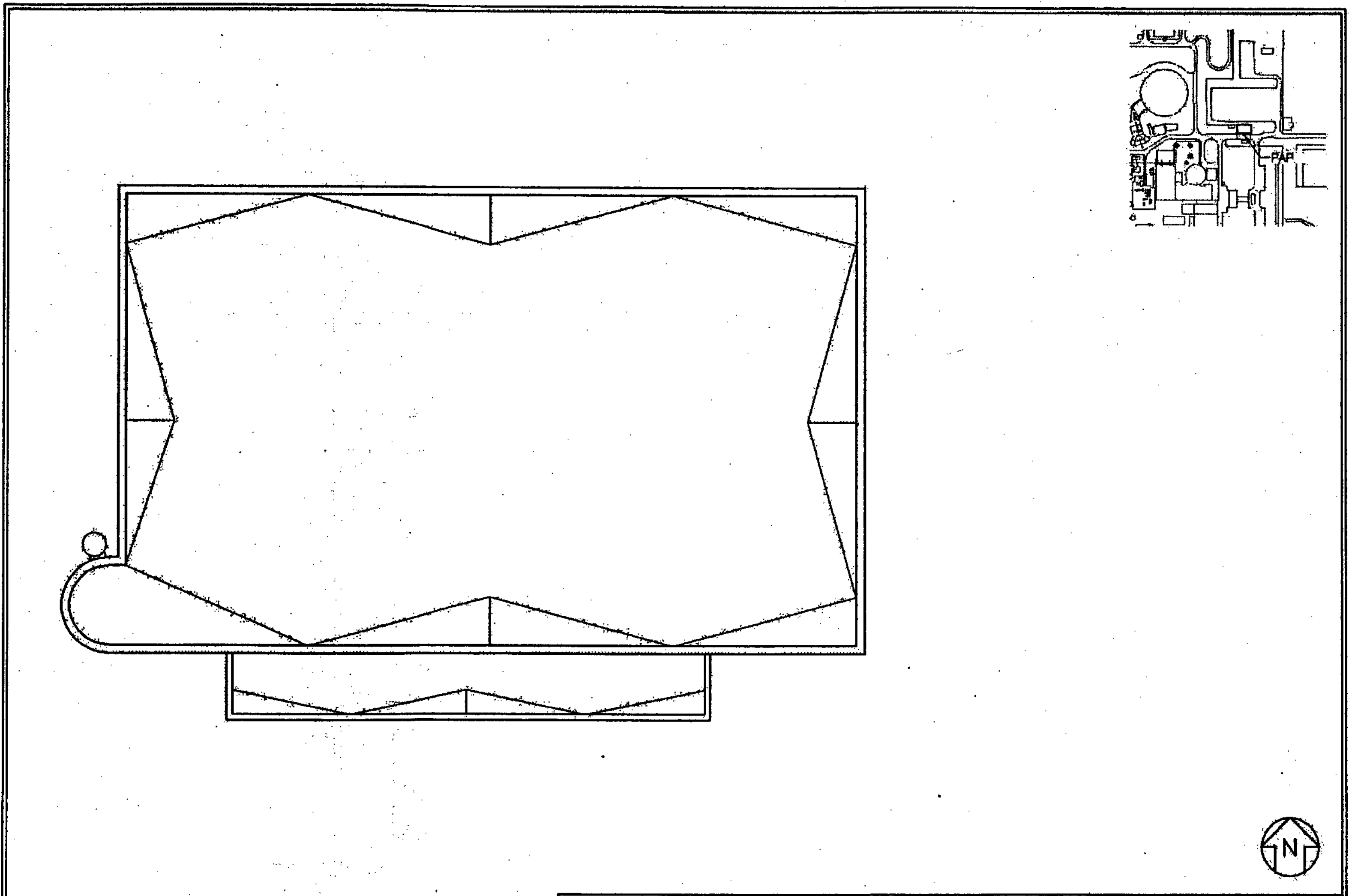
Figure 2-84
PAP Building 2nd Floor
804001

DWG: 804001.02



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UTILITY DISTRICT

Figure 2–85
PAP Building Roof
804001

DWG: 804001.03

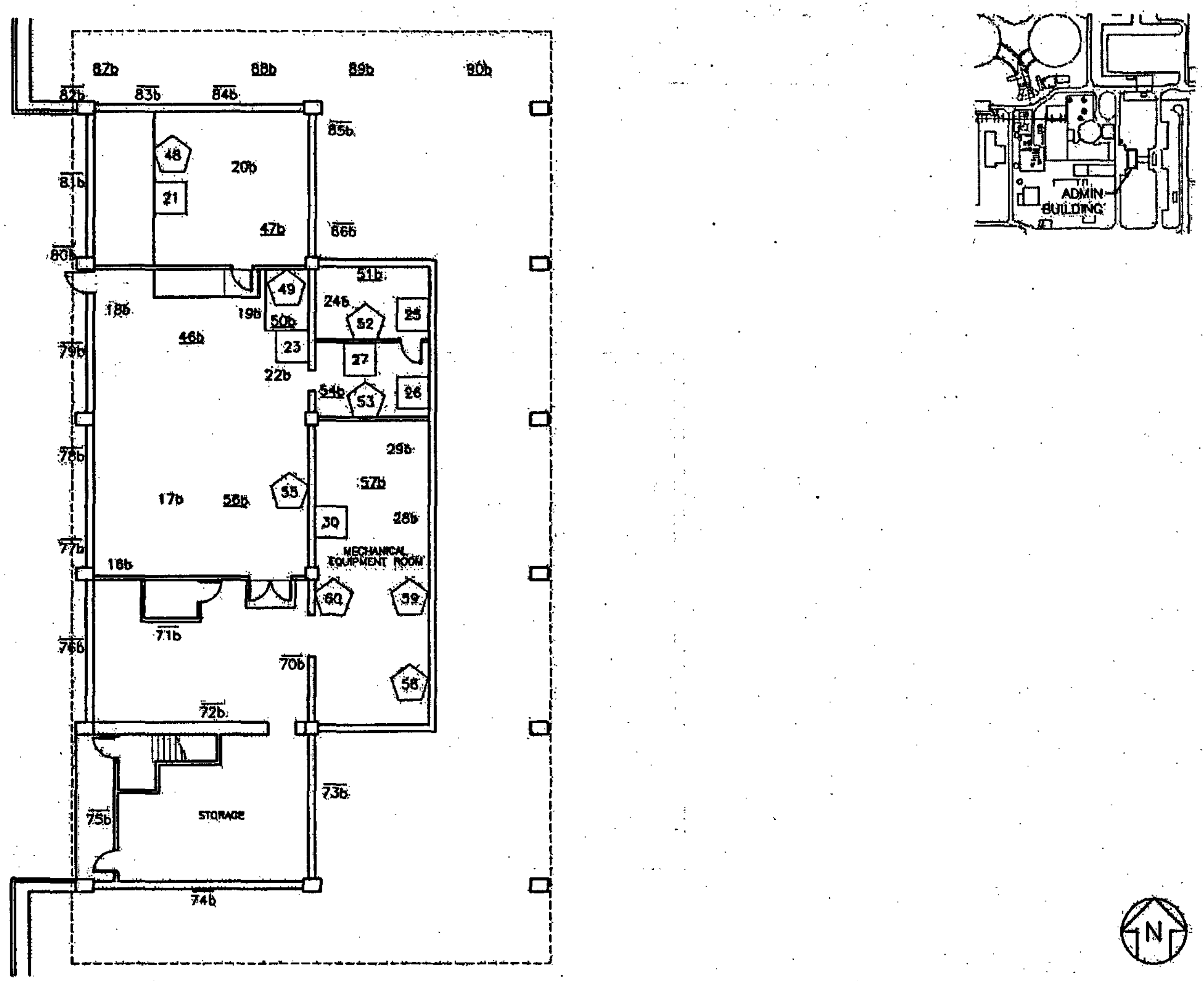

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Figure 2-86
Admin. Building El. 0
805001

DWG: 805001.01

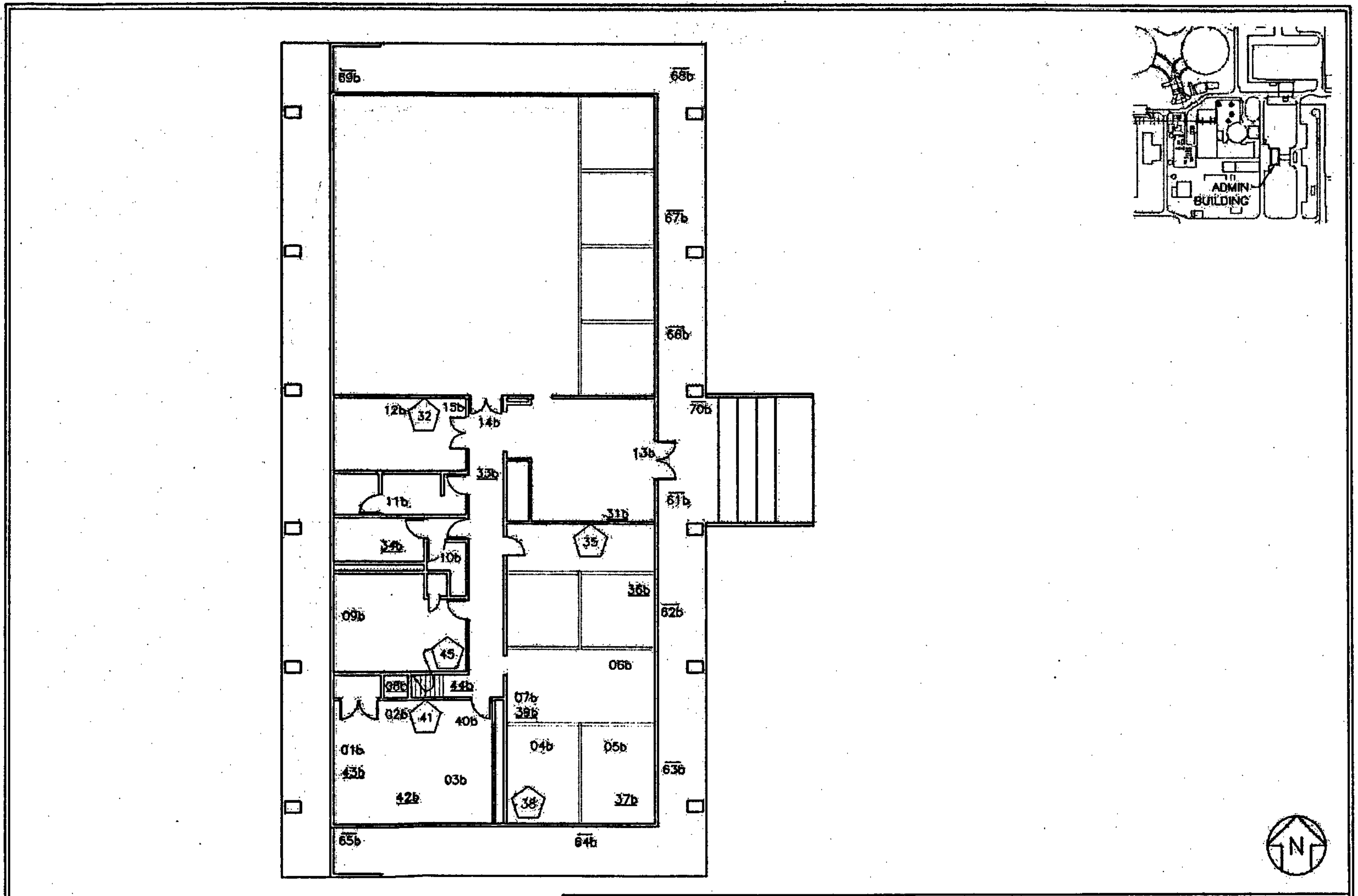

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Figure 2-87
Admin. Building 2nd Floor
805001

DWG: 805001.02

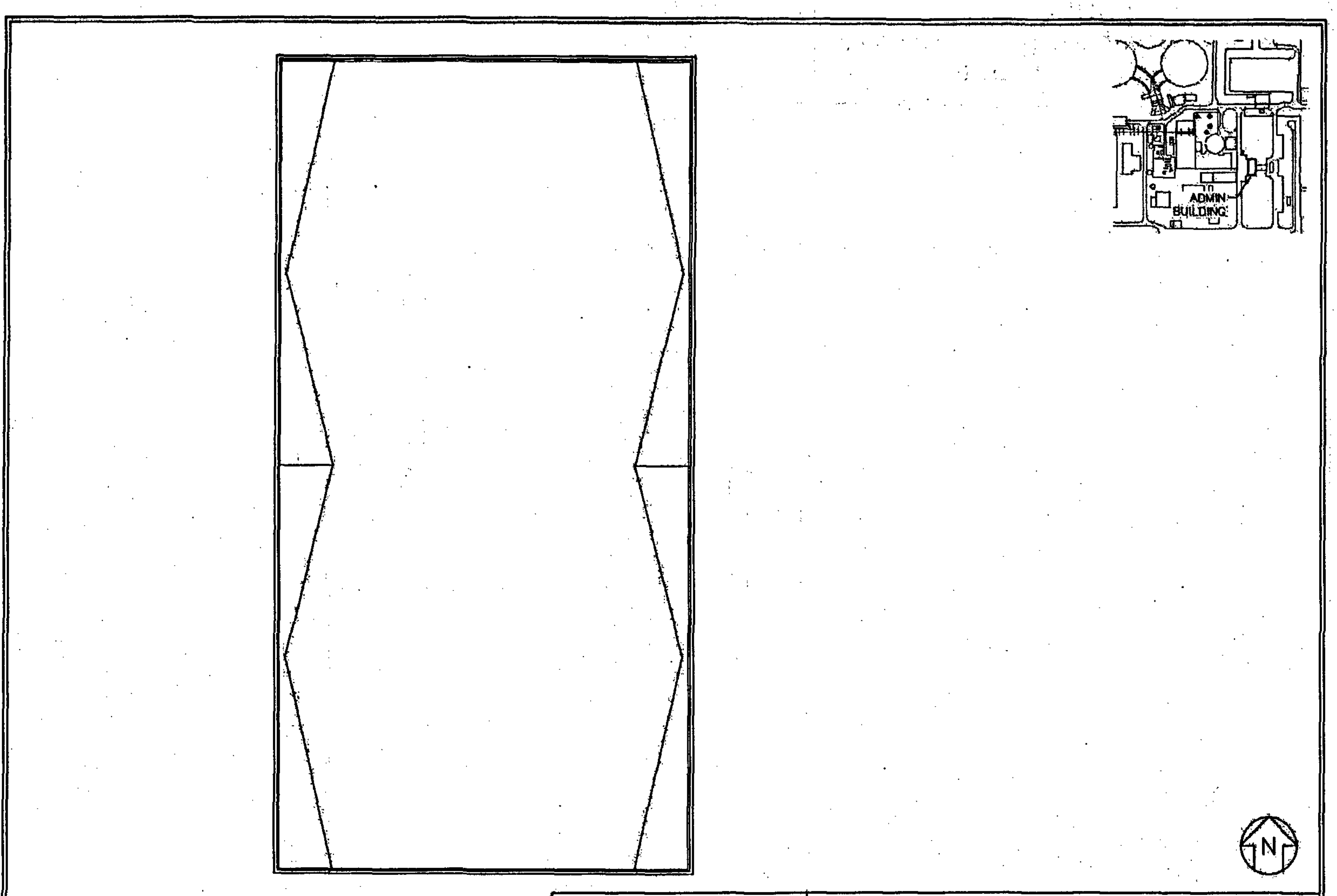

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Figure 2–88
Admin. Building Roof
805001

DWG: 805001.03

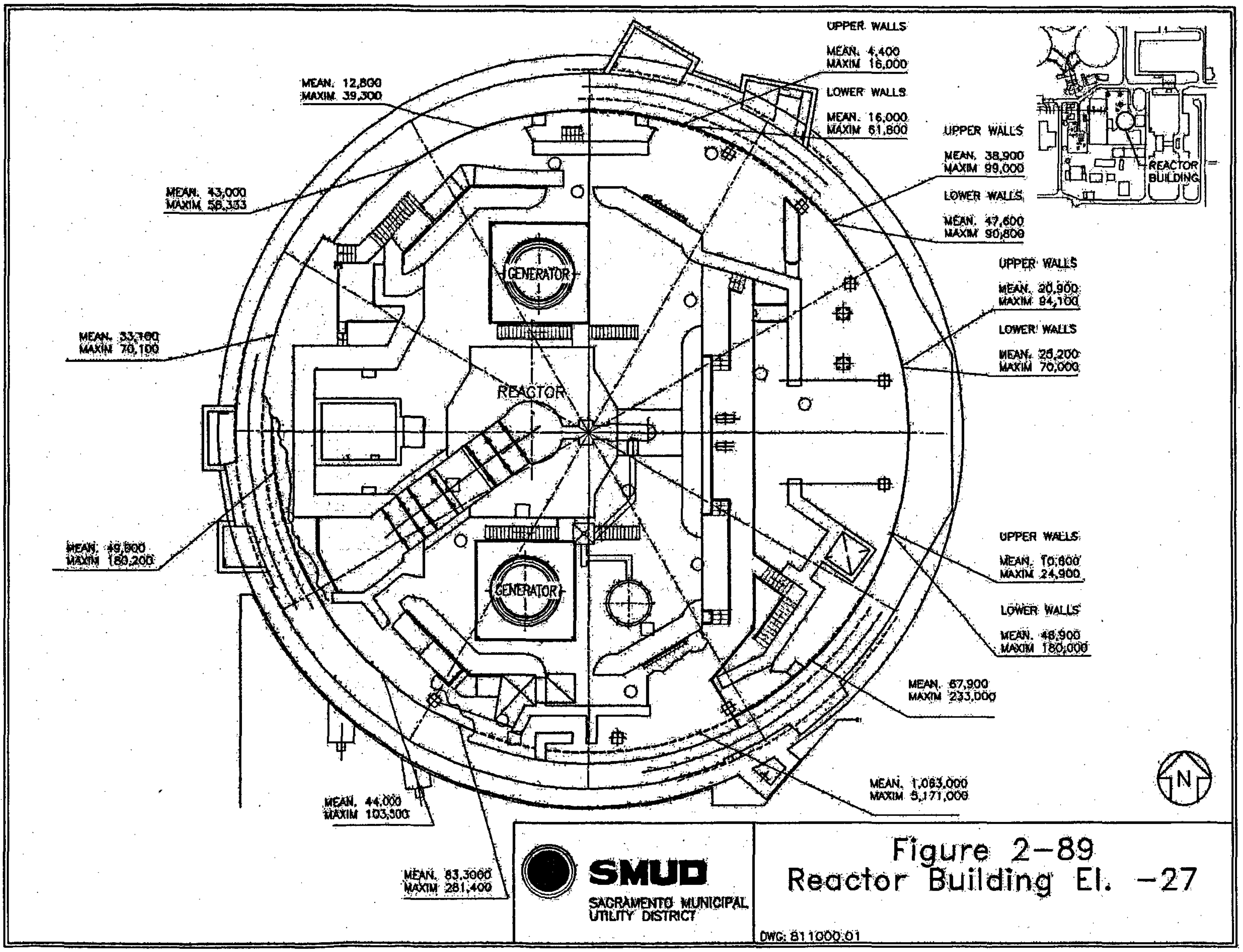

Figure 2-89
Reactor Building El. -27

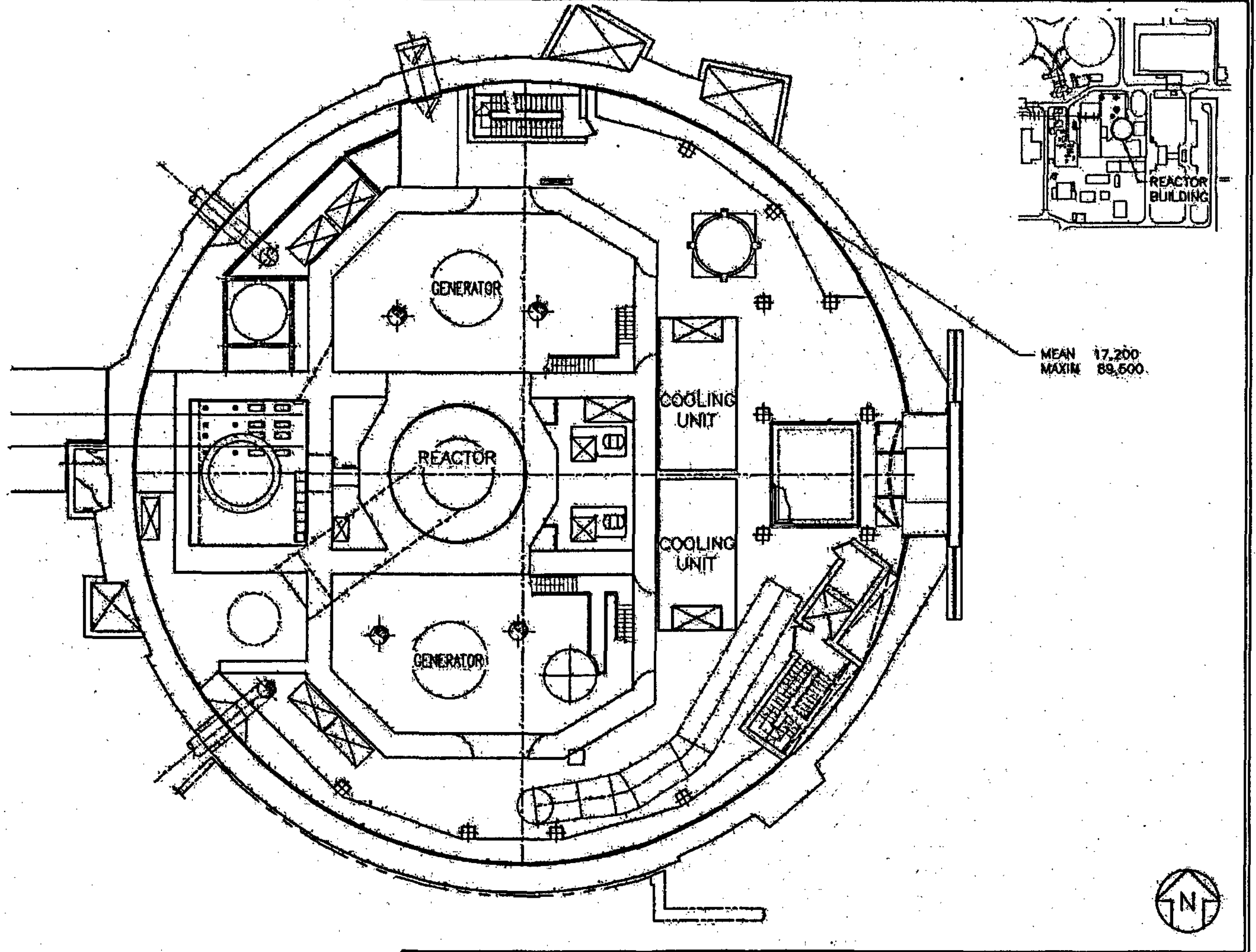

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Figure 2–90
Reactor Building El. 0

DWG: 81 1000.02

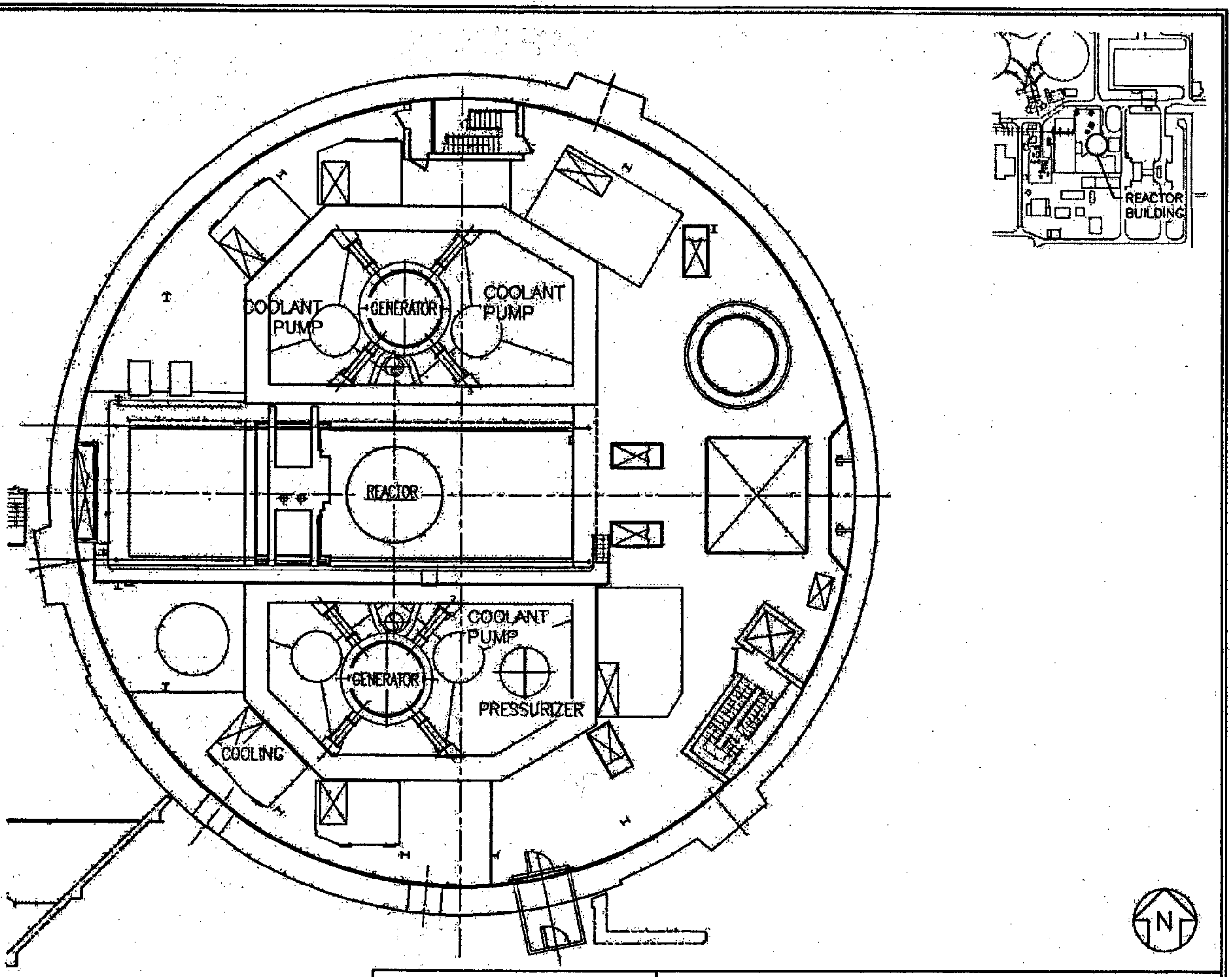

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Figure 2-91
Reactor Building El. 40

DWG: 81 1000.04

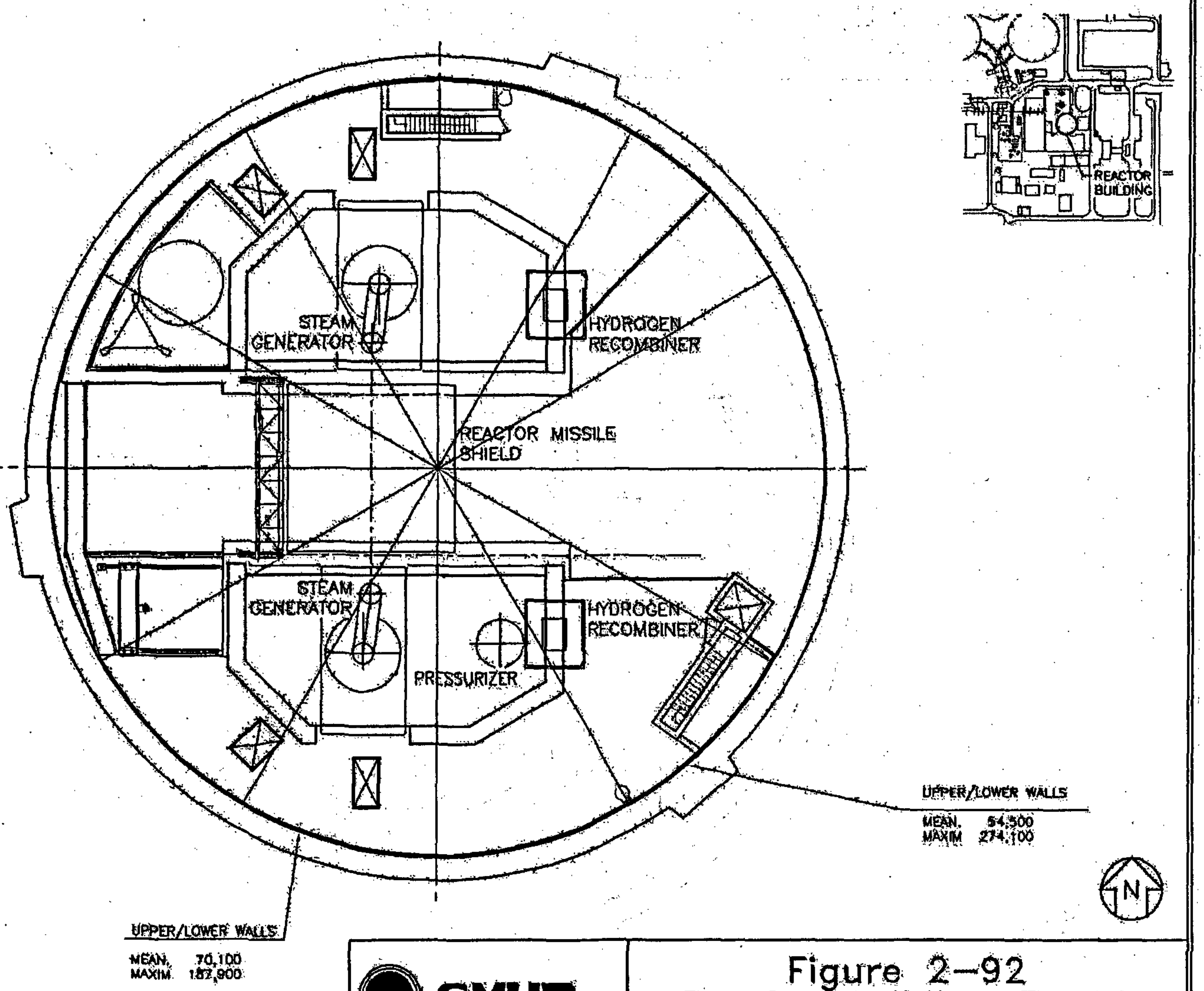

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Figure 2-92
Reactor Building El. 60
811000

DWG: 811000.05

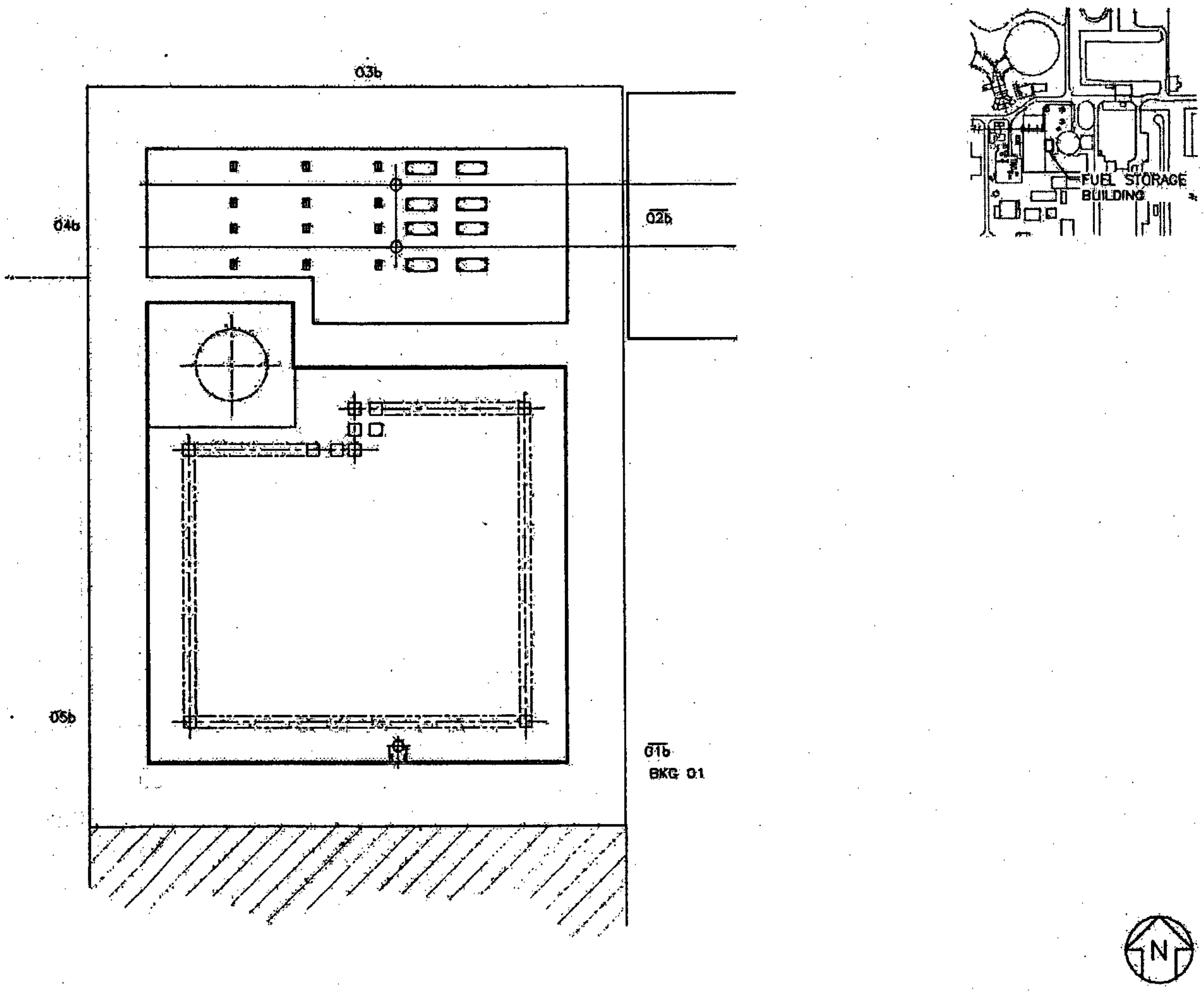

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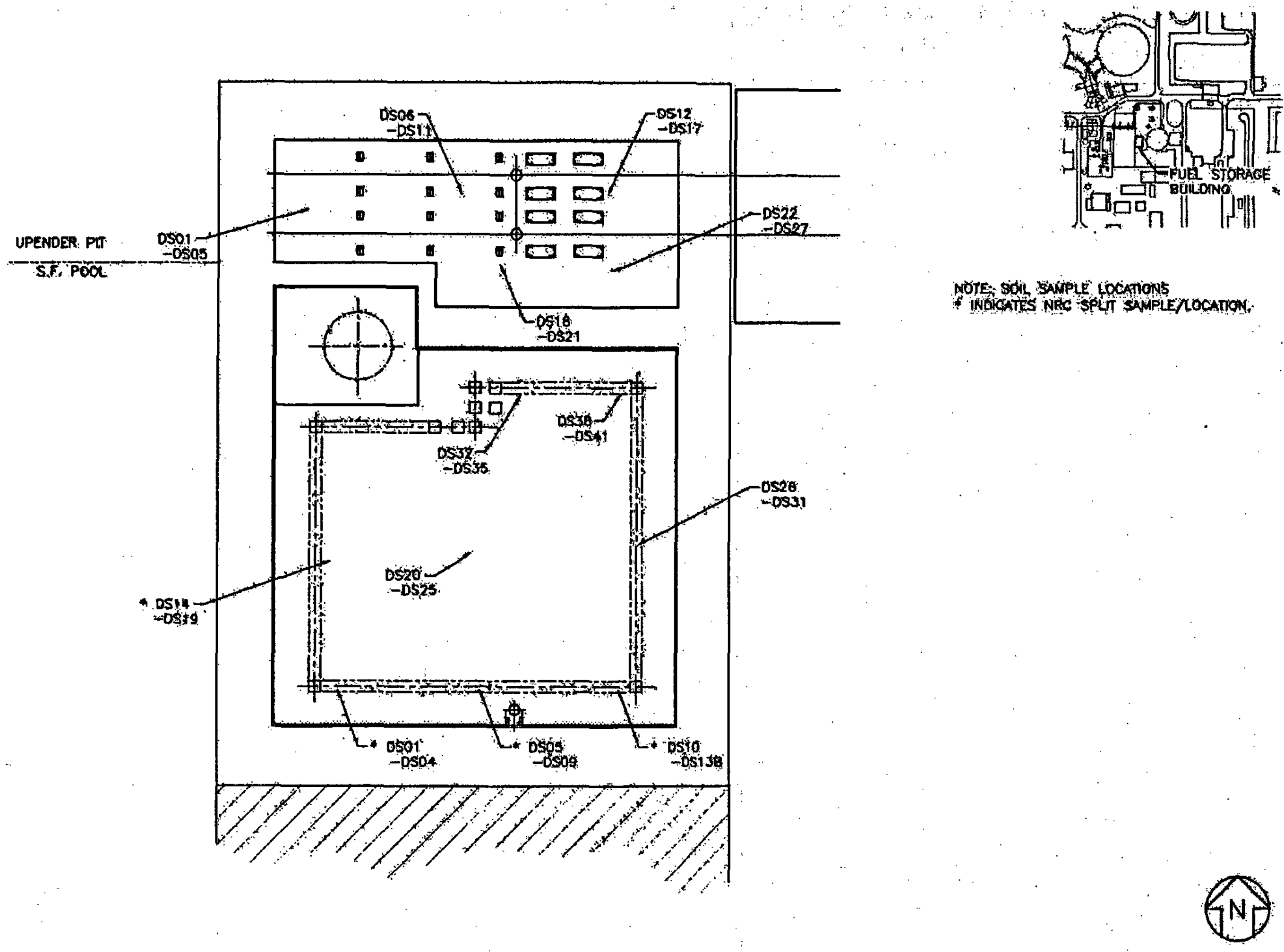
Figure 2-93
Fuel Storage Bldg Exterior El. 0
812000

DWG: 812000.01



NOTE: SOIL SAMPLE LOCATIONS
* INDICATES NRC SPLIT SAMPLE/LOCATION.

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Figure 2-94
Fuel Storage Bldg Subsoil El. 0
812000

DWG: 812000.01

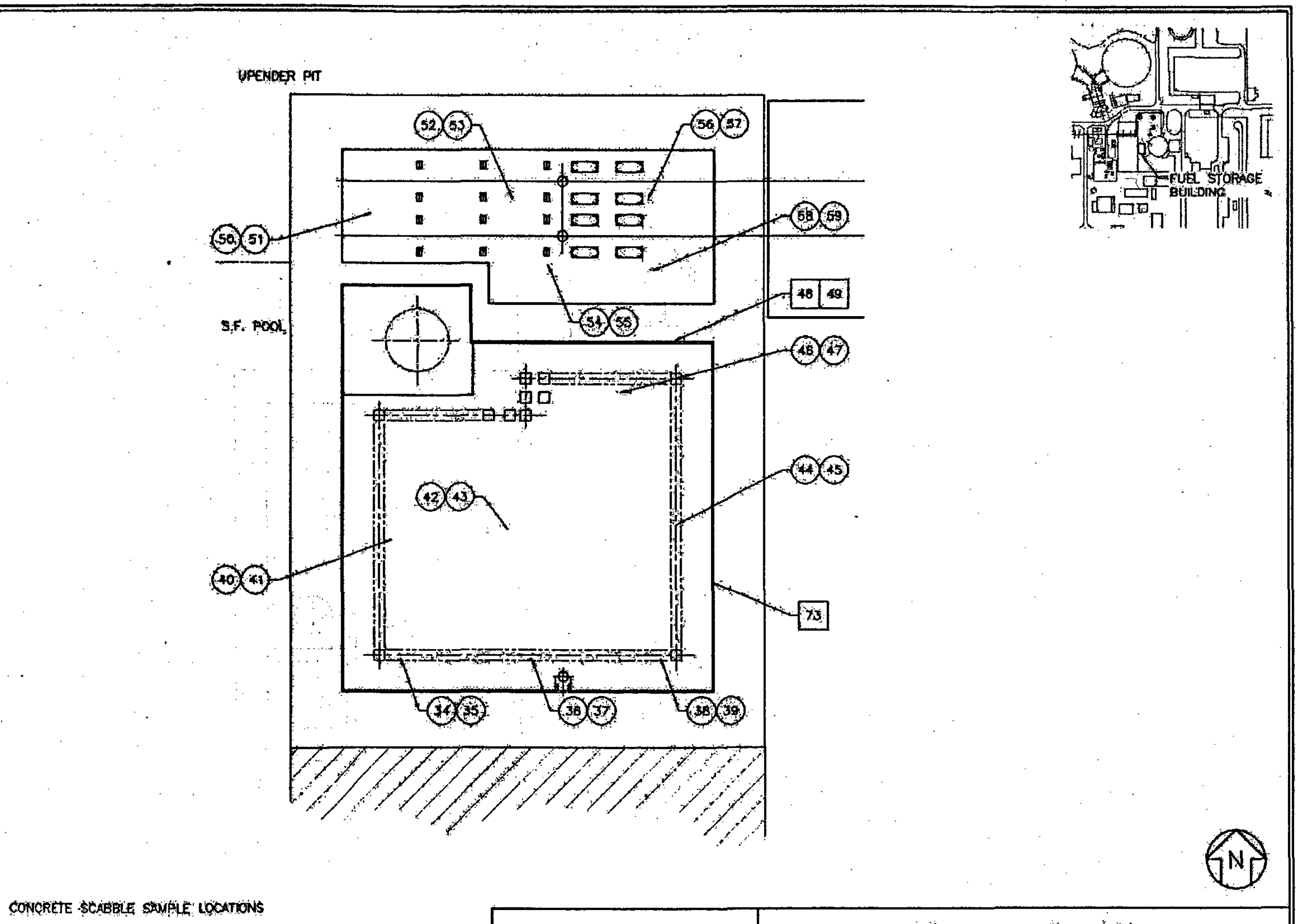

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Figure 2-95
Fuel Storage Building El. 0
812000

DWG: 812000.01

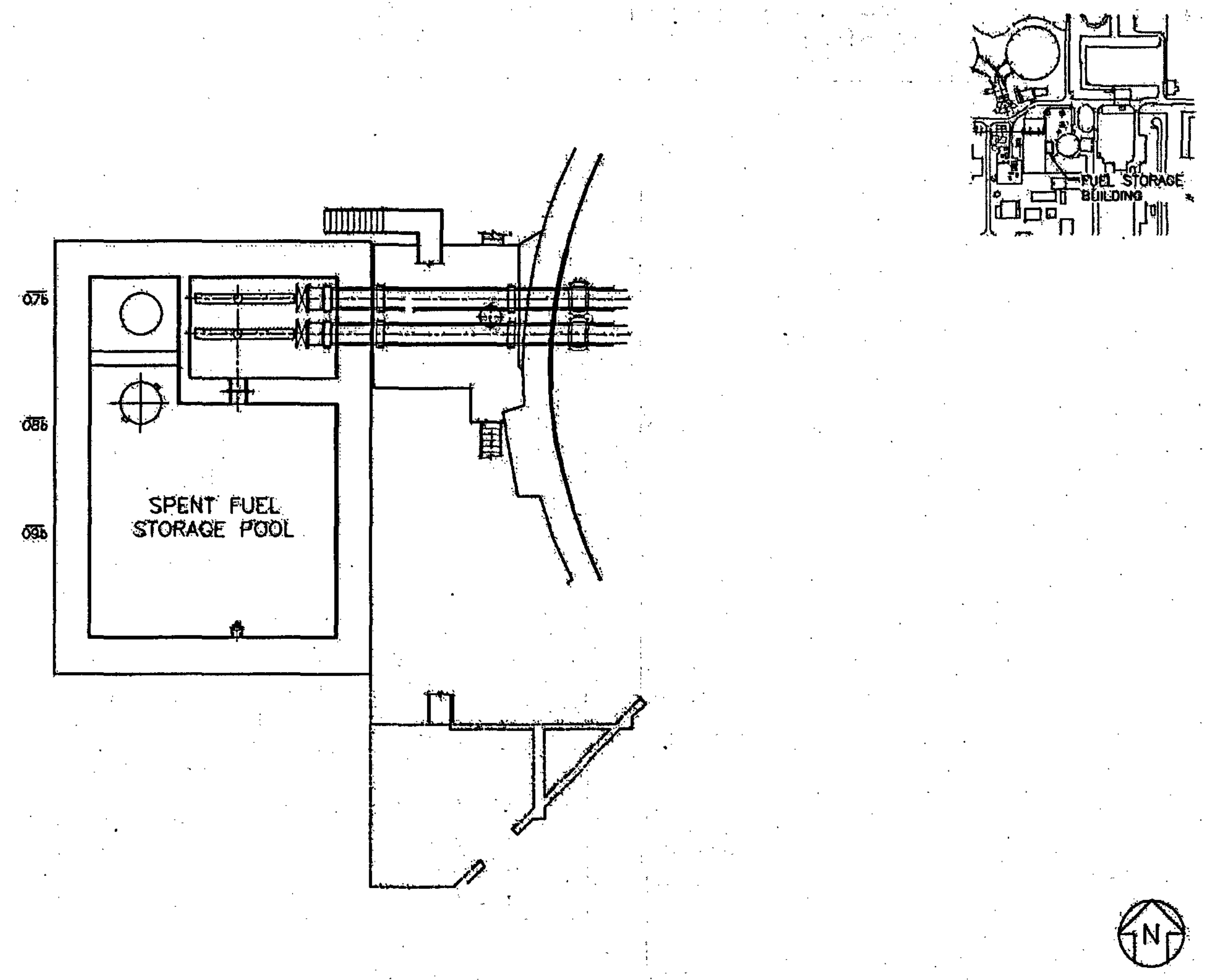

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Figure 2–96
Fuel Storage Building El. 20
812000

DWG. 812000.02

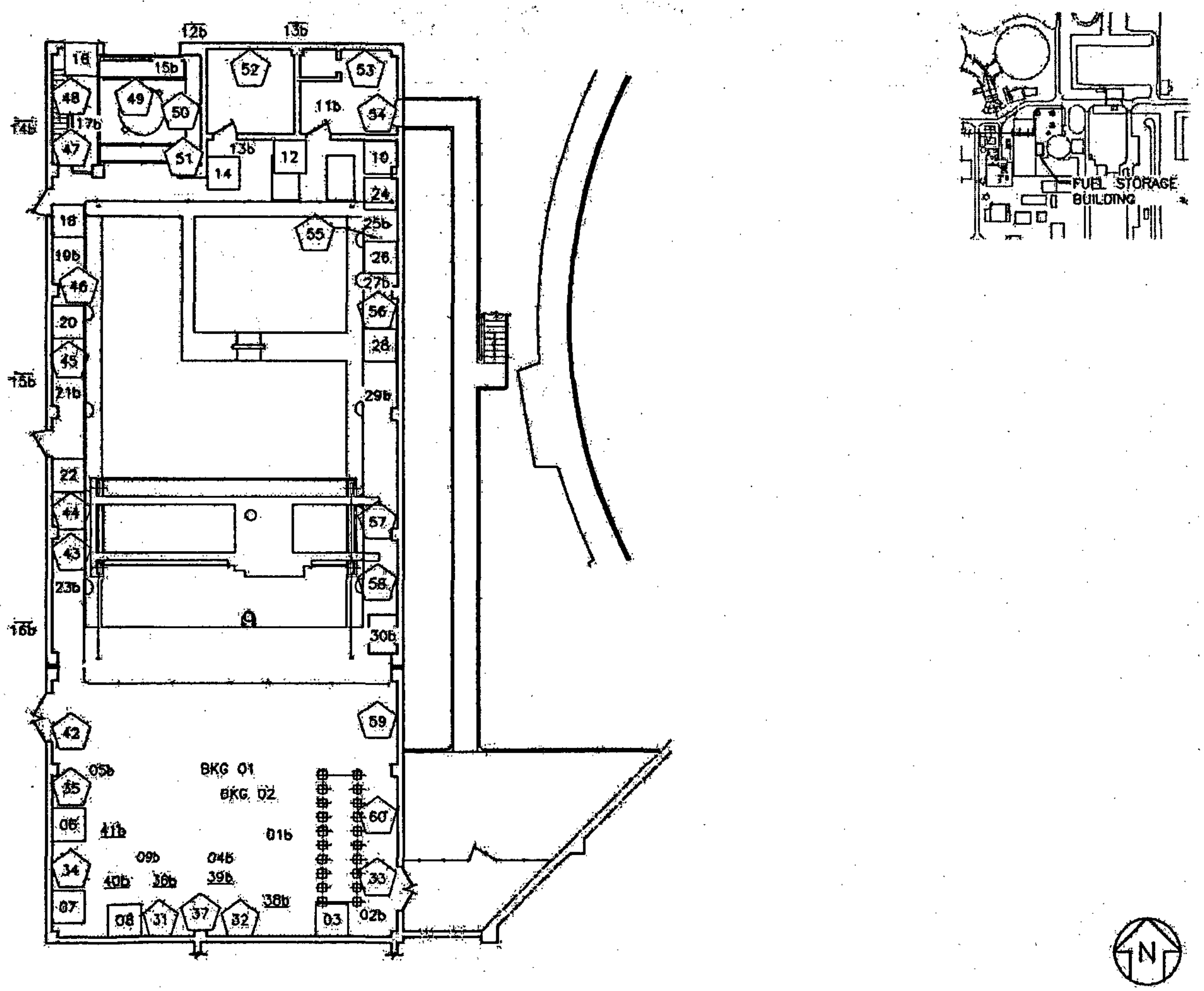

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Figure 2-97
Fuel Storage Building El. 40
812000

DWG: 812000.03

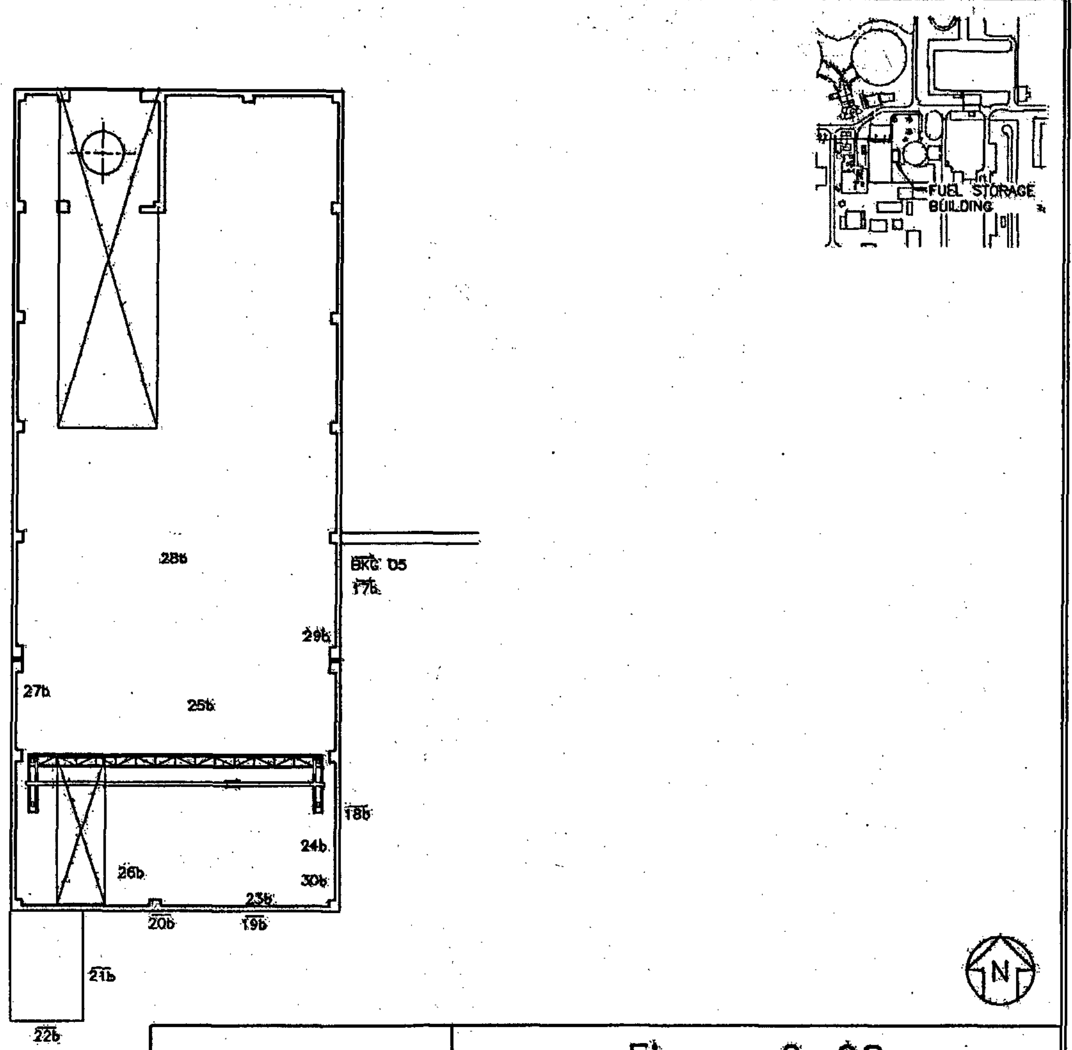

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Figure 2-98
Fuel Storage Building Roof
812000

DWG: 812000.04

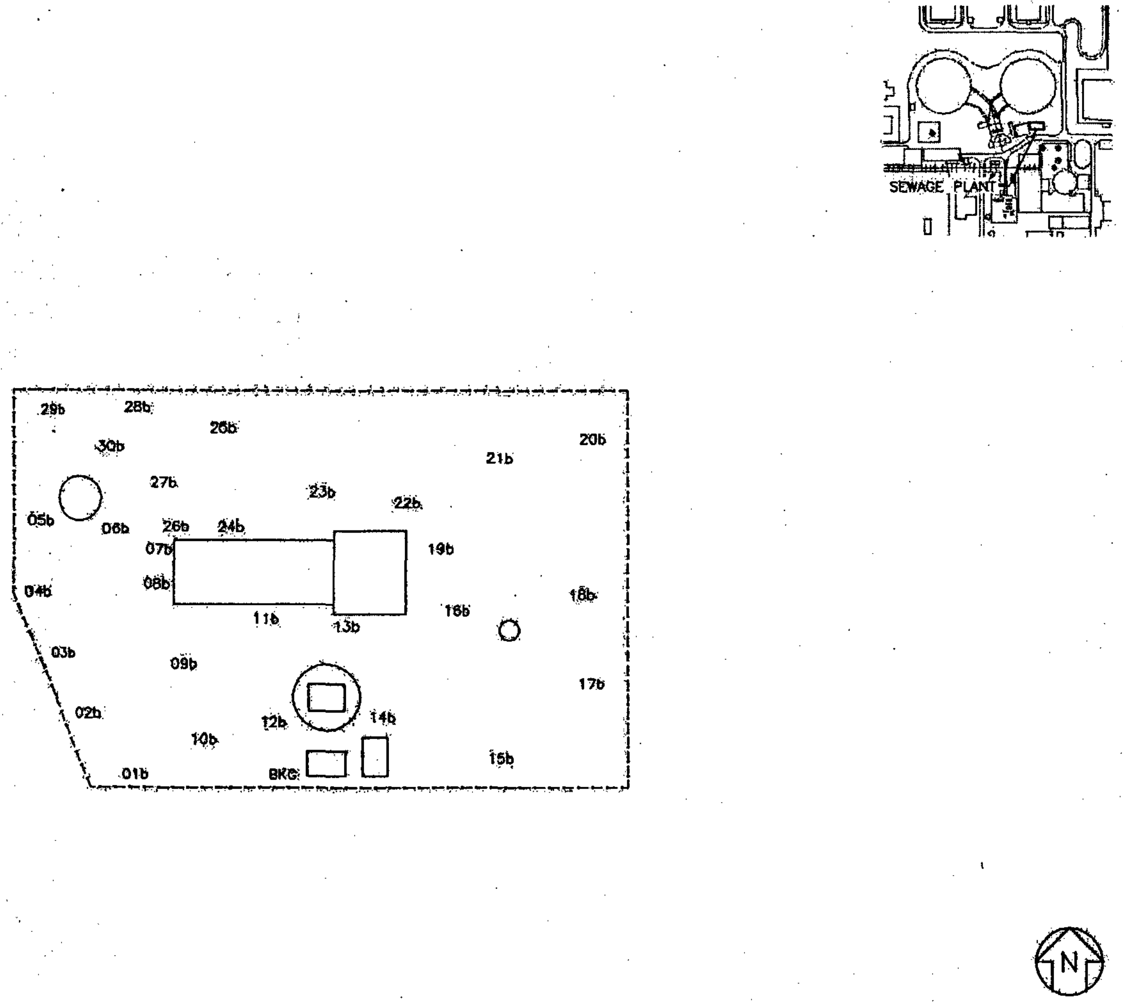

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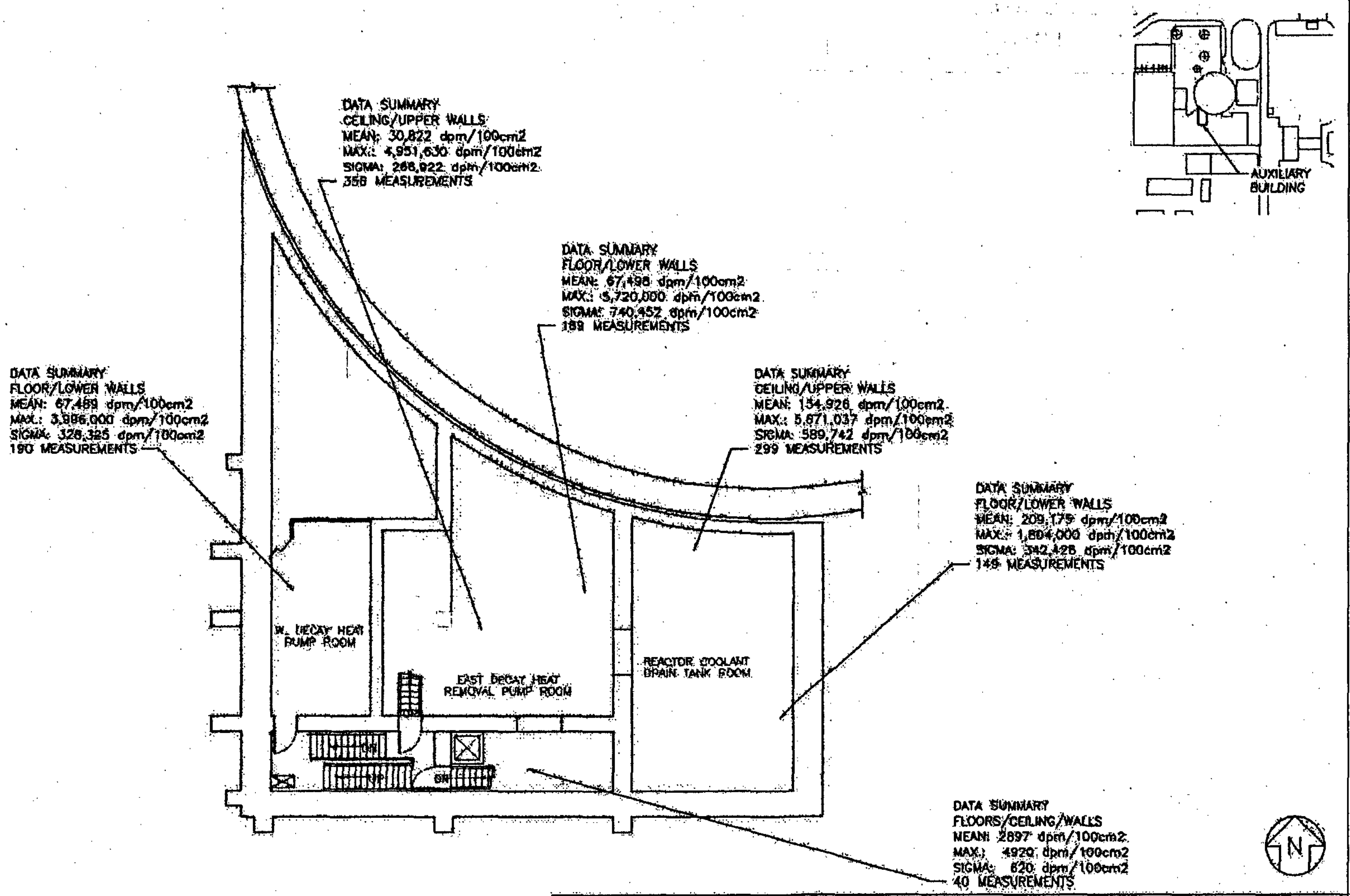
Figure 2–99
Sewage Plant
809001

DWG: 809001.01



DATA SUMMARY
CEILING/UPPER WALLS
MEAN: 30,822 dpm/100cm2
MAX.: 4,951,630 dpm/100cm2
SIGMA: 266,922 dpm/100cm2
358 MEASUREMENTS

DATA SUMMARY
FLOOR/LOWER WALLS
MEAN: 67,498 dpm/100cm2
MAX.: 5,720,000 dpm/100cm2
SIGMA: 740,452 dpm/100cm2
189 MEASUREMENTS

DATA SUMMARY
FLOOR/LOWER WALLS
MEAN: 67,489 dpm/100cm2
MAX.: 3,996,000 dpm/100cm2
SIGMA: 328,325 dpm/100cm2
190 MEASUREMENTS

DATA SUMMARY
CEILING/UPPER WALLS
MEAN: 154,926 dpm/100cm2
MAX.: 5,671,037 dpm/100cm2
SIGMA: 589,742 dpm/100cm2
299 MEASUREMENTS

DATA SUMMARY
FLOOR/LOWER WALLS
MEAN: 209,175 dpm/100cm2
MAX.: 1,804,000 dpm/100cm2
SIGMA: 342,428 dpm/100cm2
149 MEASUREMENTS

DATA SUMMARY
FLOORS/CEILING/WALLS
MEAN: 2897 dpm/100cm2
MAX.: 4920 dpm/100cm2
SIGMA: 620 dpm/100cm2
40 MEASUREMENTS

NOTE:
MEASUREMENTS INDICATED ARE THE HIGHEST/LOGGED VALUES OF THE GRIDS SCANNED.

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UTILITY DISTRICT

Figure 2-100
Auxiliary Building El.-47
813000

DWG: 813000.01

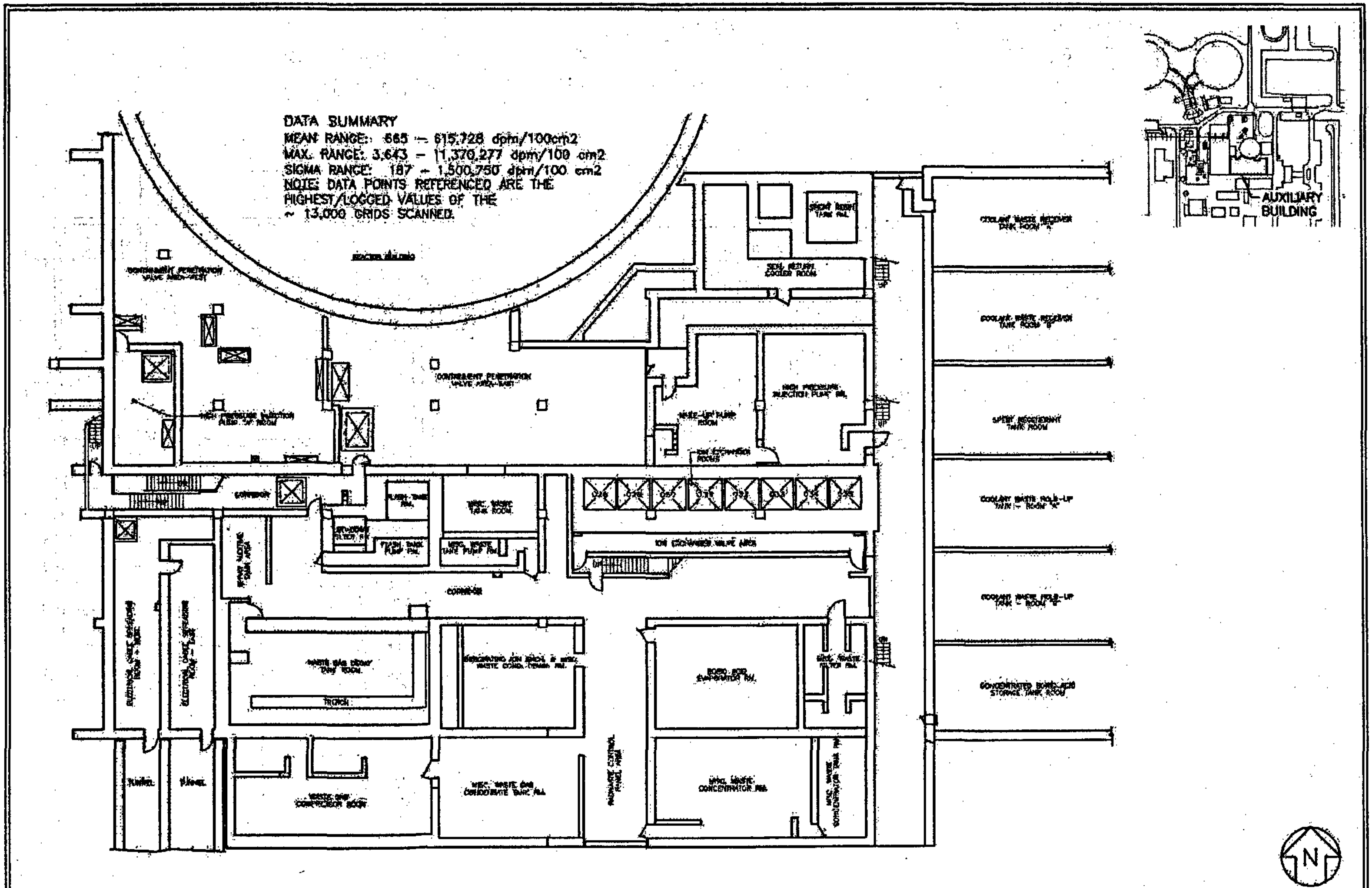

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Figure 2-101
Auxiliary Building El.-20/-29
813000

DWG: 813000.02

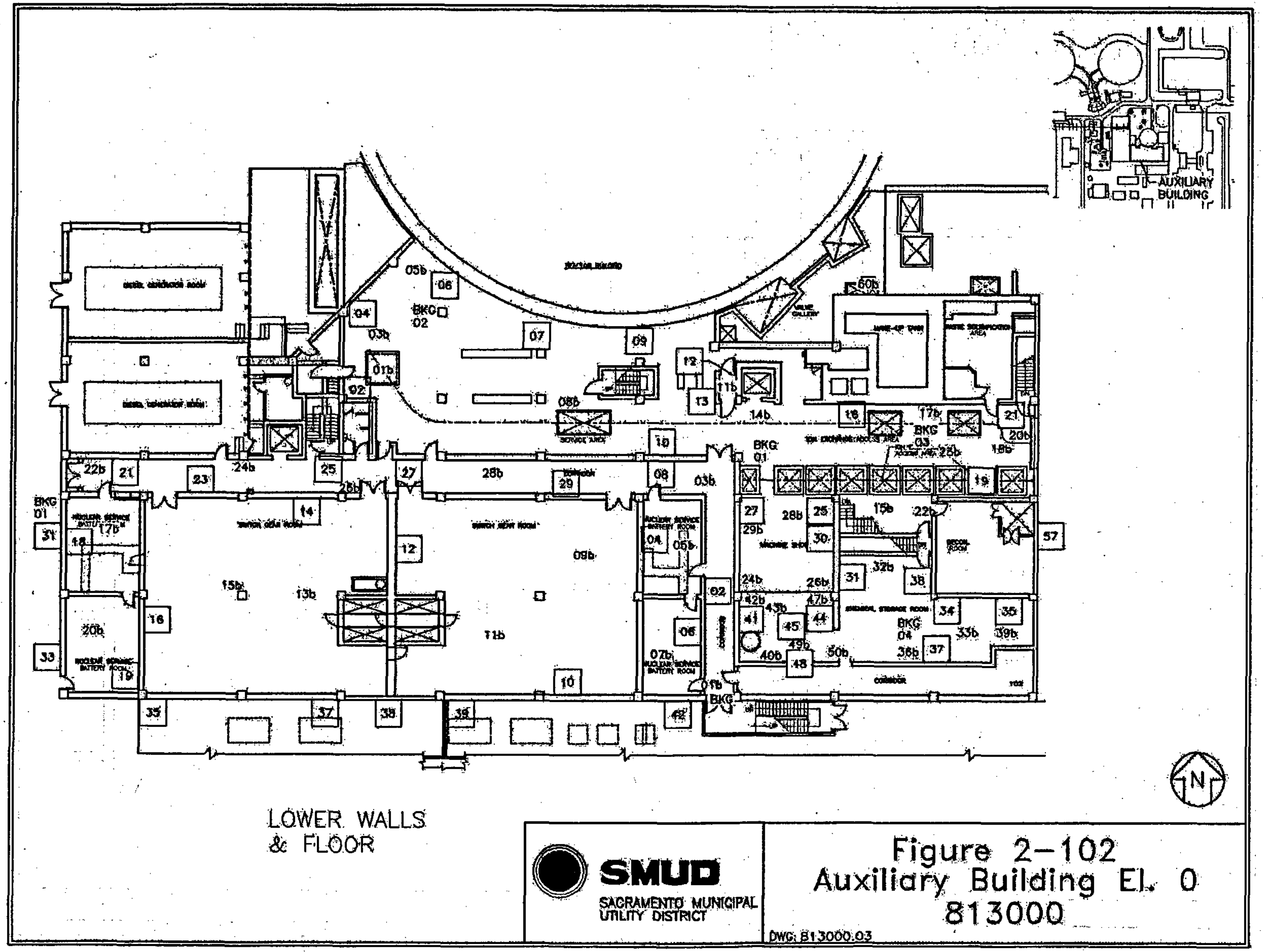

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Figure 2–102
Auxiliary Building El. 0
813000

DWG: 813000.03

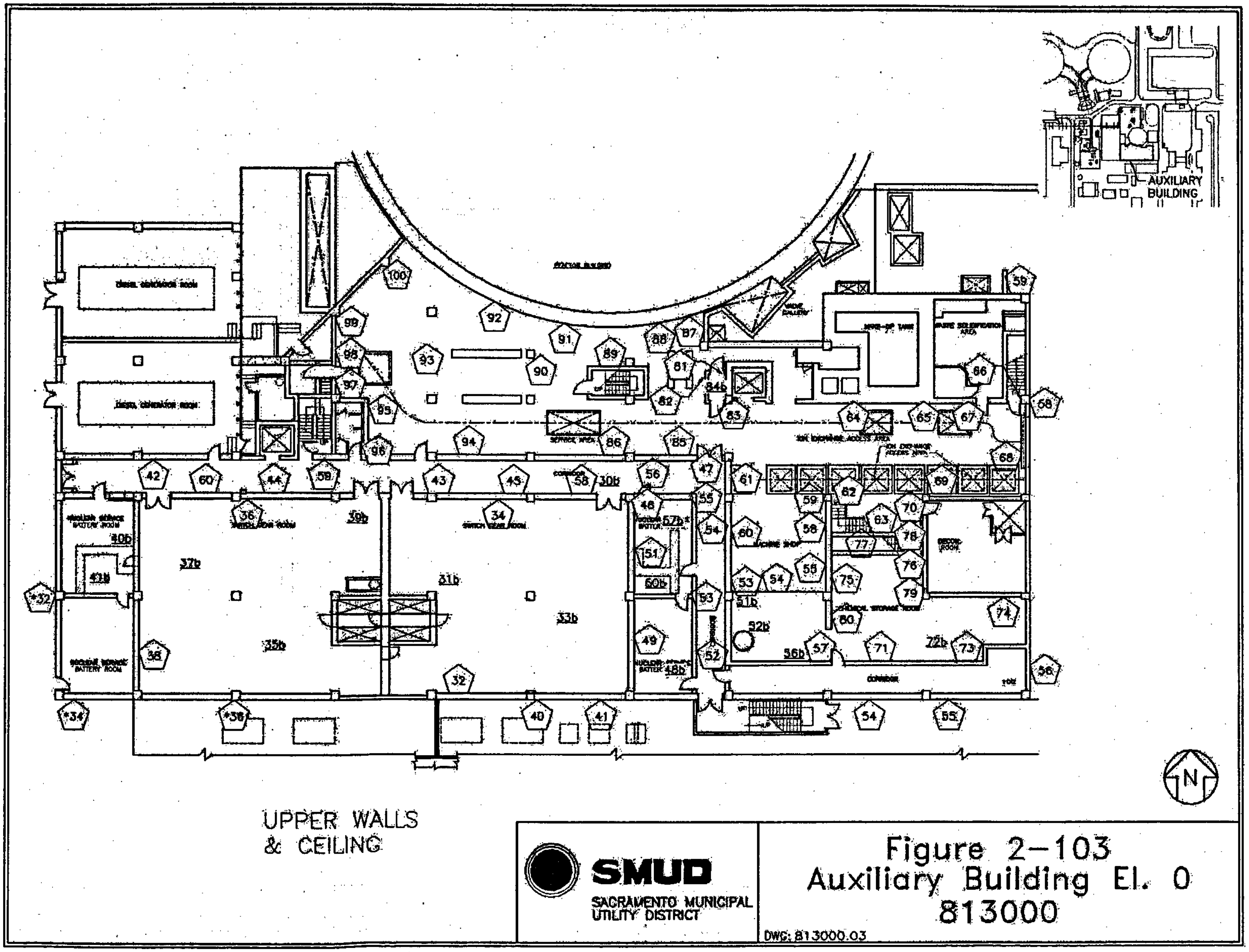
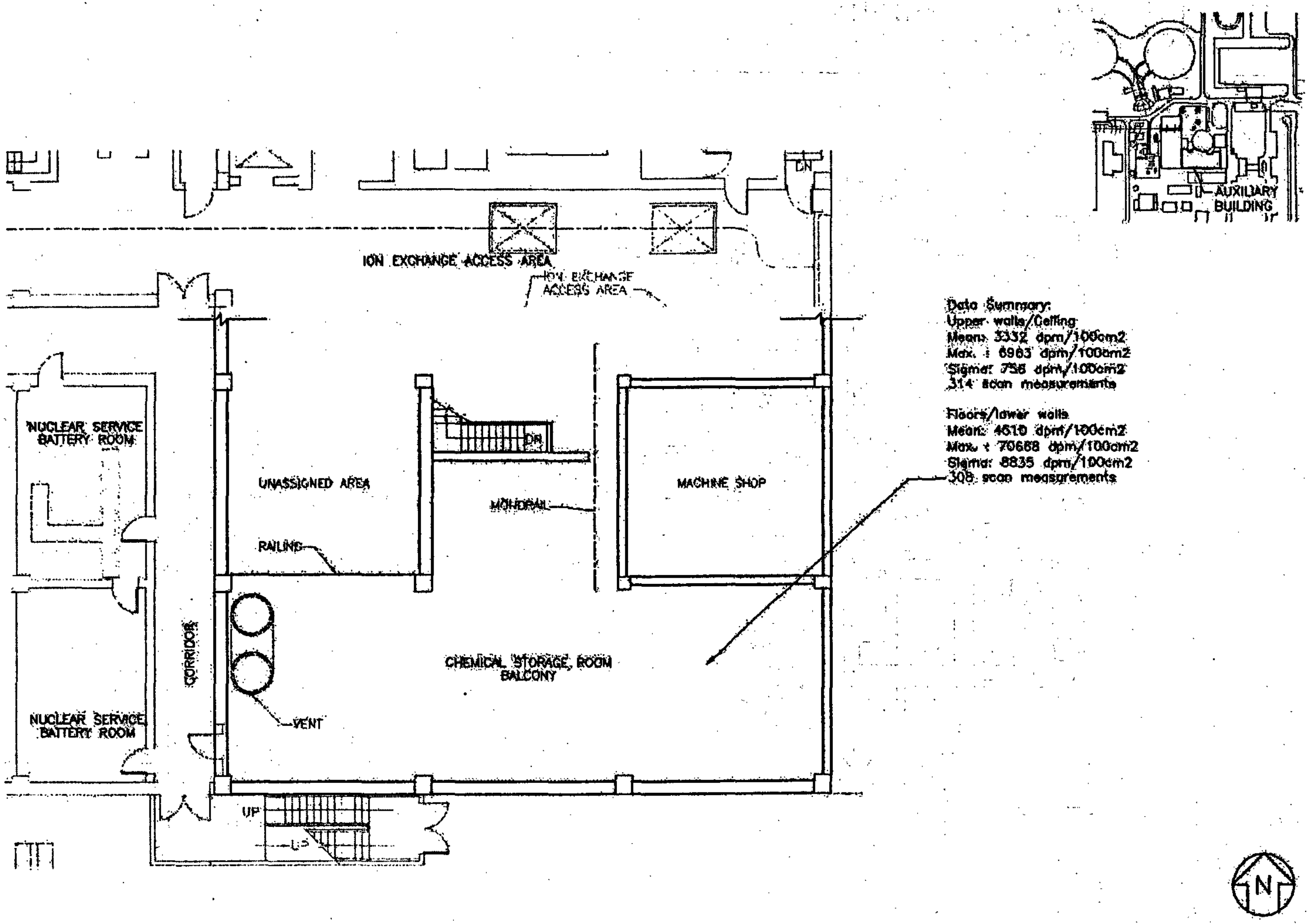


Figure 2–103
Auxiliary Building El. 0
813000



Data Summary:
Upper walls/Ceiling
Mean: 3332 dpm/100cm2
Max. : 6963 dpm/100cm2
Sigma: 758 dpm/100cm2
314 scan measurements

Floors/lower walls
Mean: 4610 dpm/100cm2
Max. : 70668 dpm/100cm2
Sigma: 8835 dpm/100cm2
308 scan measurements

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Figure 2-104
Auxiliary Building El. 10
813000

DWG: 813000.04

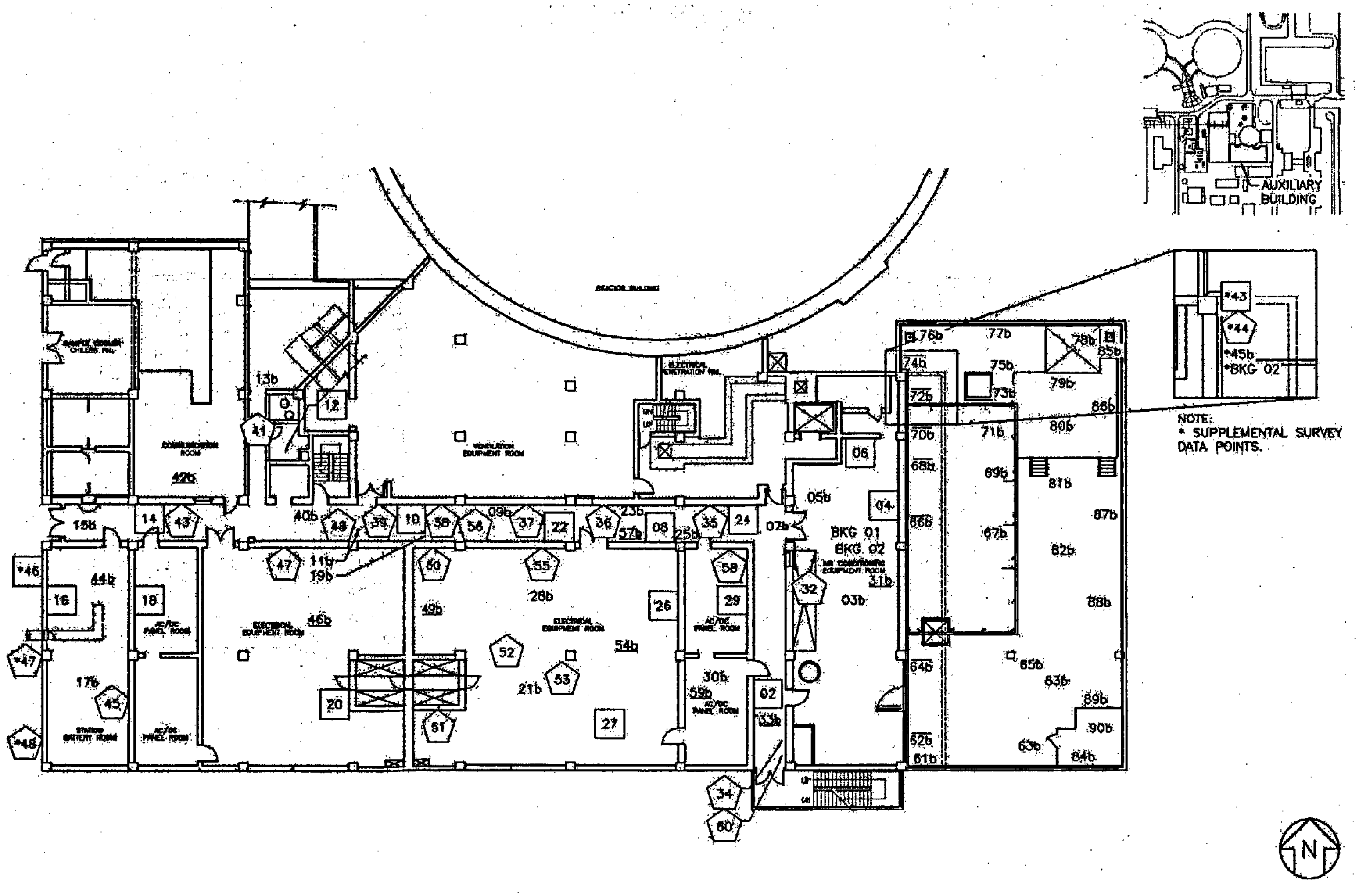

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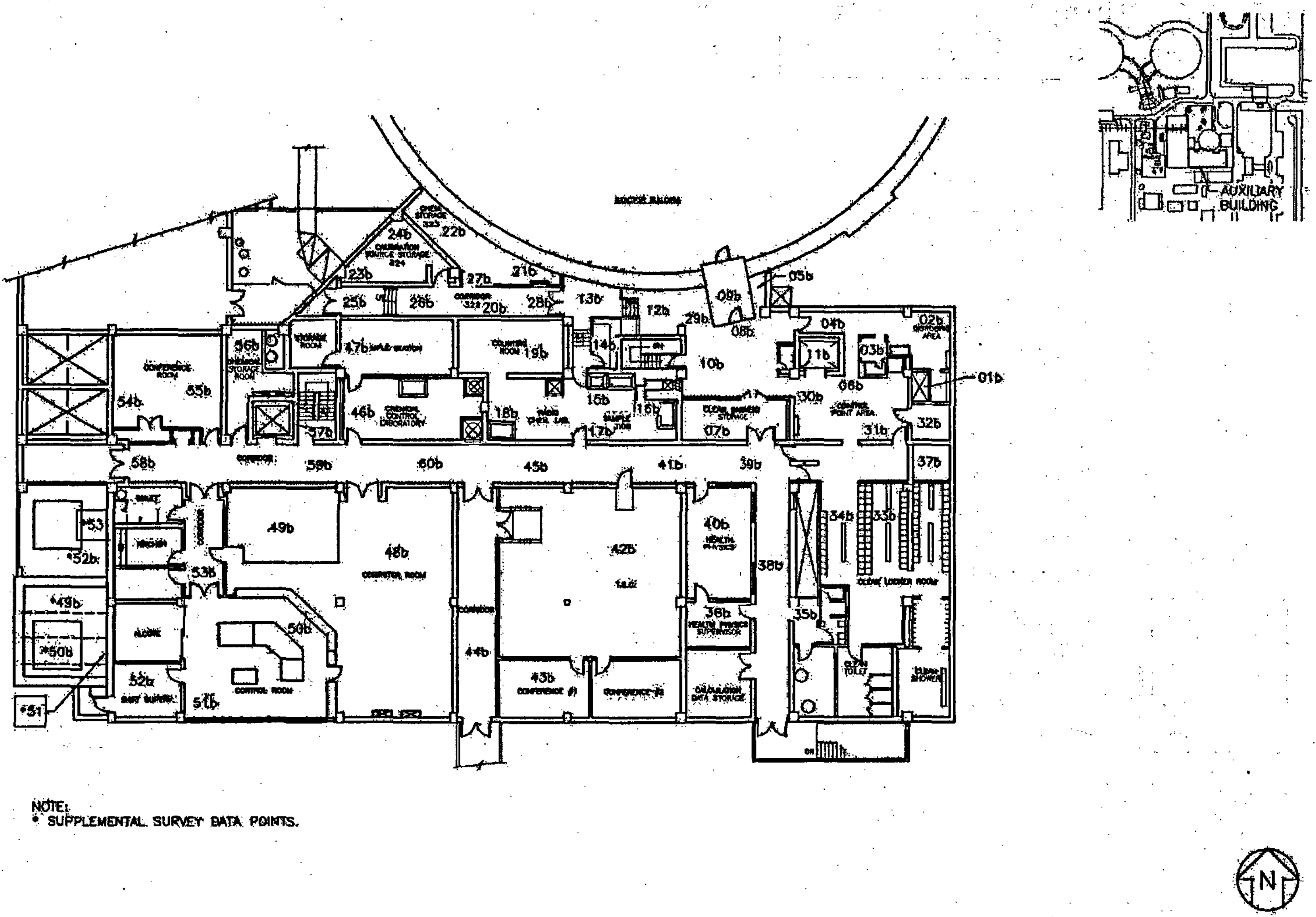
Figure 2–105
Auxiliary Building El. 20
813000

DWG: 813000.05



NOTE:
* SUPPLEMENTAL SURVEY DATA POINTS.

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Figure 2-106
Auxiliary Building El. 40
813000

DWG: 813000.06

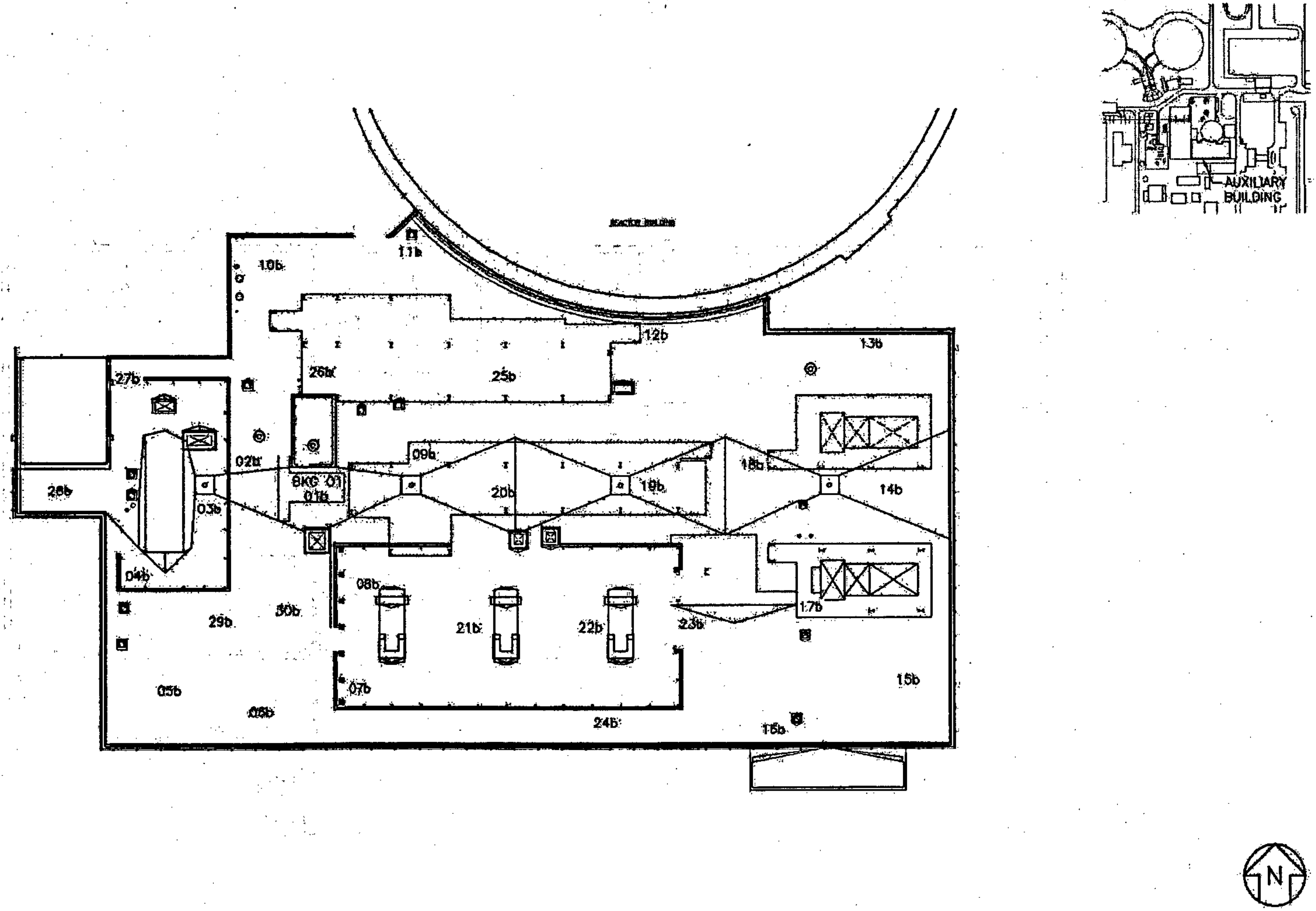

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Figure 2–107
Auxiliary Building Roof
813000

DWG: 813000.07

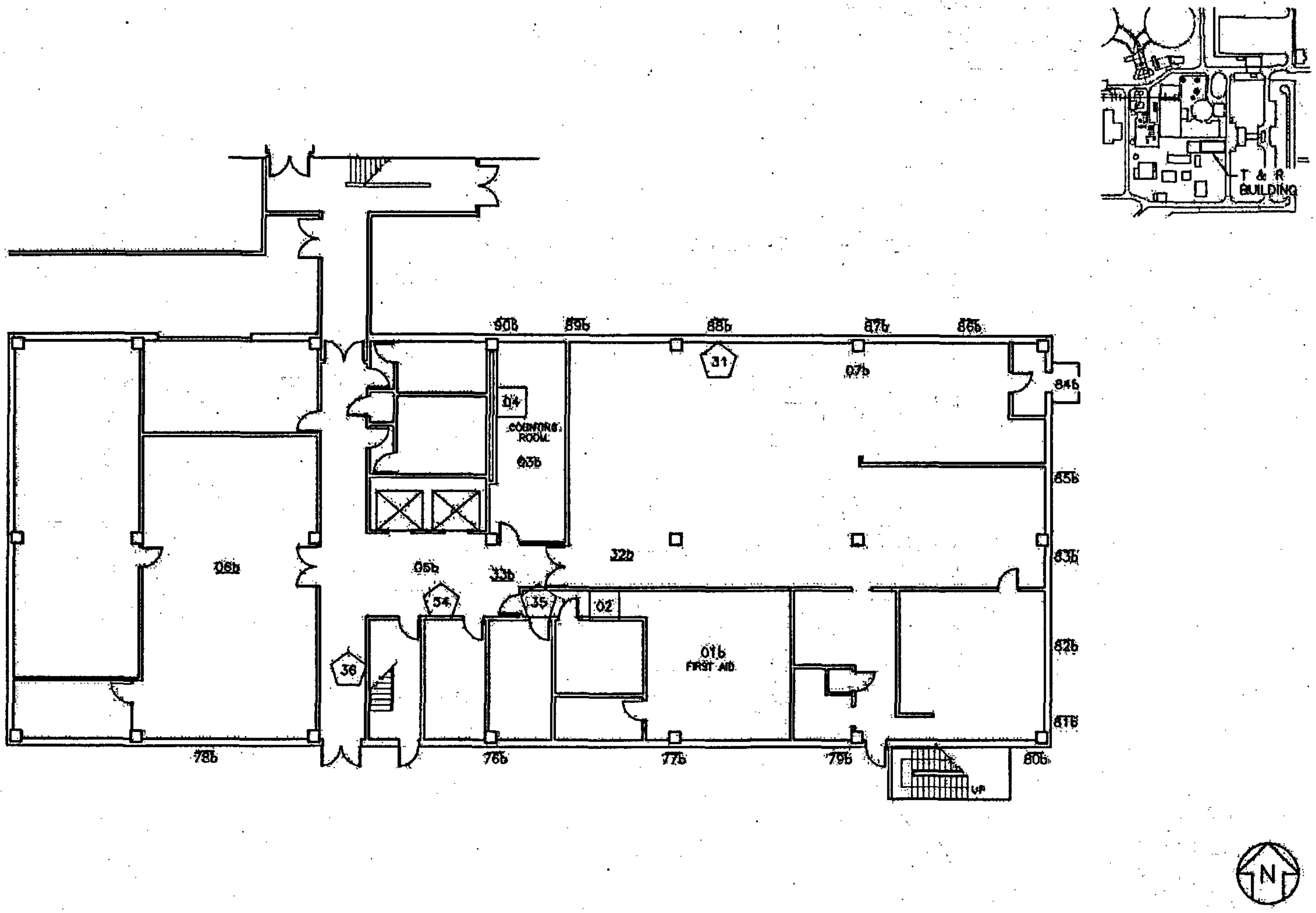

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Figure 2-108
T & R Building El. 0
814000

DWG: 814000.01

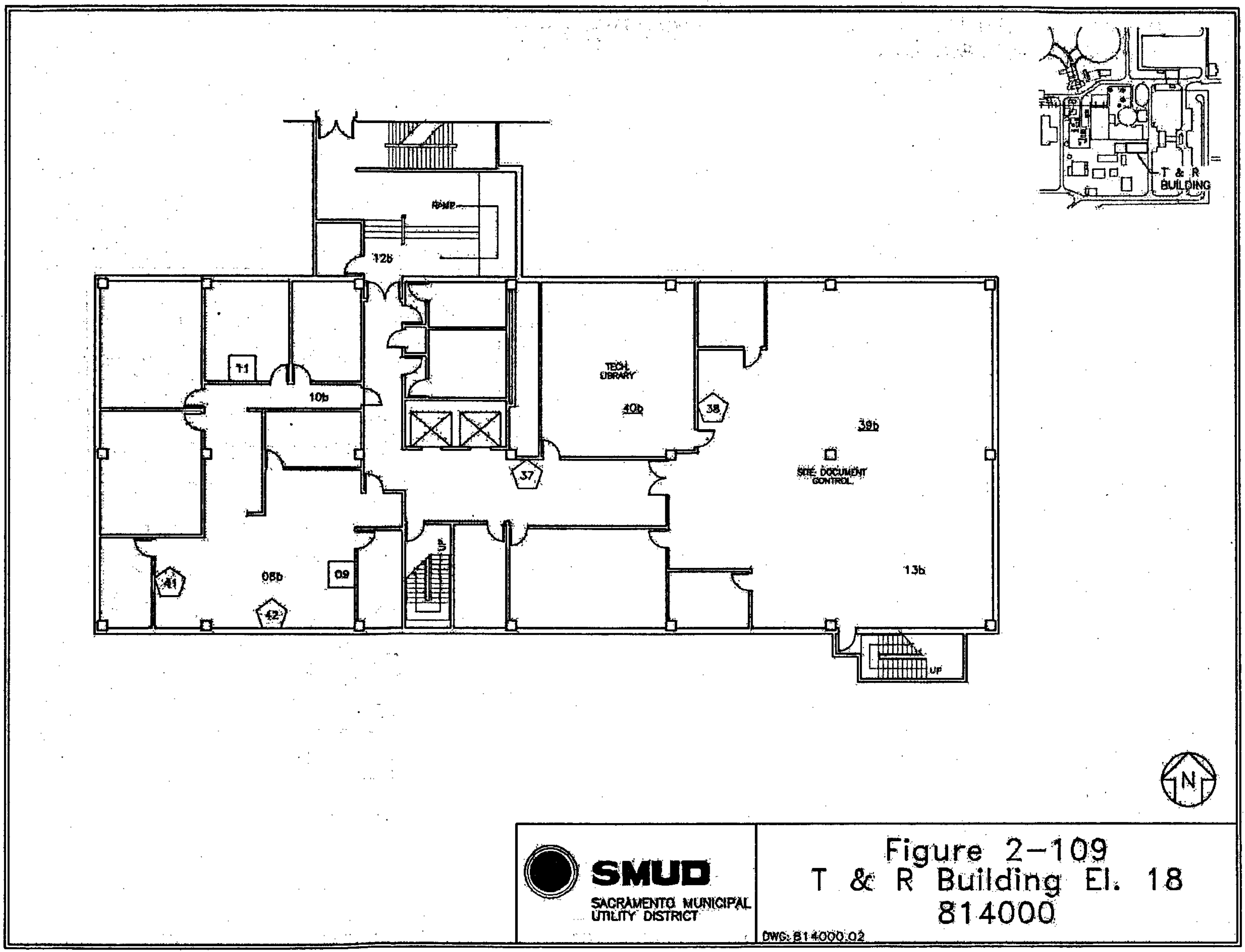

Figure 2-109
T & R Building El. 18
814000

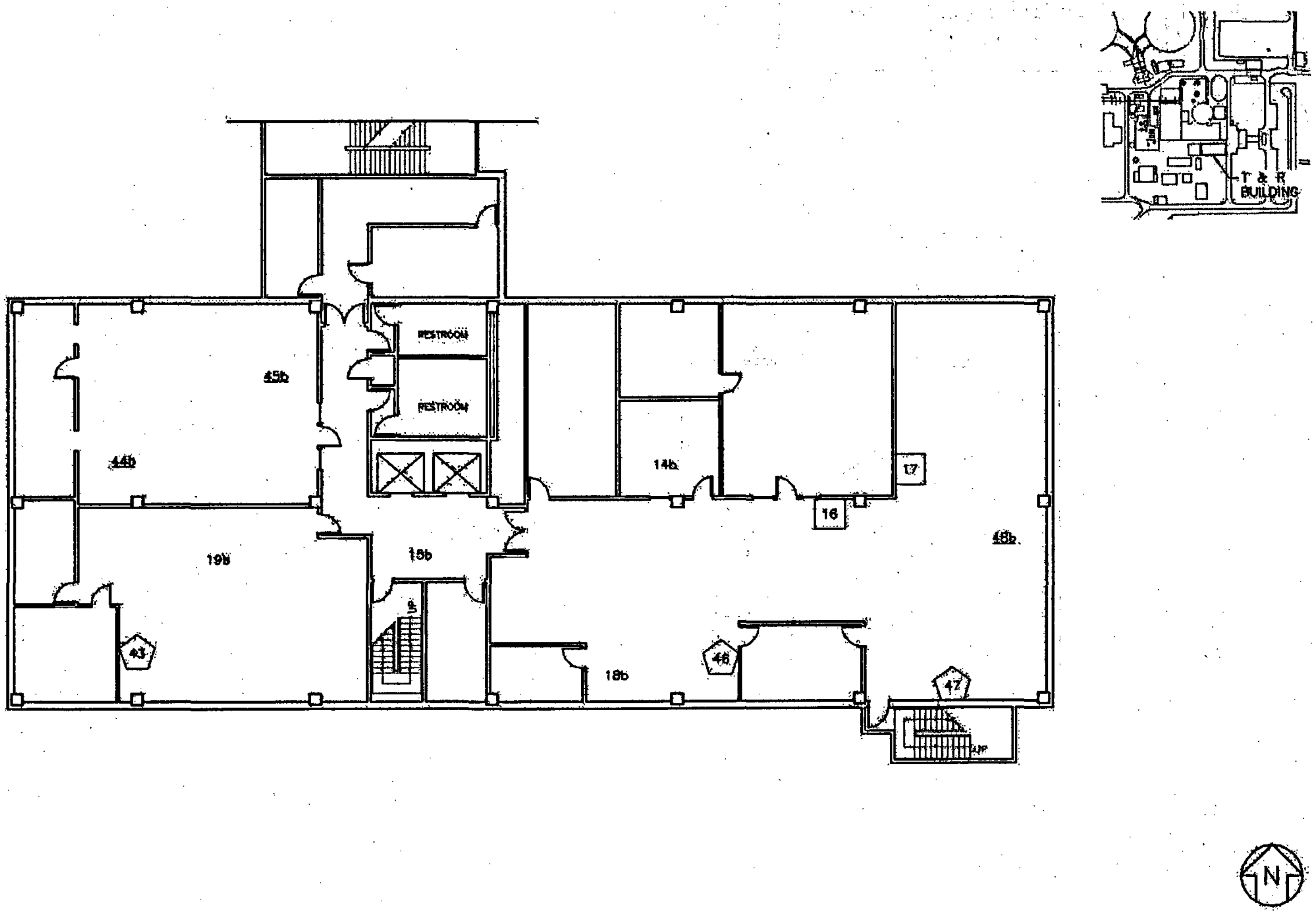

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Figure 2-110
T & R Building El. 30
814000

DWG: 814000.03

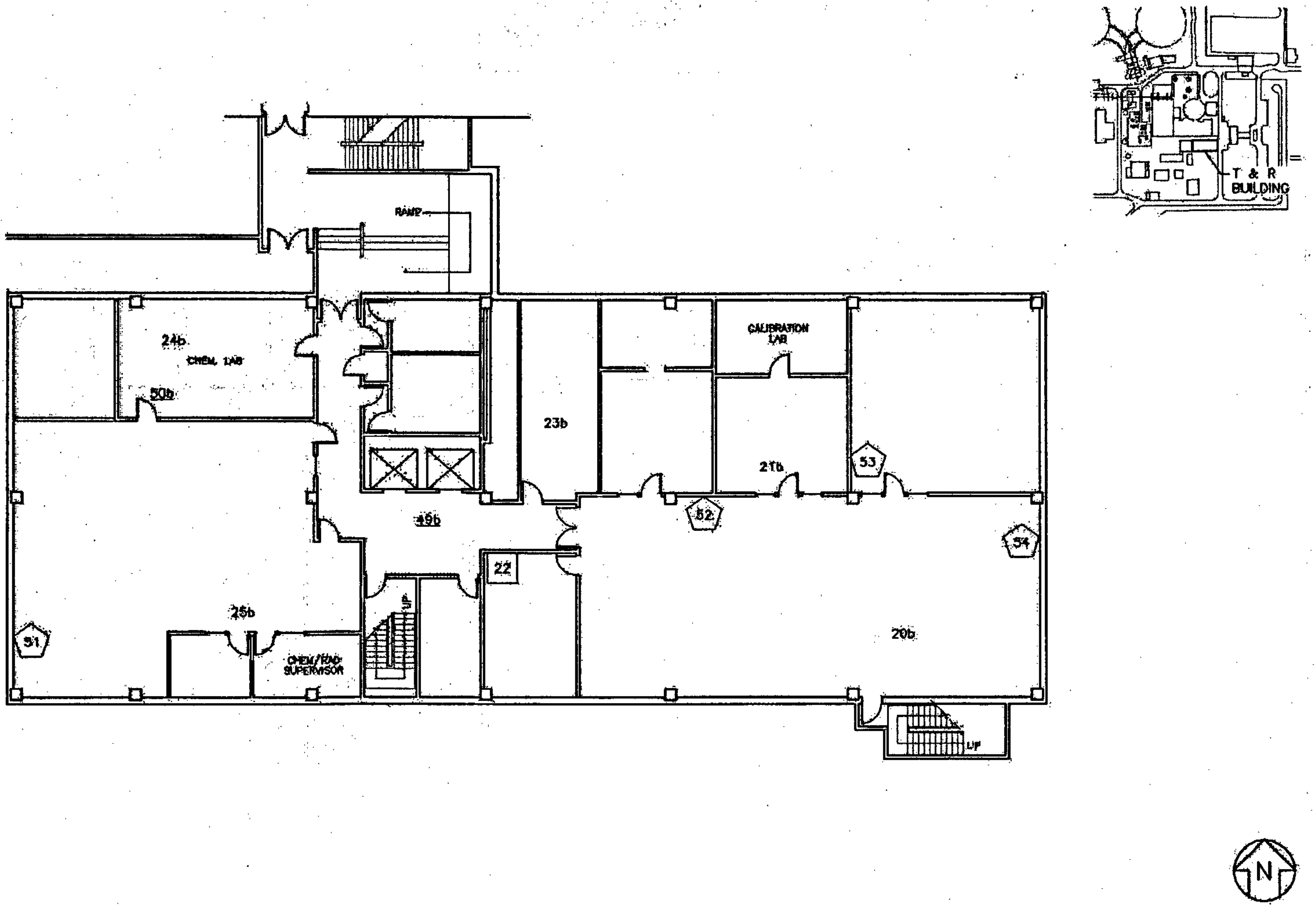

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Figure 2-111
T & R Building El. 42
814000

DWG: 814000.04

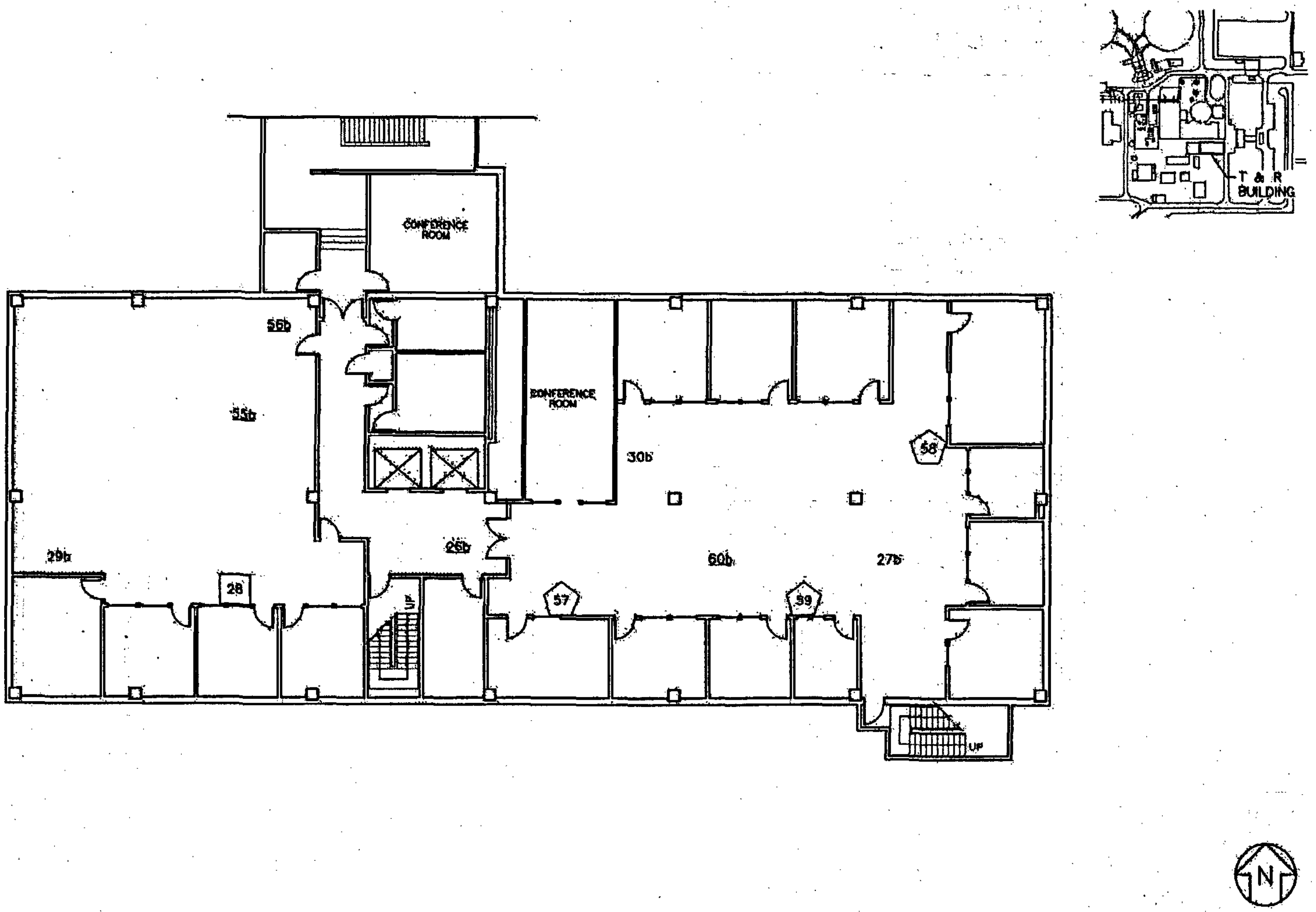

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Figure 2-112
T & R Building El. 54
814000

DWG: 814000.05

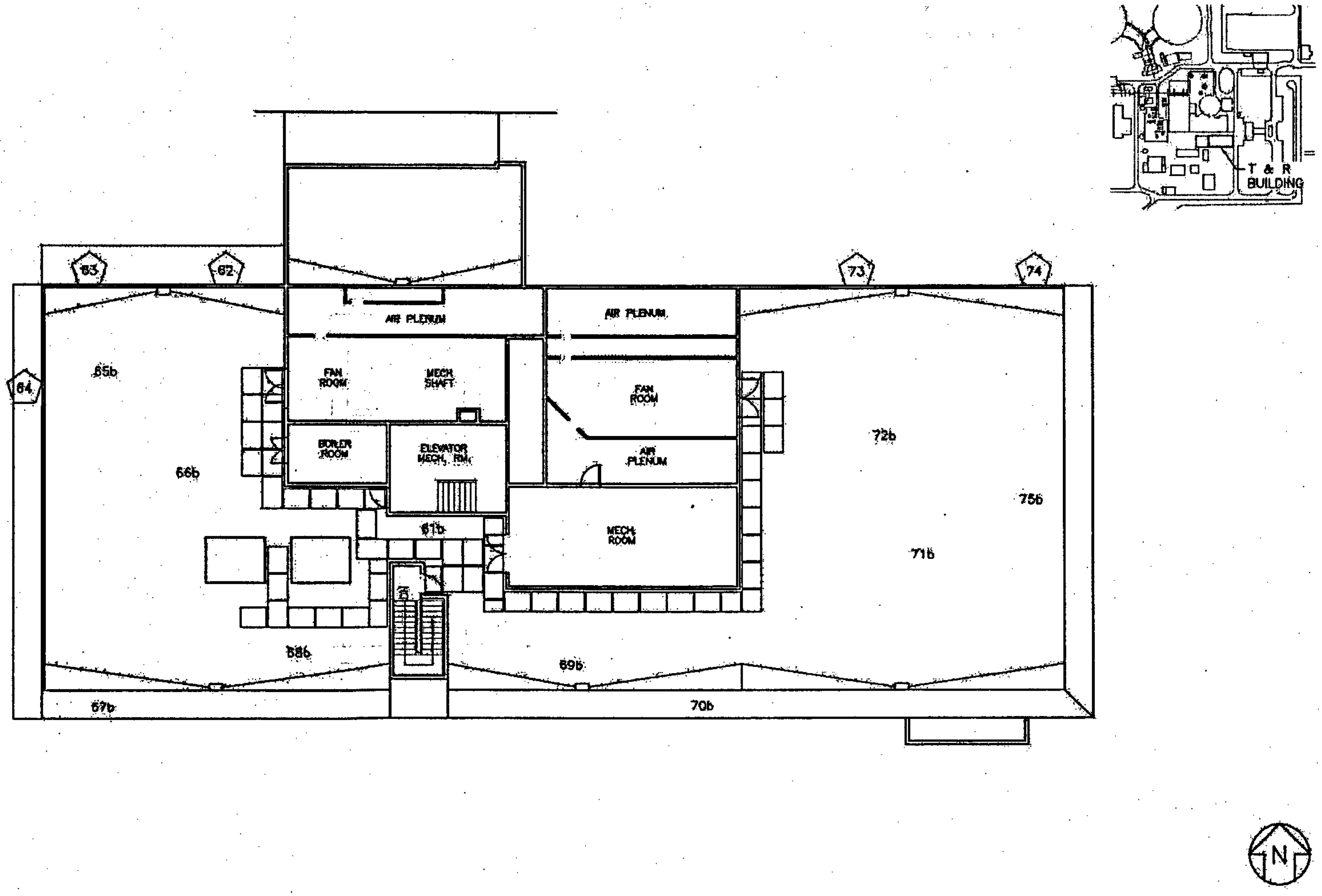

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Figure 2-113
T & R Building Roof
814000

DWG: 814000.06

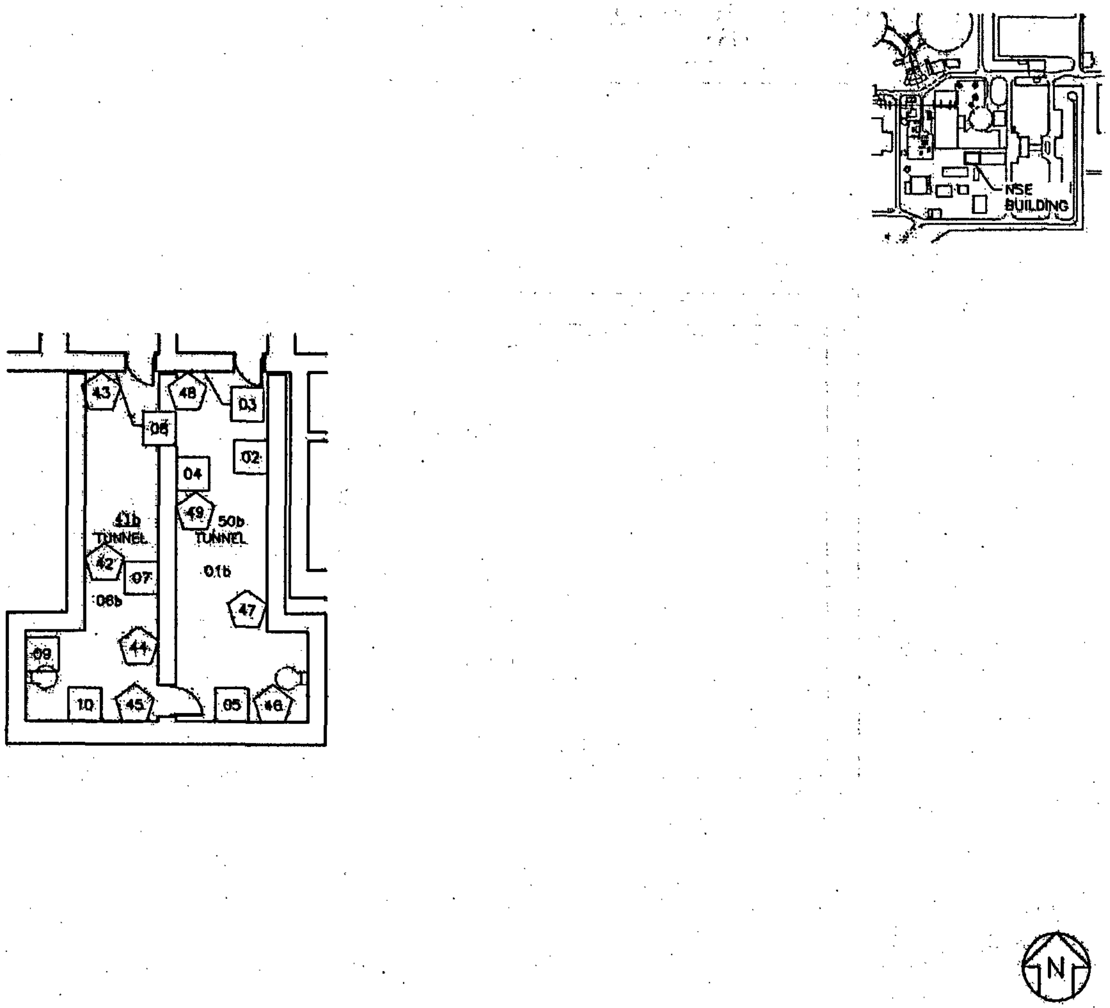

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Figure 2-114
NSEB El. -20
815000

DWG 815000.01

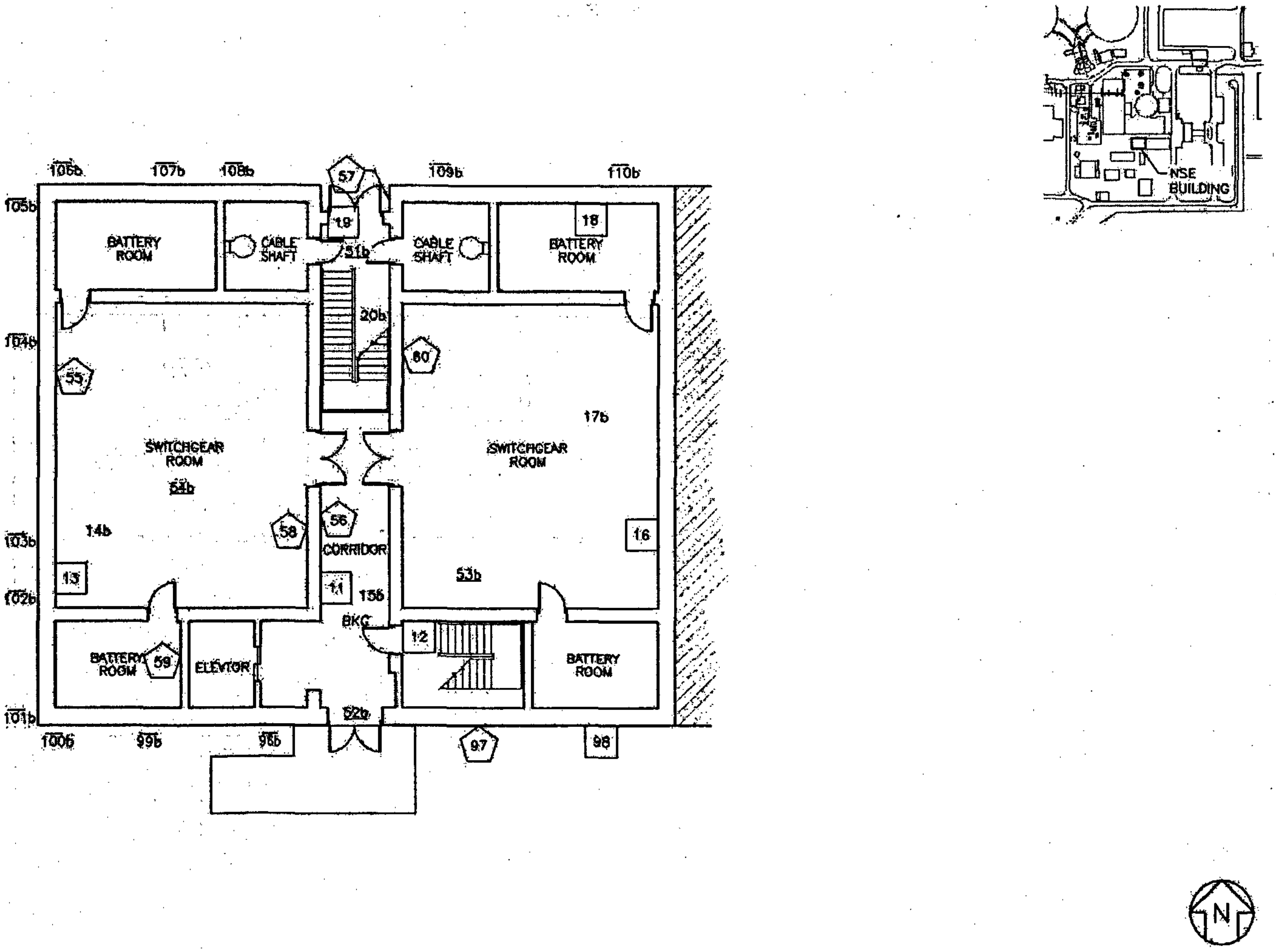

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Figure 2-115
NSEB El. 0
815000

DWG 815000.02

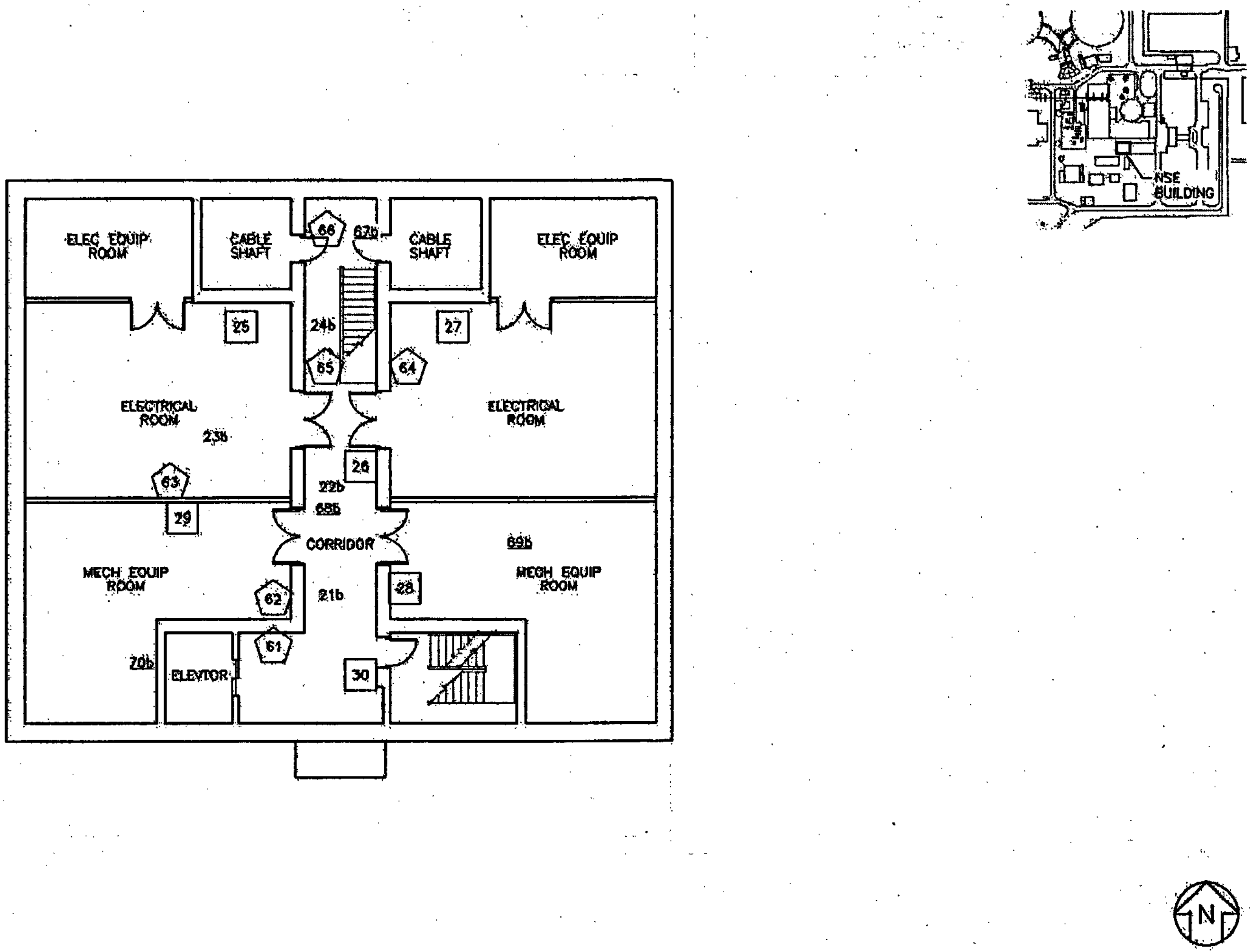

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Figure 2-116
NSEB El. 20
815000

DWG: 815000.03

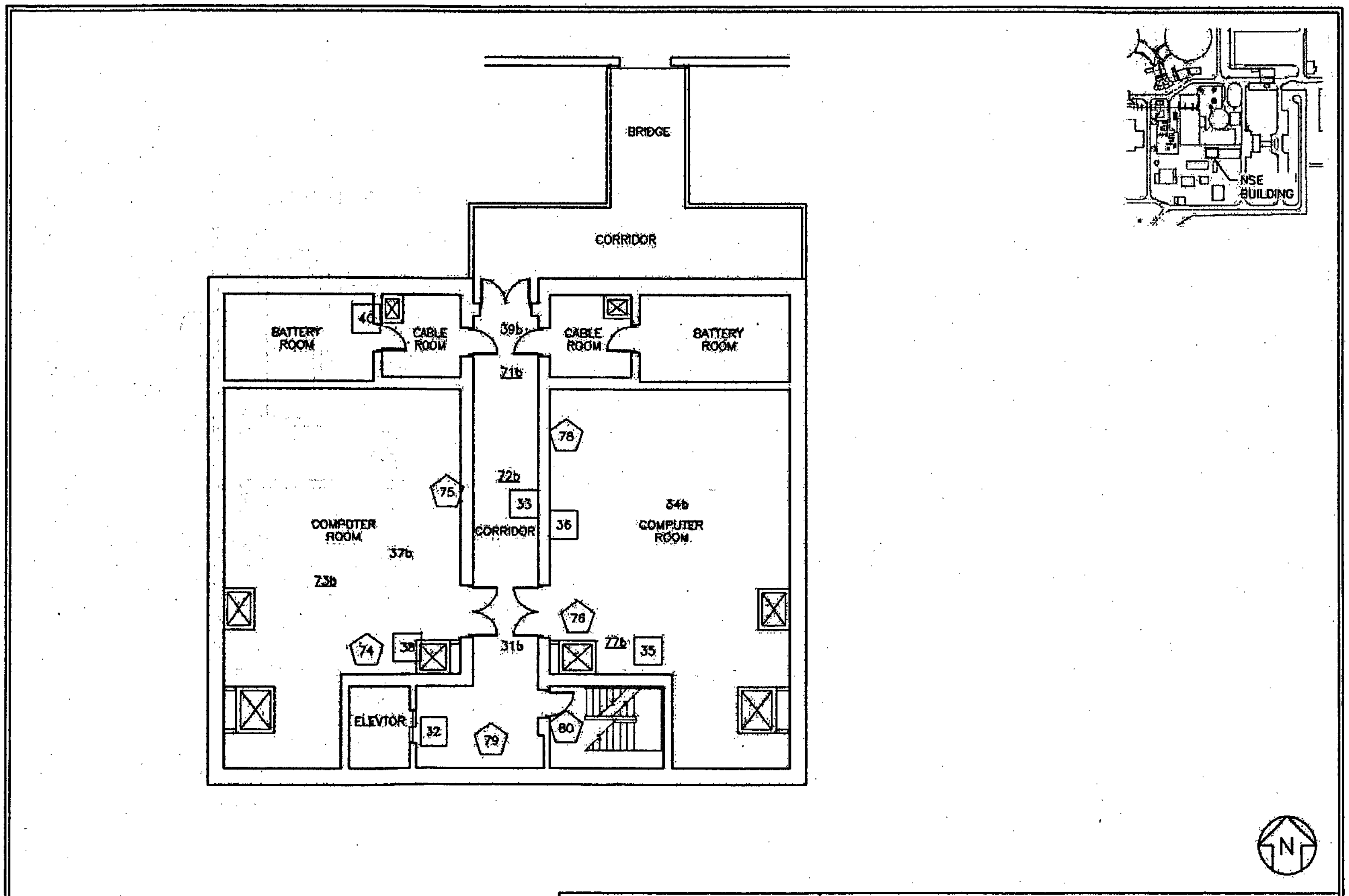

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Figure 2-117
NSEB El. 40
815000

DWG. 815000.04

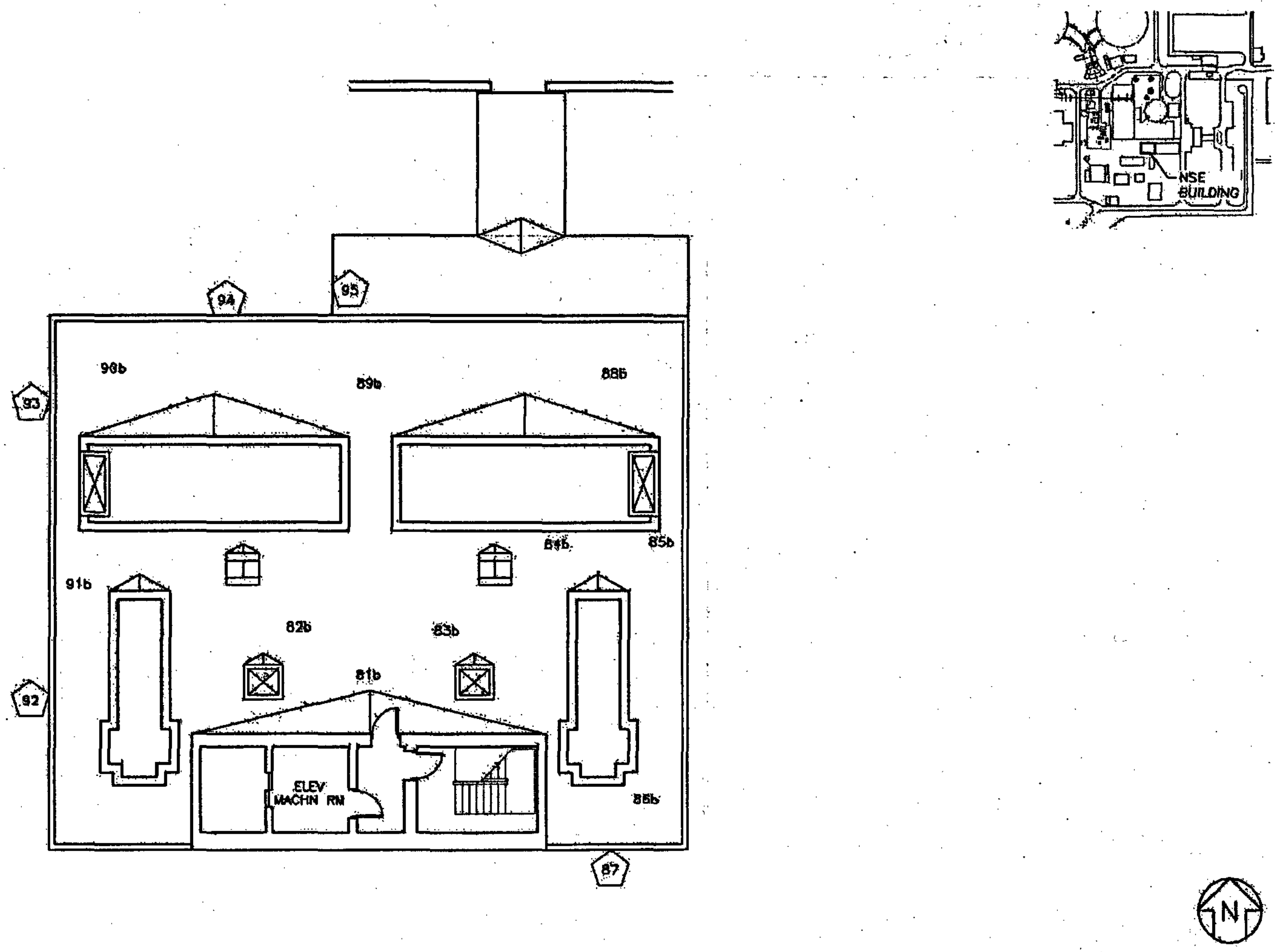

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Figure 2-118
NSEB Roof
815000

DWG: 815000.05

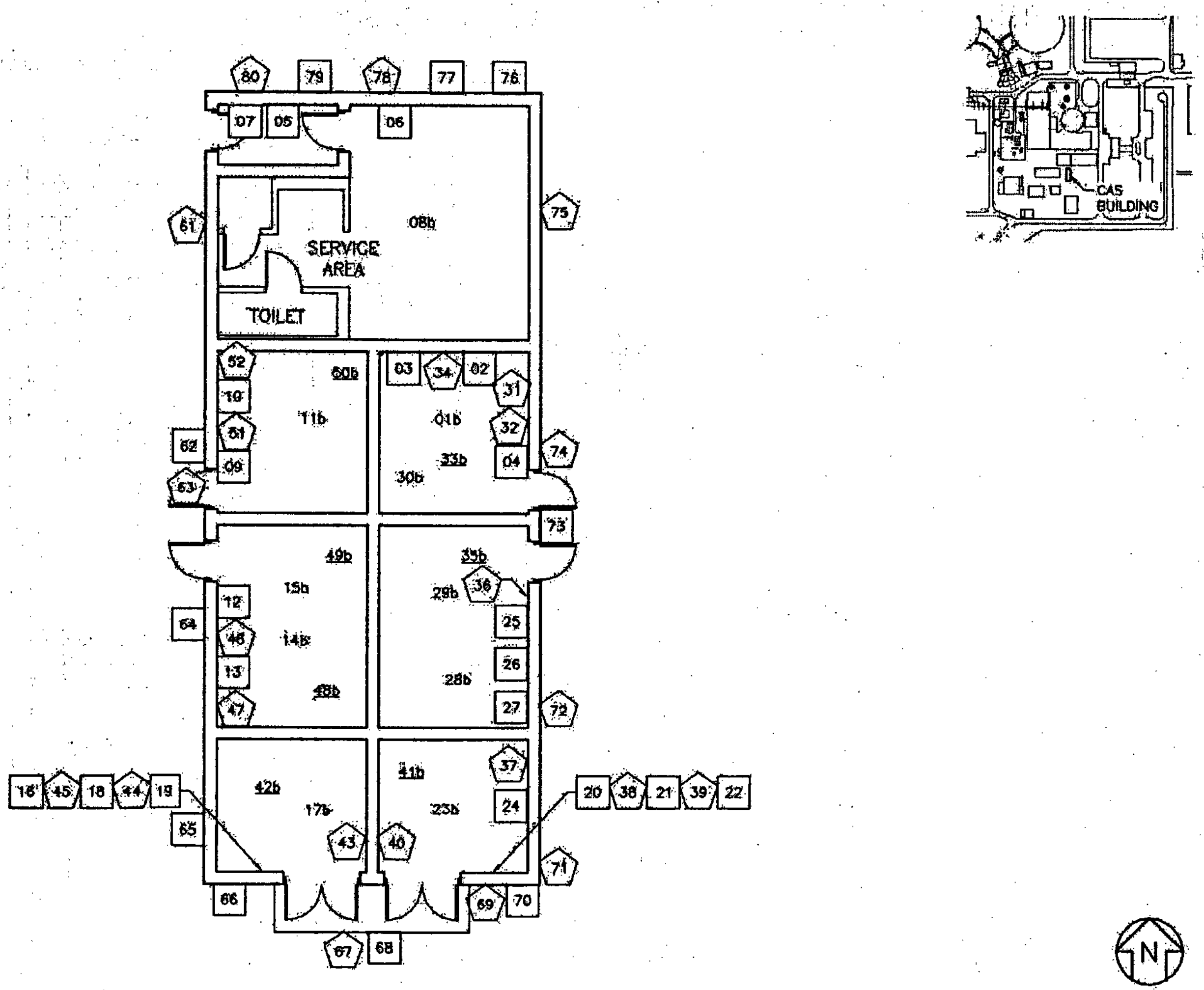

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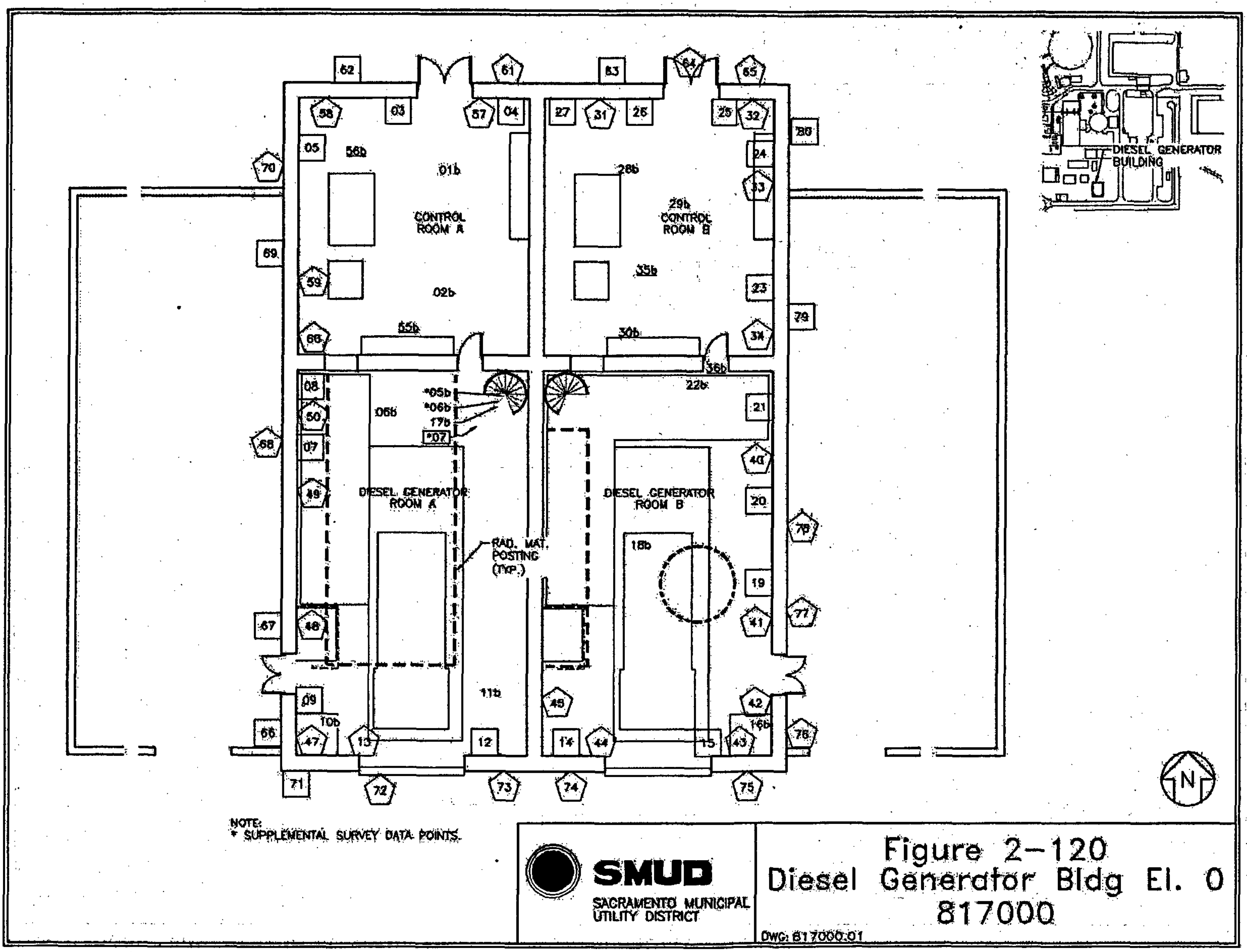
Figure 2-119
CAS Building El. 0
816000

DWG: 816000.01



NOTE:
* SUPPLEMENTAL SURVEY DATA POINTS.

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-120
Diesel Generator Bldg El. 0
817000

DWG: 817000.01

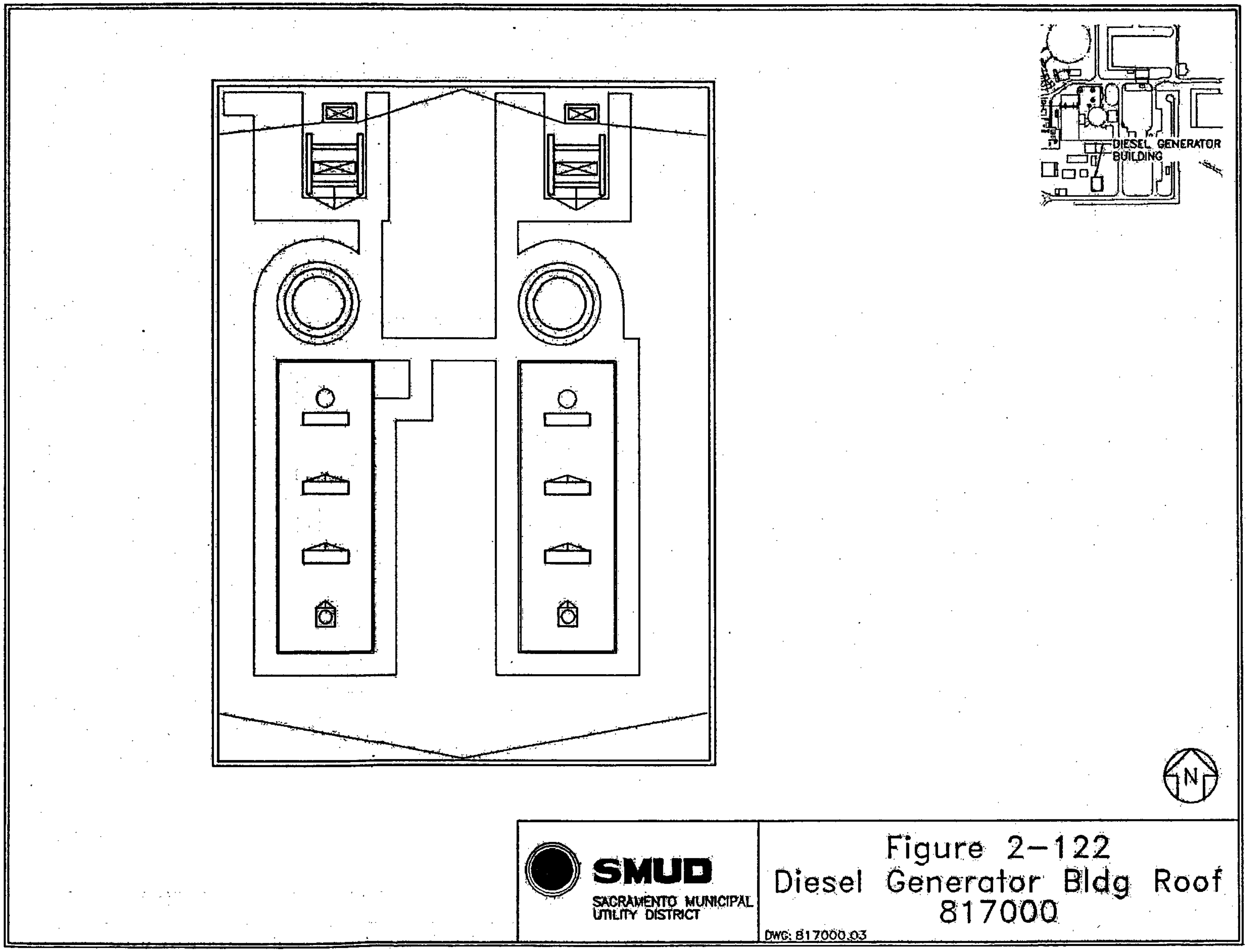


Figure 2-122
Diesel Generator Bldg Roof
817000

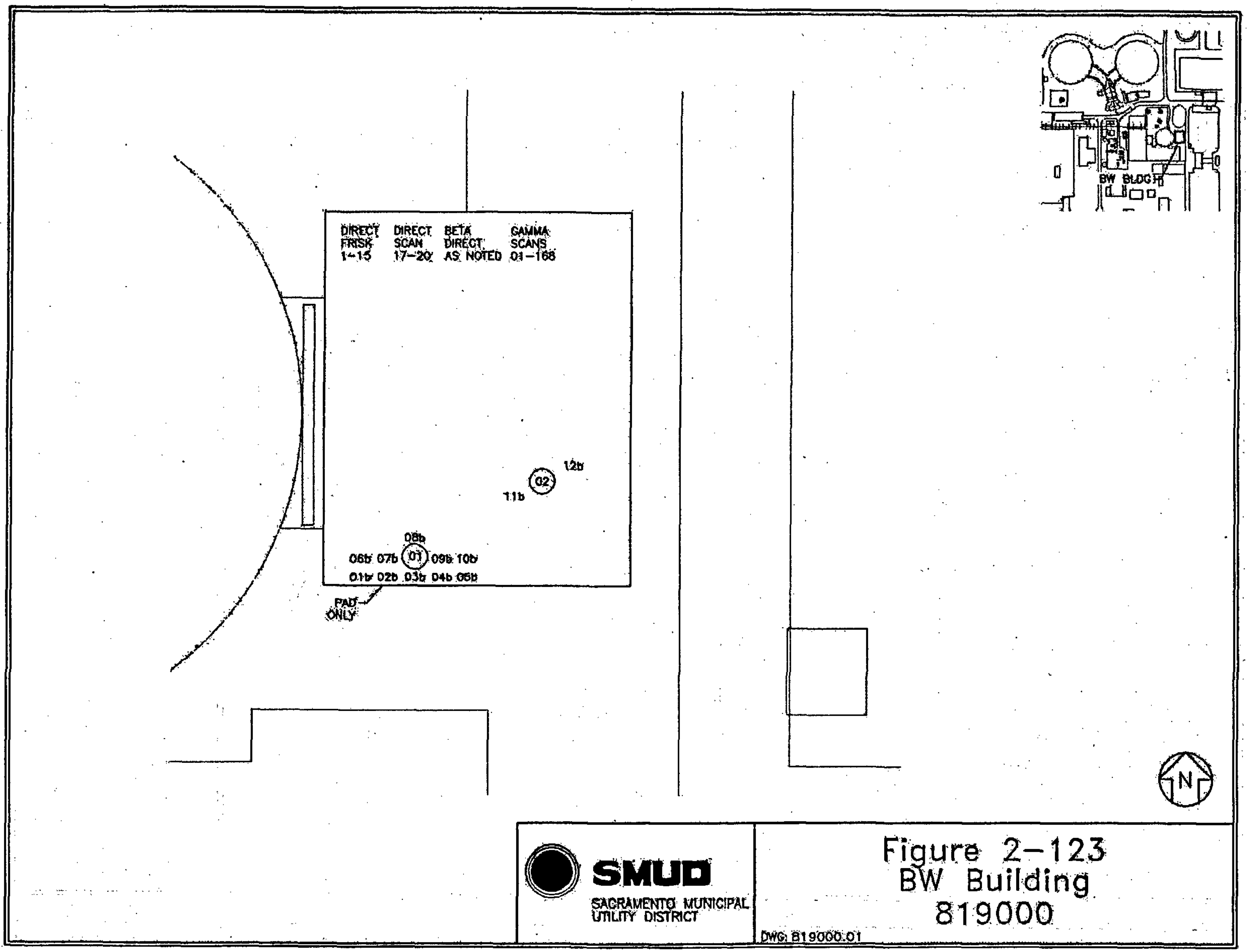
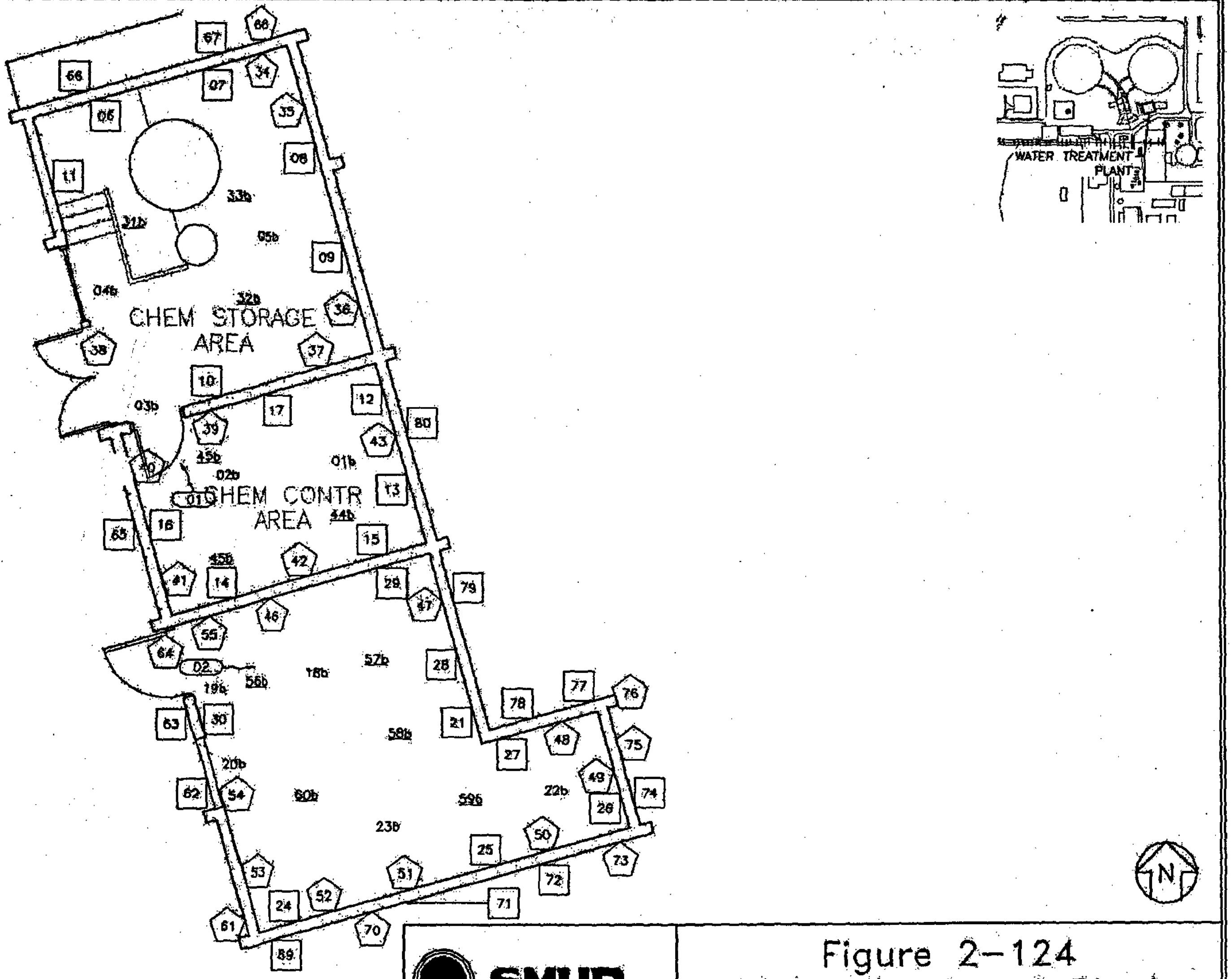


Figure 2-123
BW Building
819000



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-124
Water Treatment Plant
821000

DWG: 821000.01

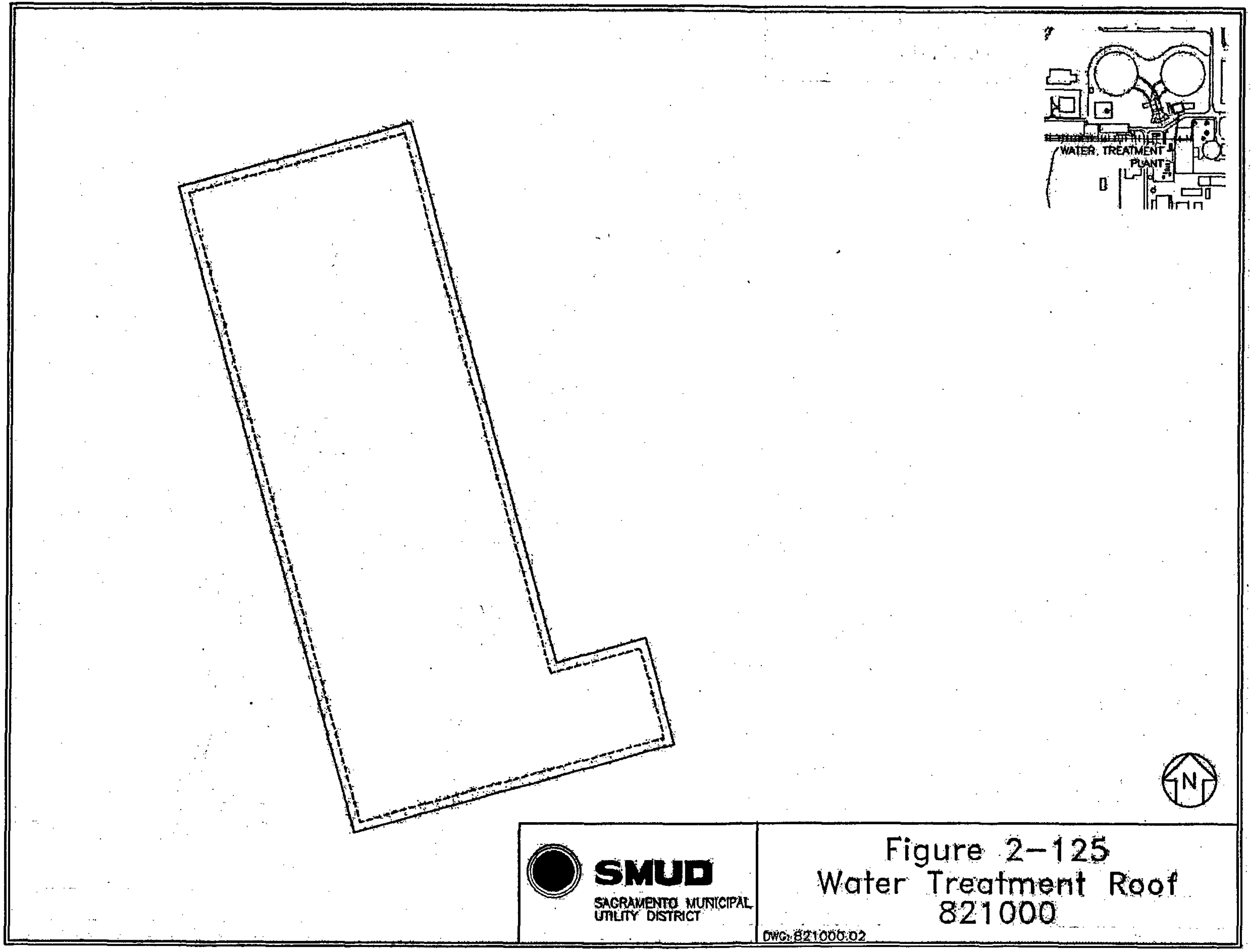


Figure 2-125
Water Treatment Roof
821000

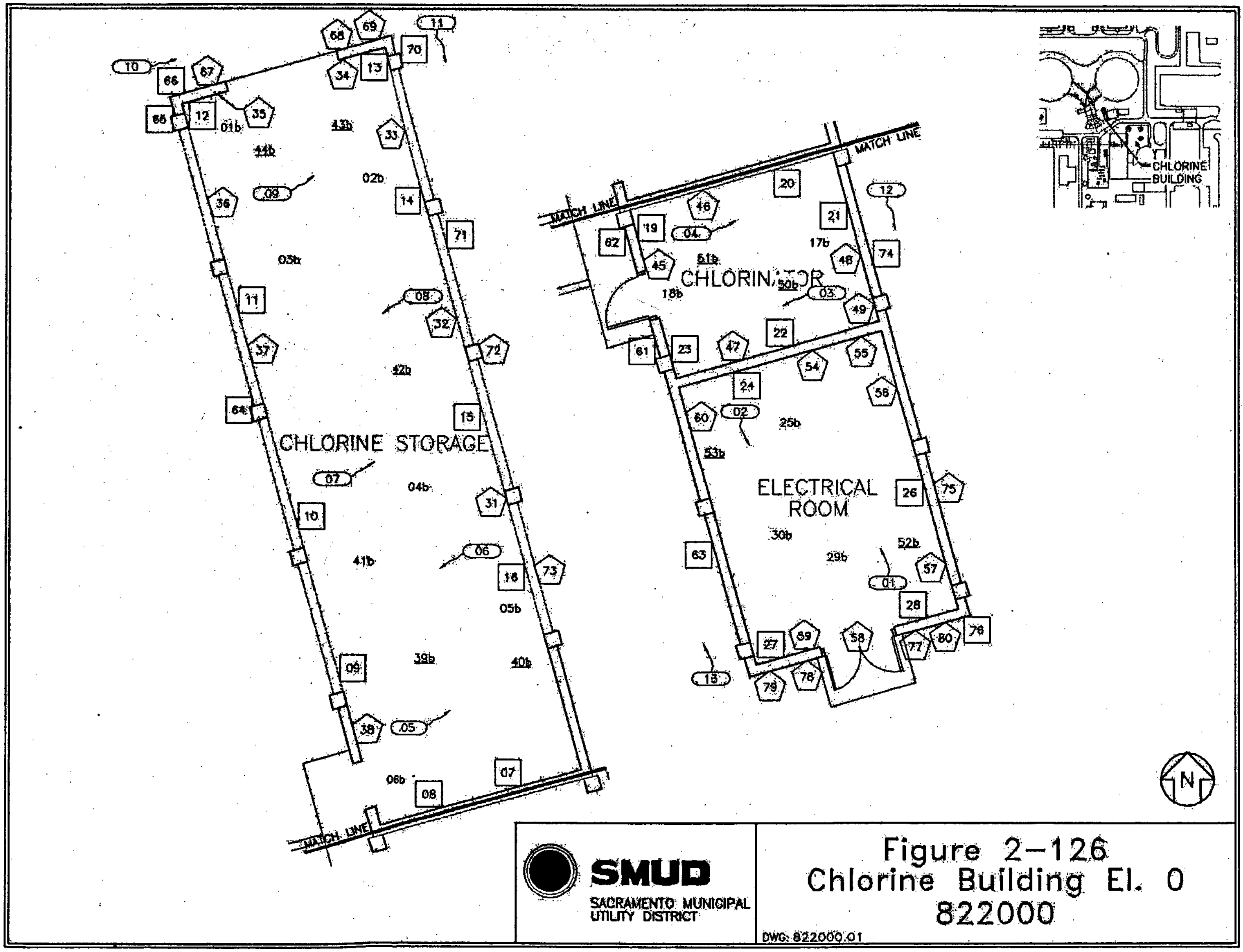
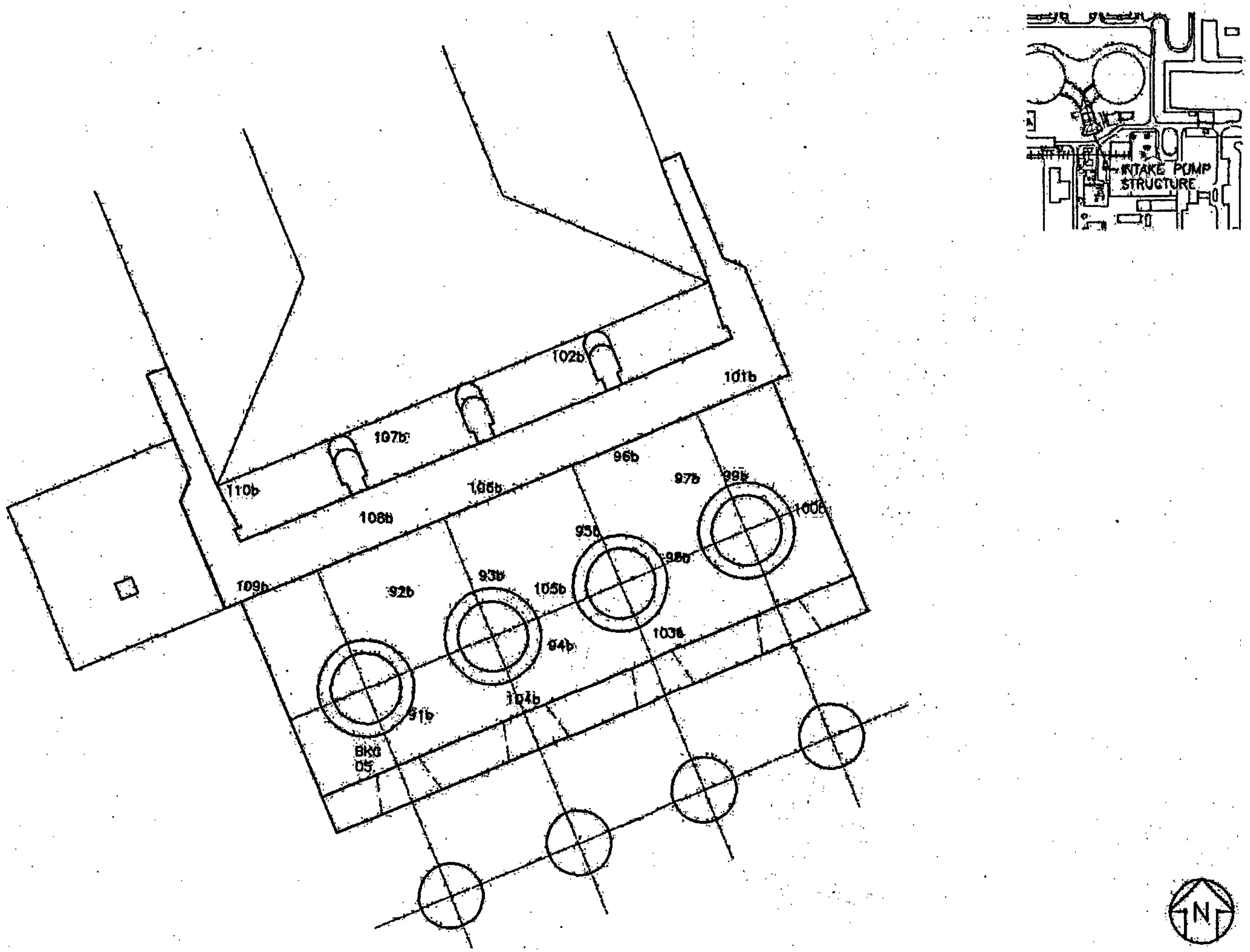


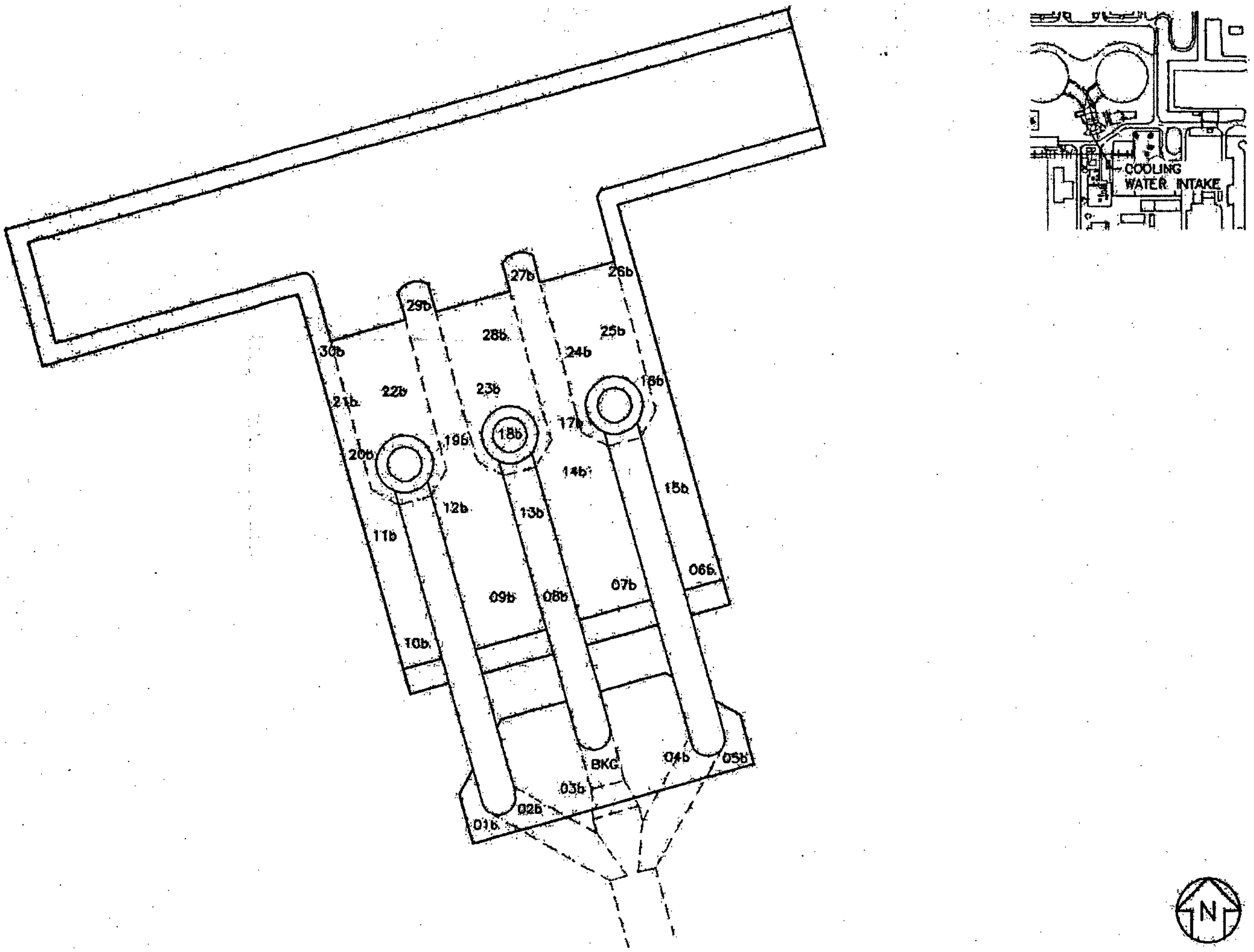
Figure 2-126
Chlorine Building El. 0
822000



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-127
Intake Pump Structure
823000

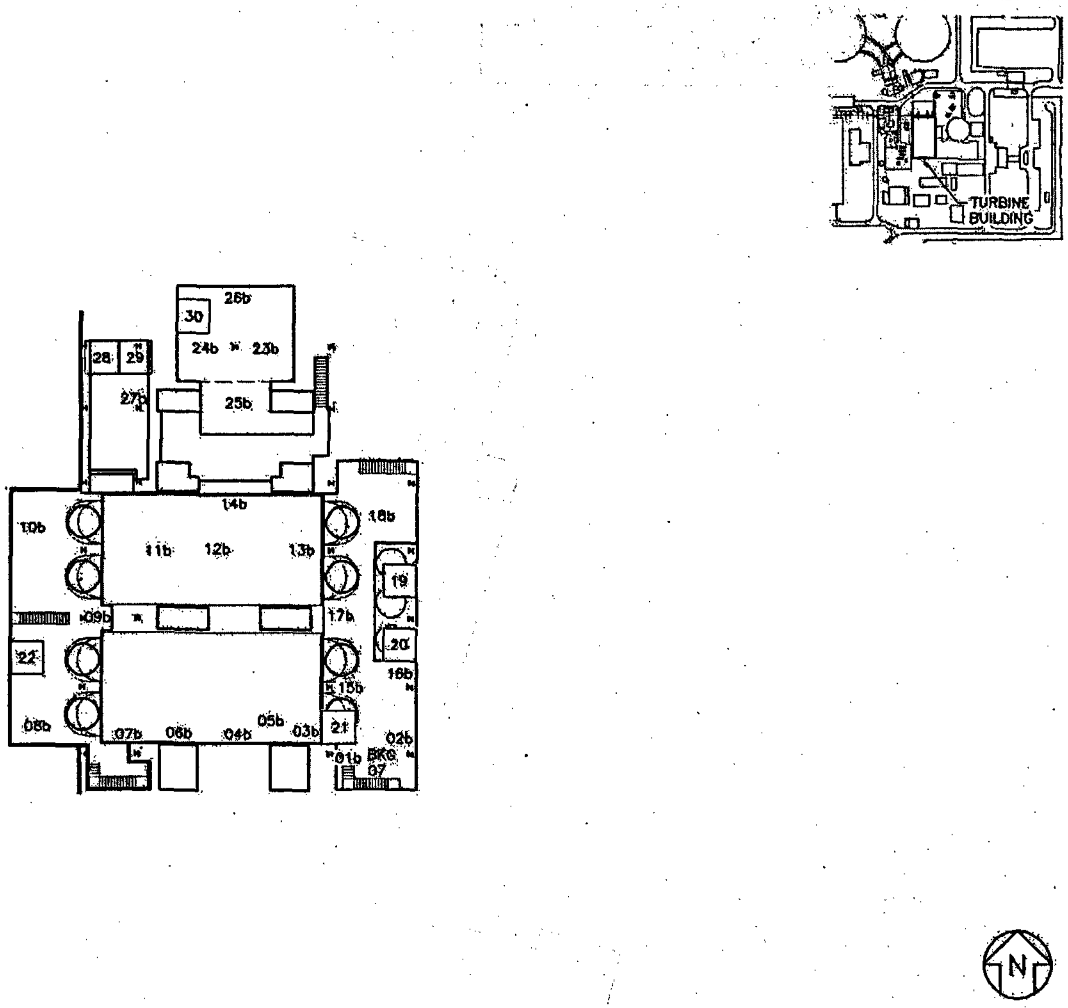
DWG: 823000.01



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2–128
Plant Cooling Water Intake
824000

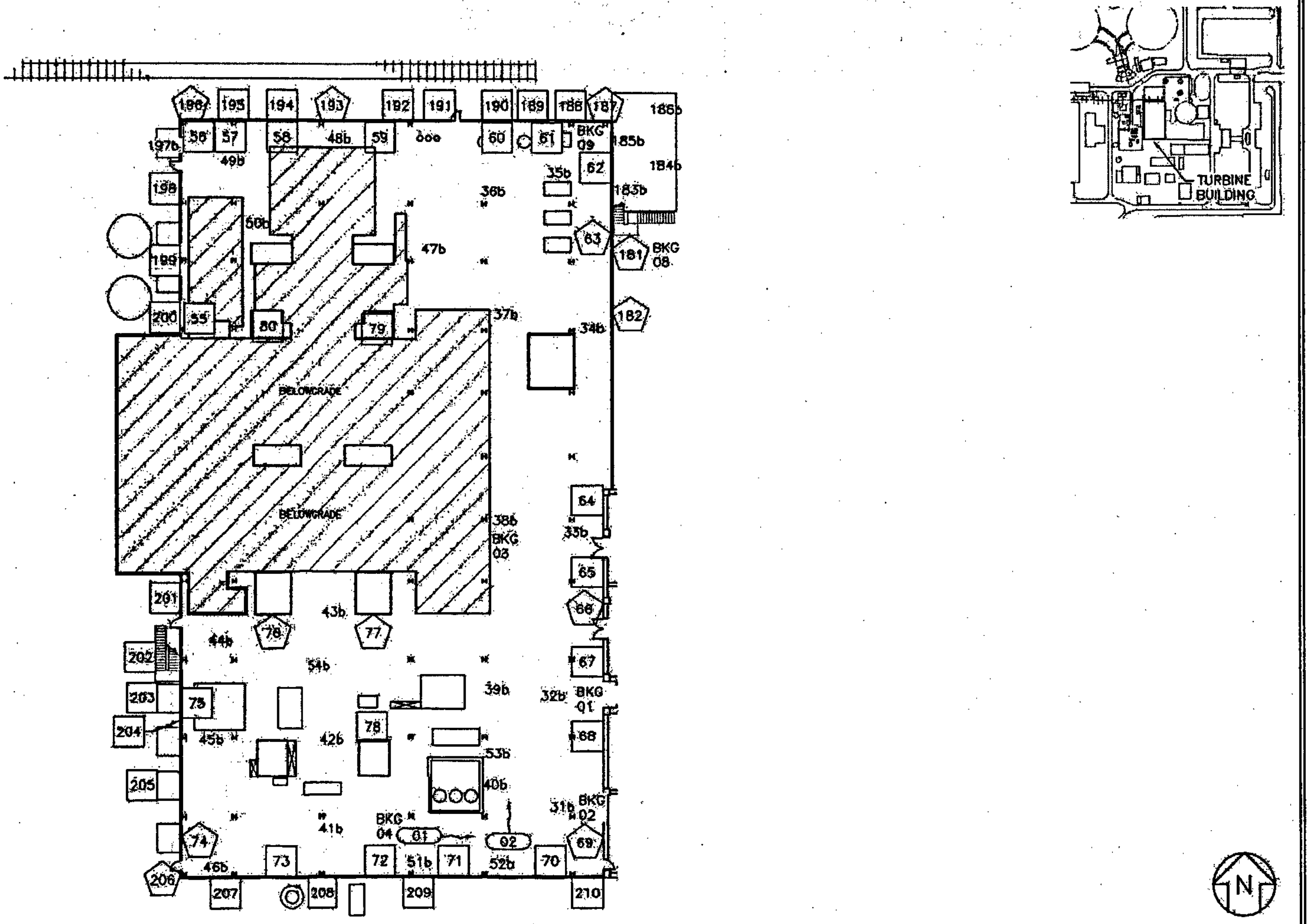
DWG: 824000.01



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-129
Turbine Building El. -10
826000

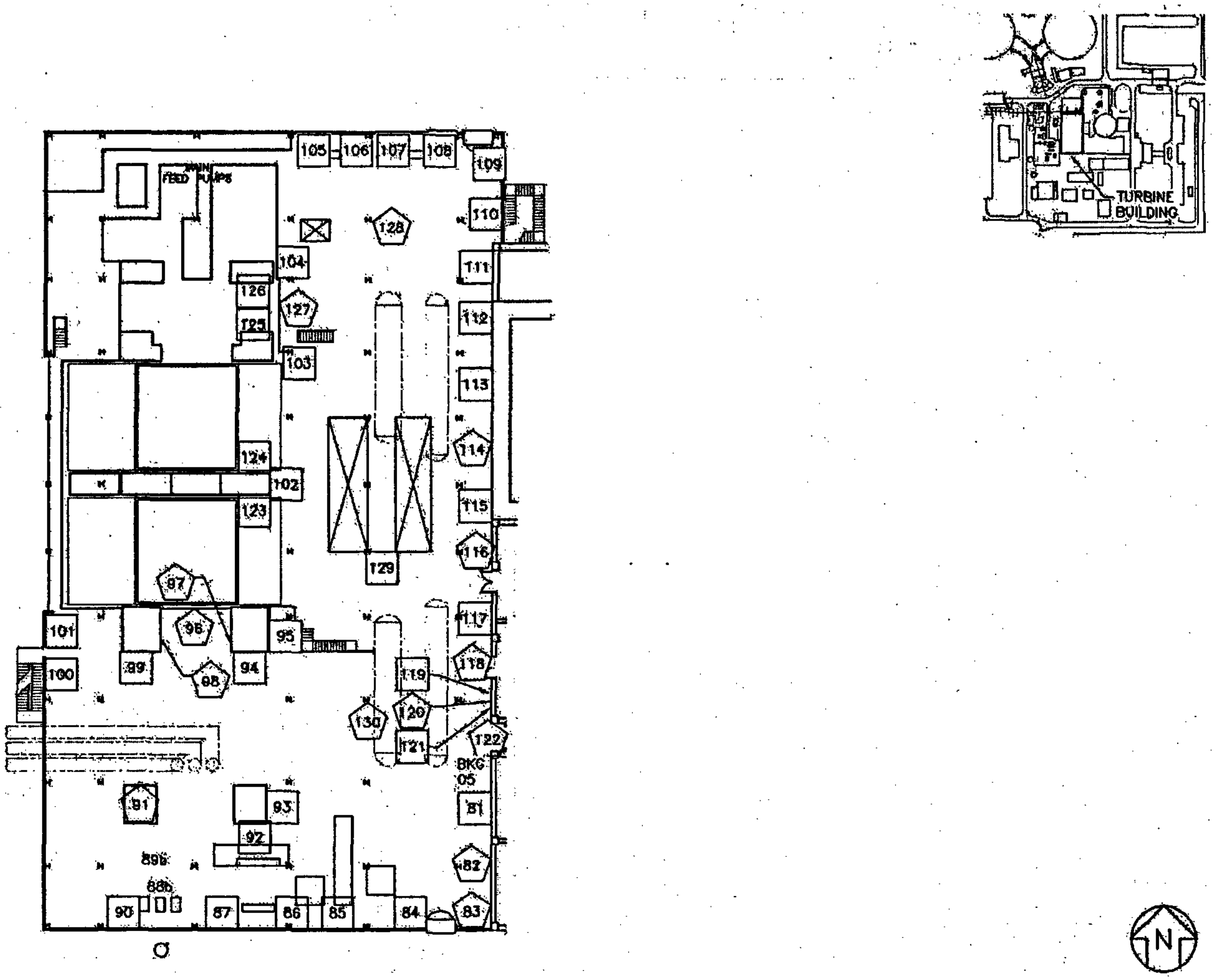
DWG: 826000.01



SMUD
SACRAMENTO MUNICIPAL
UTILITY DISTRICT

Figure 2-130
Turbine Building El. 0
826000

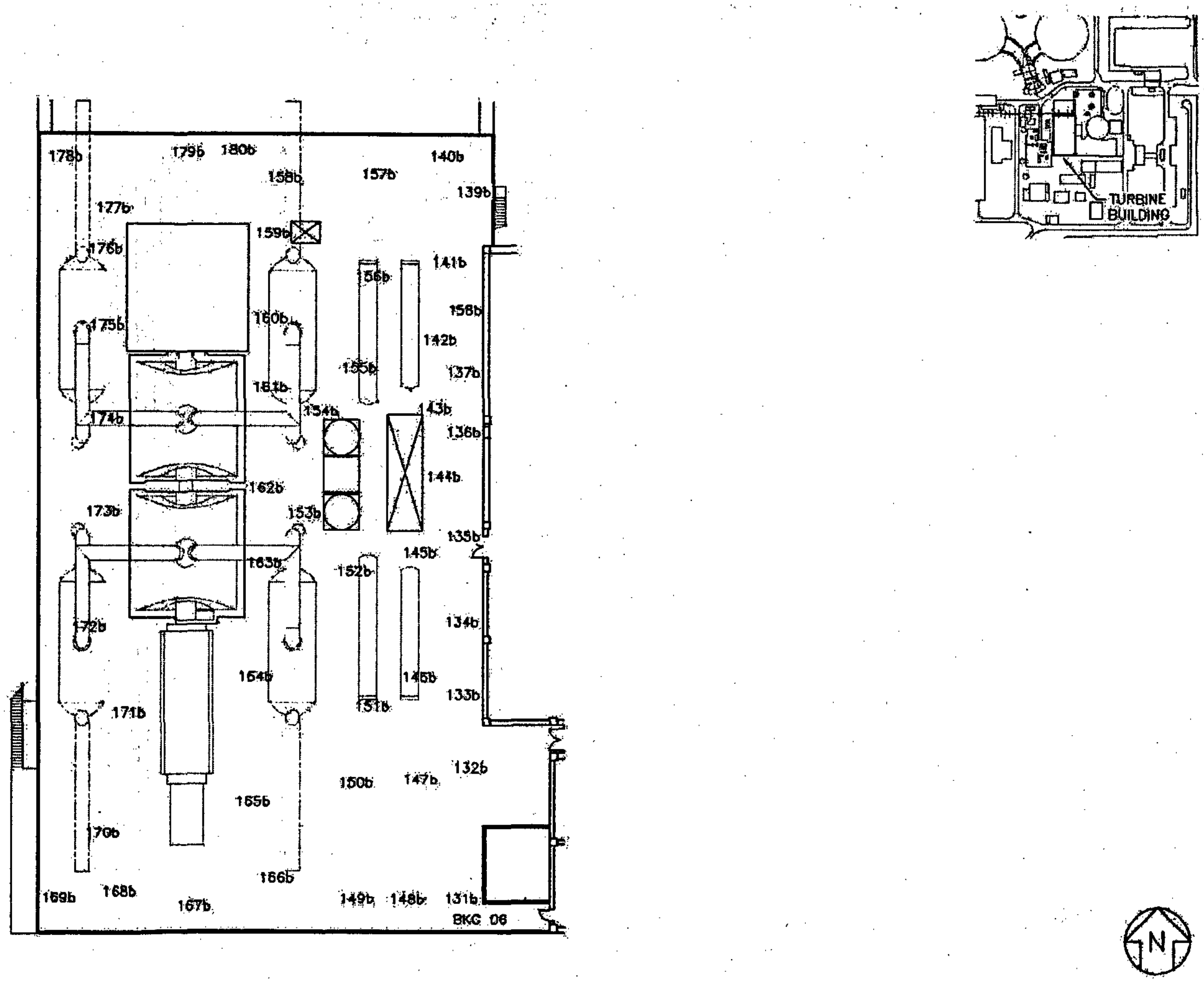
DWG: 826000.02



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2–131
Turbine Building El. 20
826000

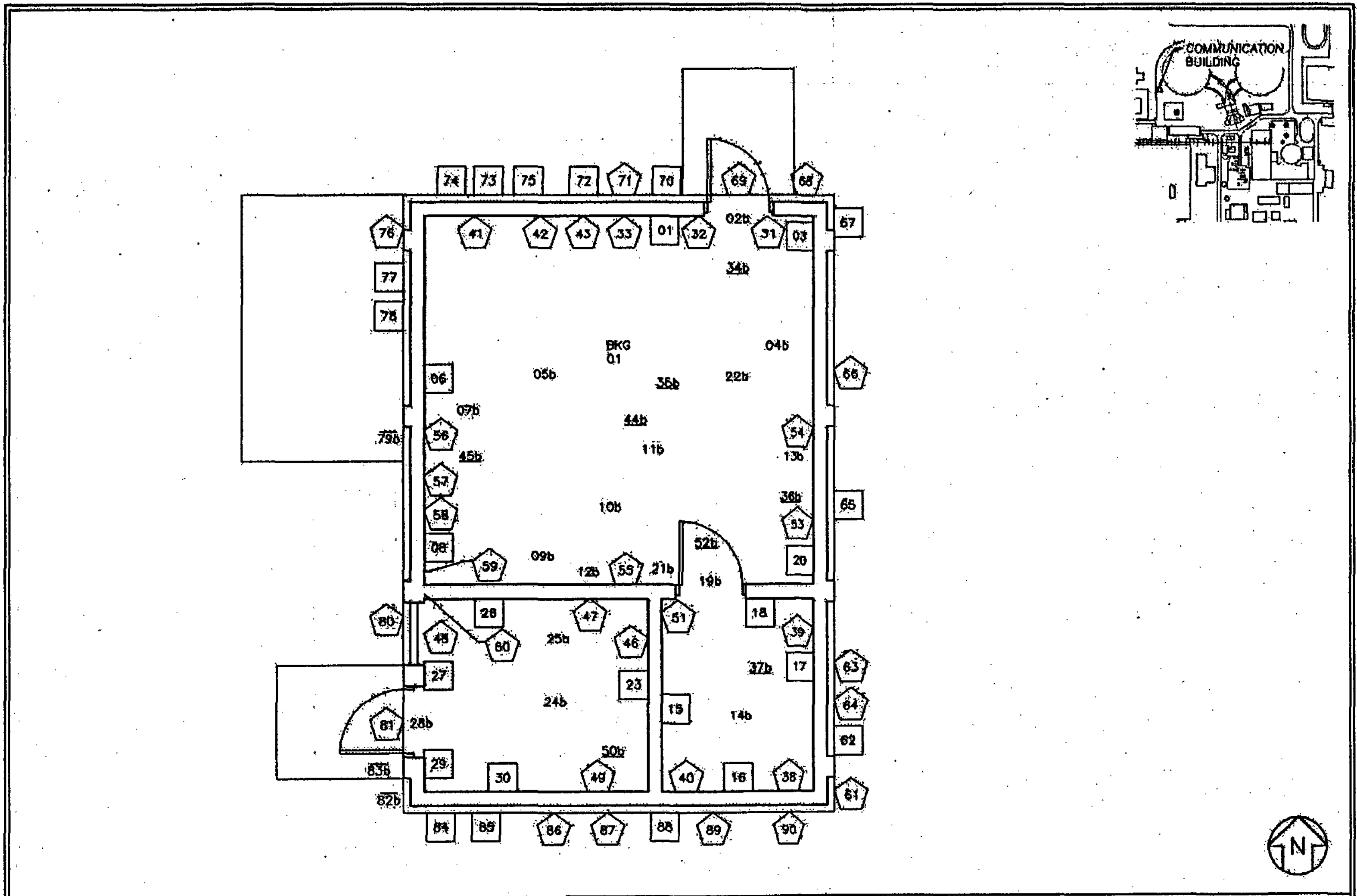
DWG: 826000.03



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2–132
Turbine Building El. 40
826000

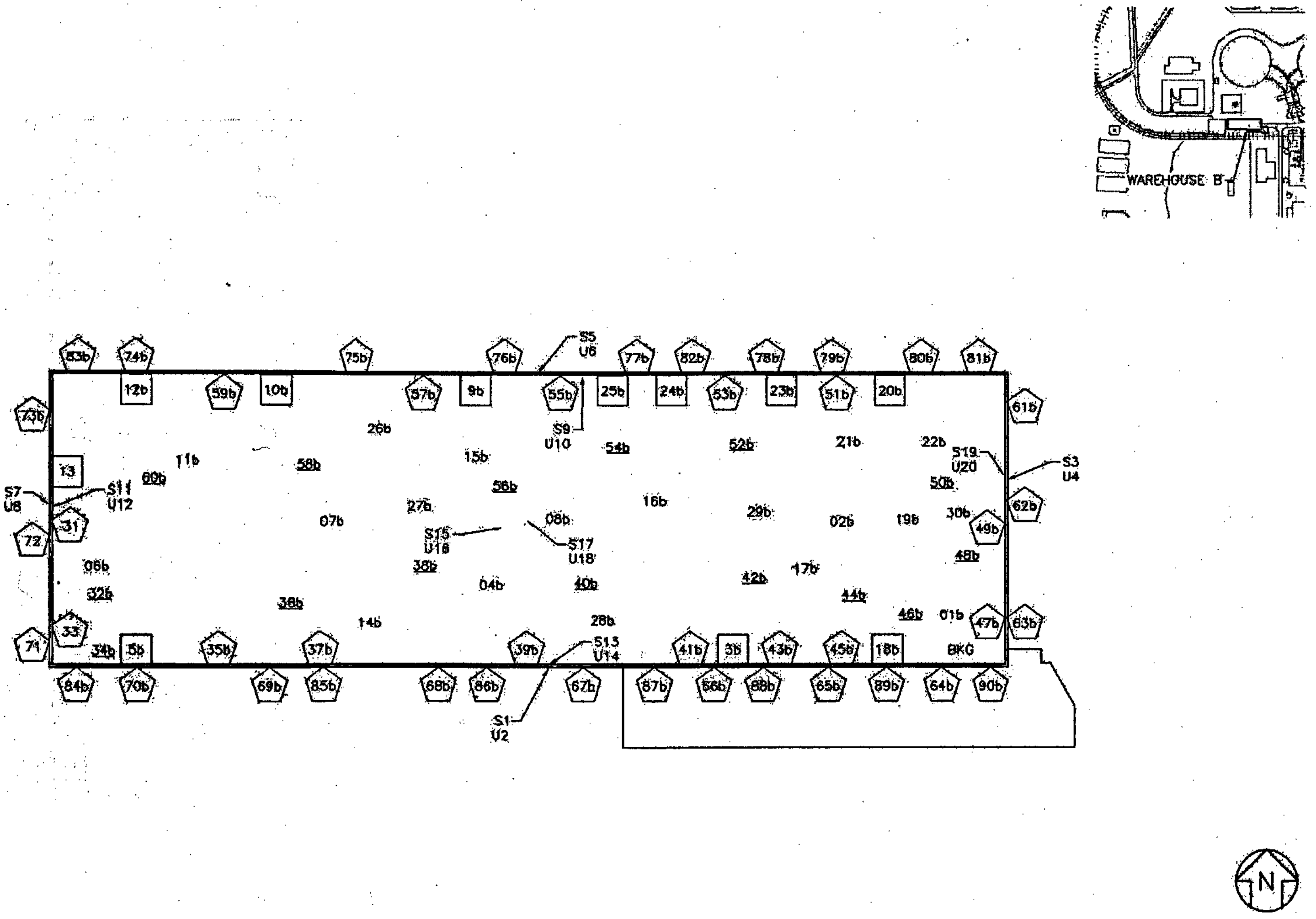
DWG: 826000.04



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-133
Microwave Communication Bldg
831000

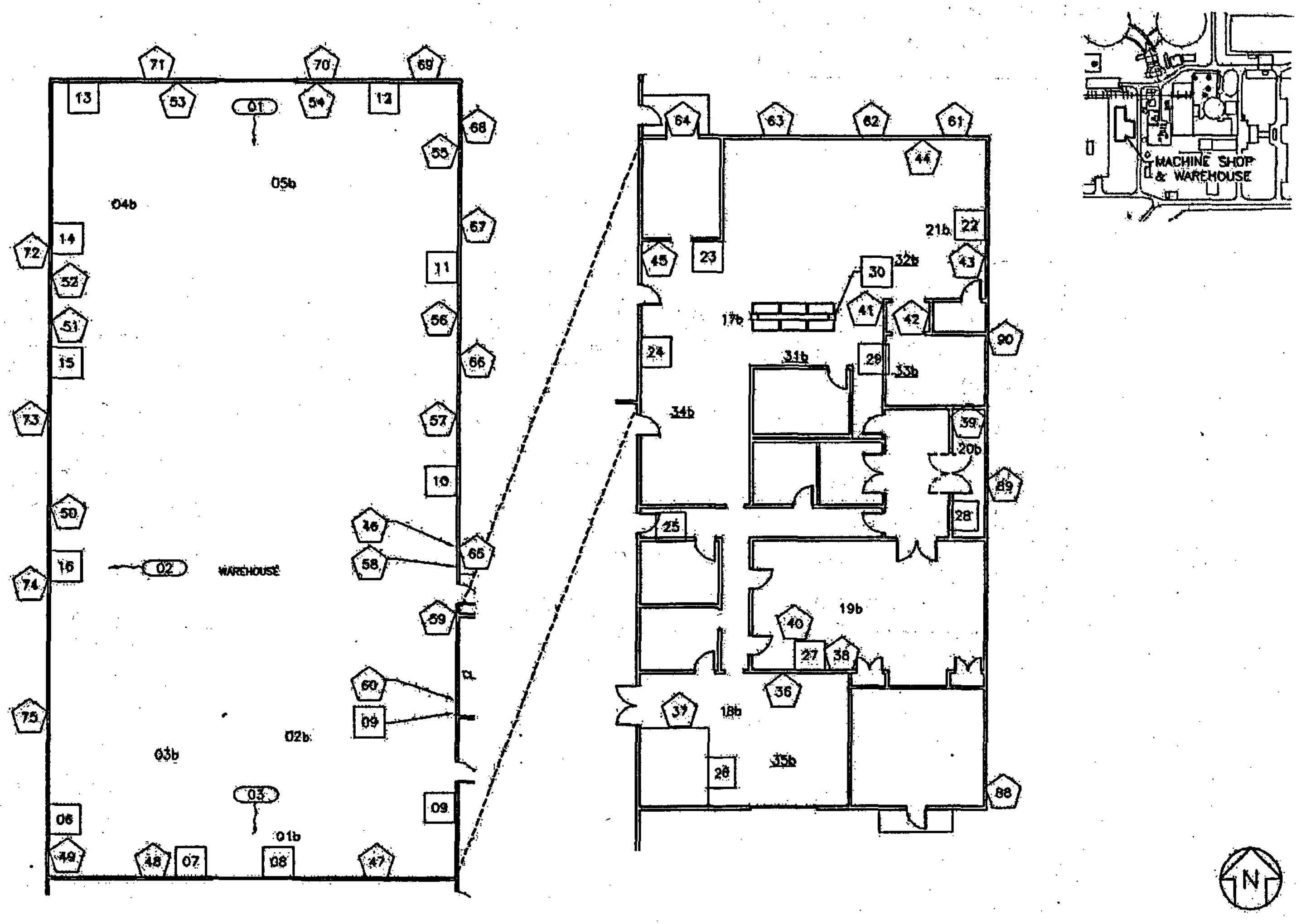
DWG: 831000.01



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-134
Warehouse B
833000

DWG: 833000.01



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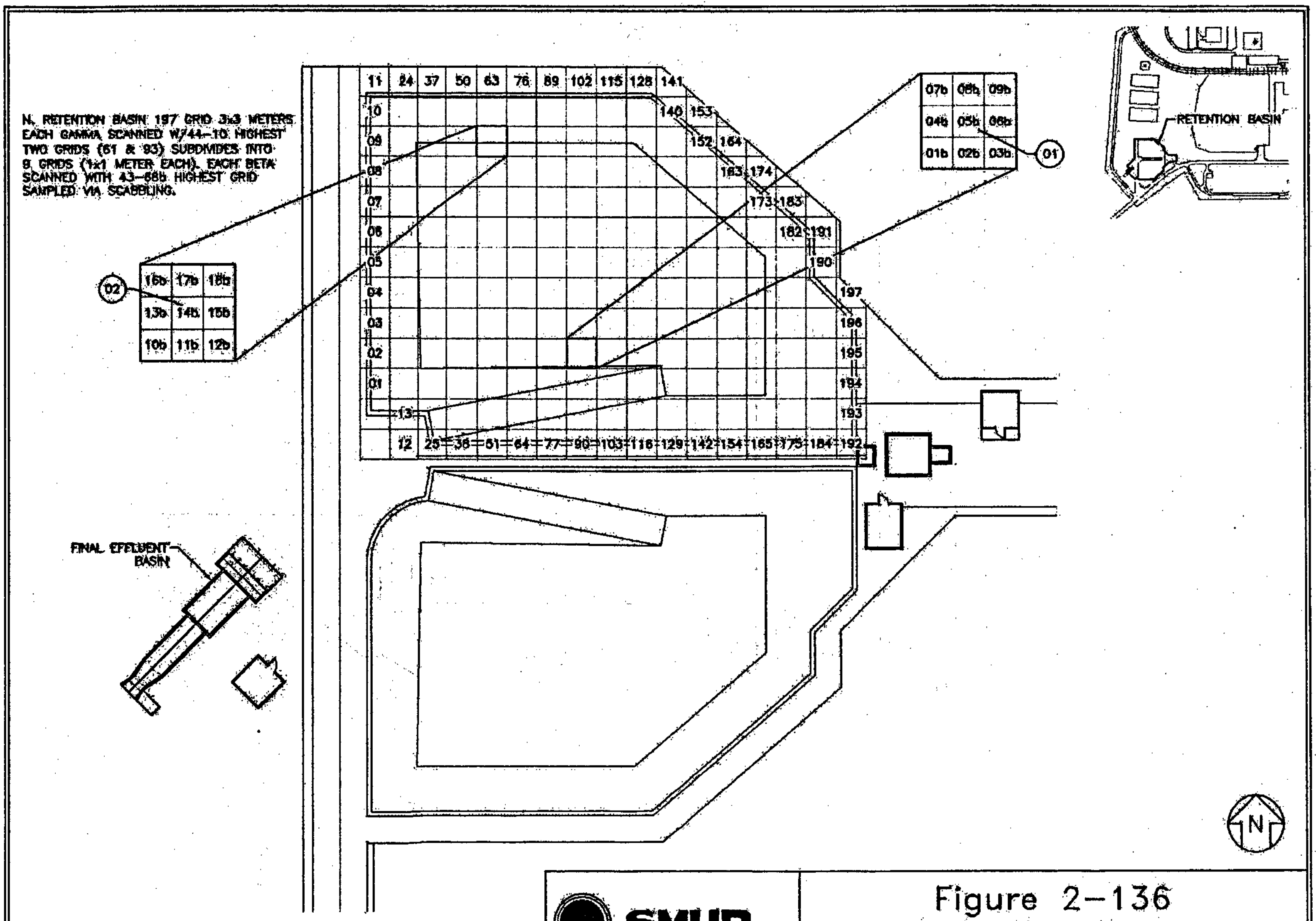
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2–135
Warehouse
840000

DWG: 840000.01



N. RETENTION BASIN 197 GRID 3x3 METERS EACH GAMMA SCANNED W/44-10. HIGHEST TWO GRIDS (61 & 93) SUBDIVIDES INTO 9 GRIDS (1x1 METER EACH). EACH BETA SCANNED WITH 43-68b. HIGHEST GRID SAMPLED VIA SCABBLING.

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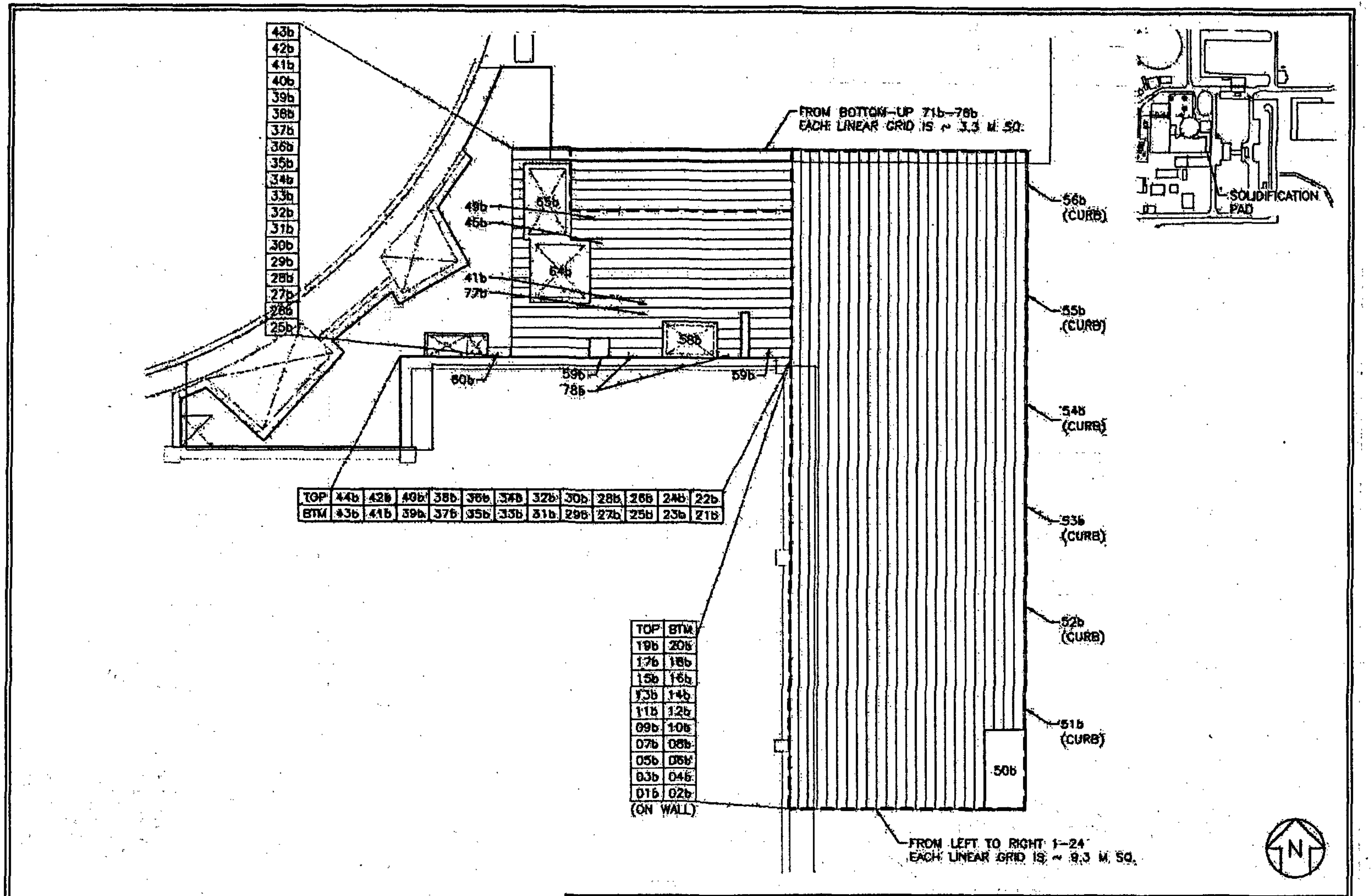
Figure 2-136
Retention Basin
848000

DWG: 848000.01



TOP	44b	42b	40b	38b	36b	34b	32b	30b	28b	26b	24b	22b
BTM	43b	41b	39b	37b	35b	33b	31b	29b	27b	25b	23b	21b

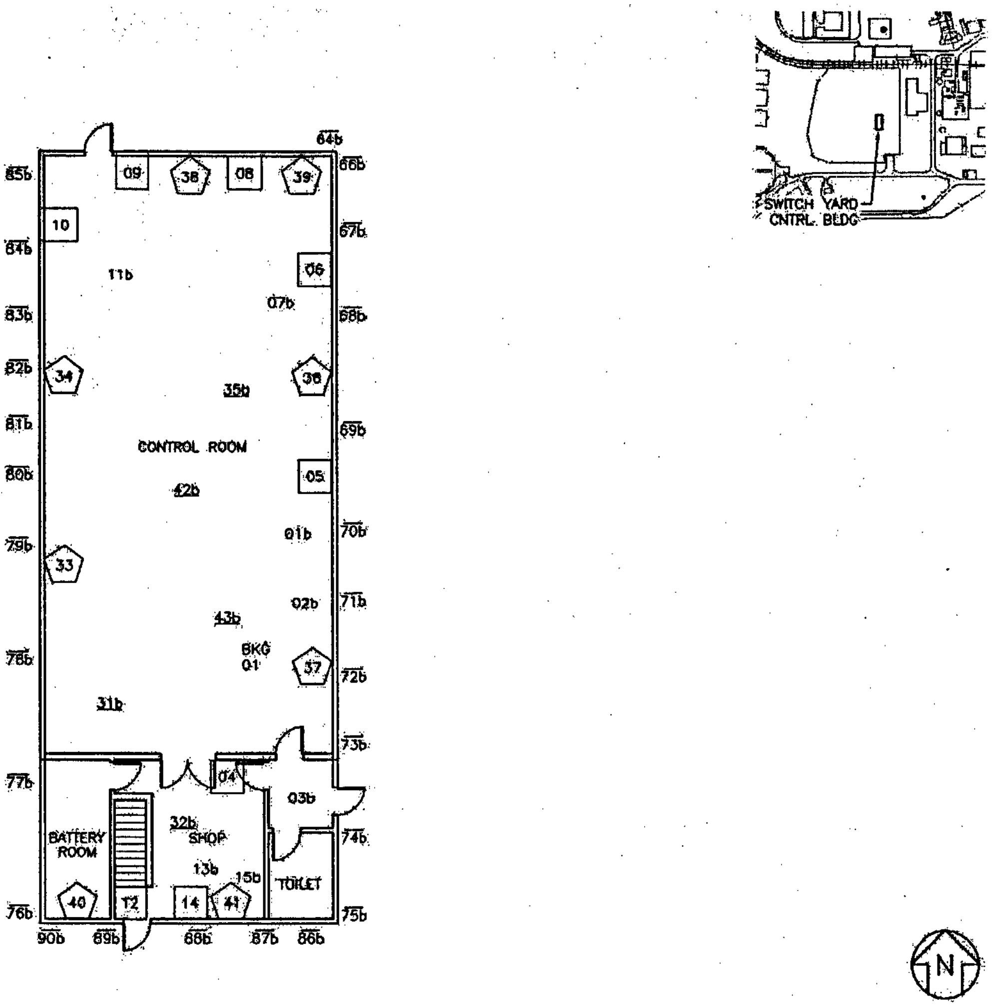
TOP	BTM
19b	20b
17b	18b
15b	16b
13b	14b
11b	12b
09b	10b
07b	08b
05b	06b
03b	04b
01b	02b

(ON WALL)

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-137
Solidification Pad
850001

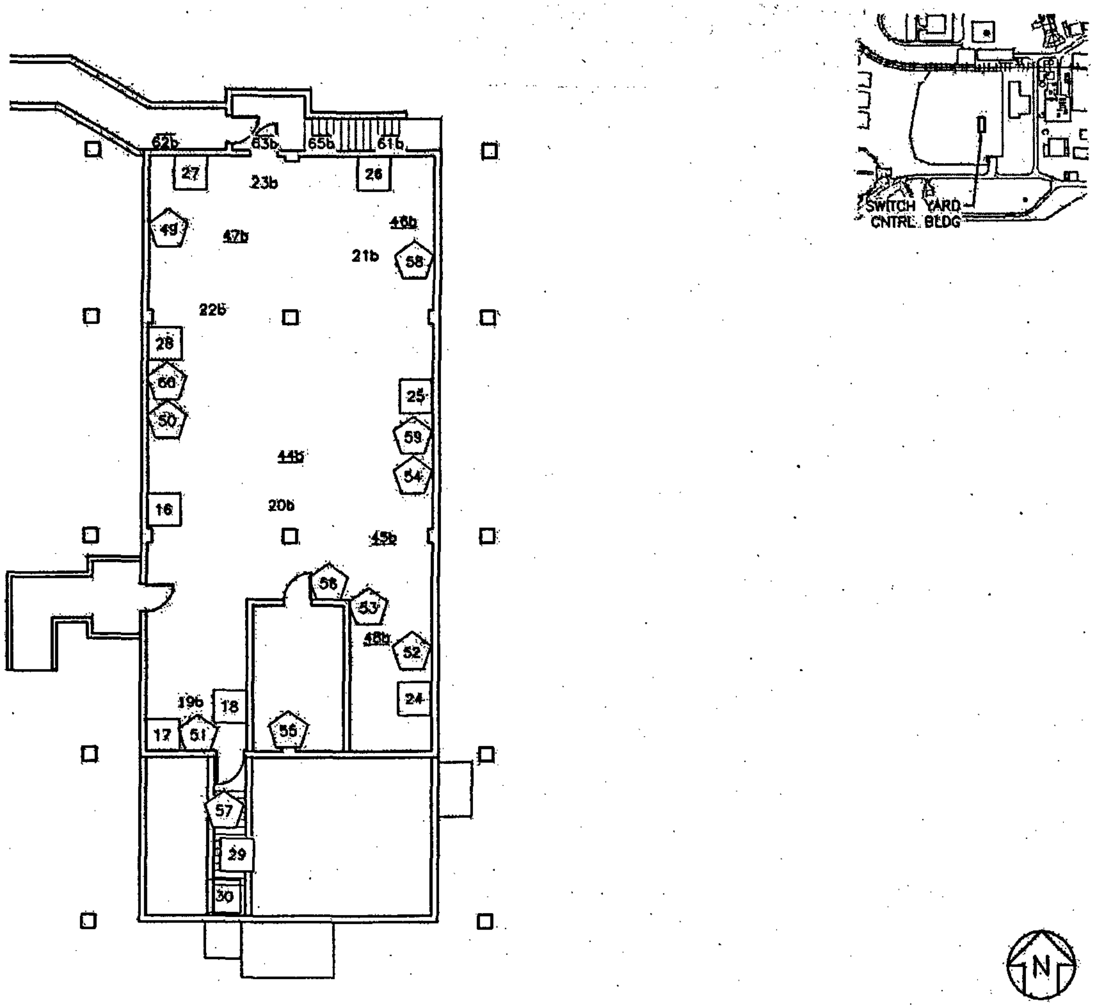
DWG: 850001.01



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-138
Switchyard Control Bldg El. 0
851003

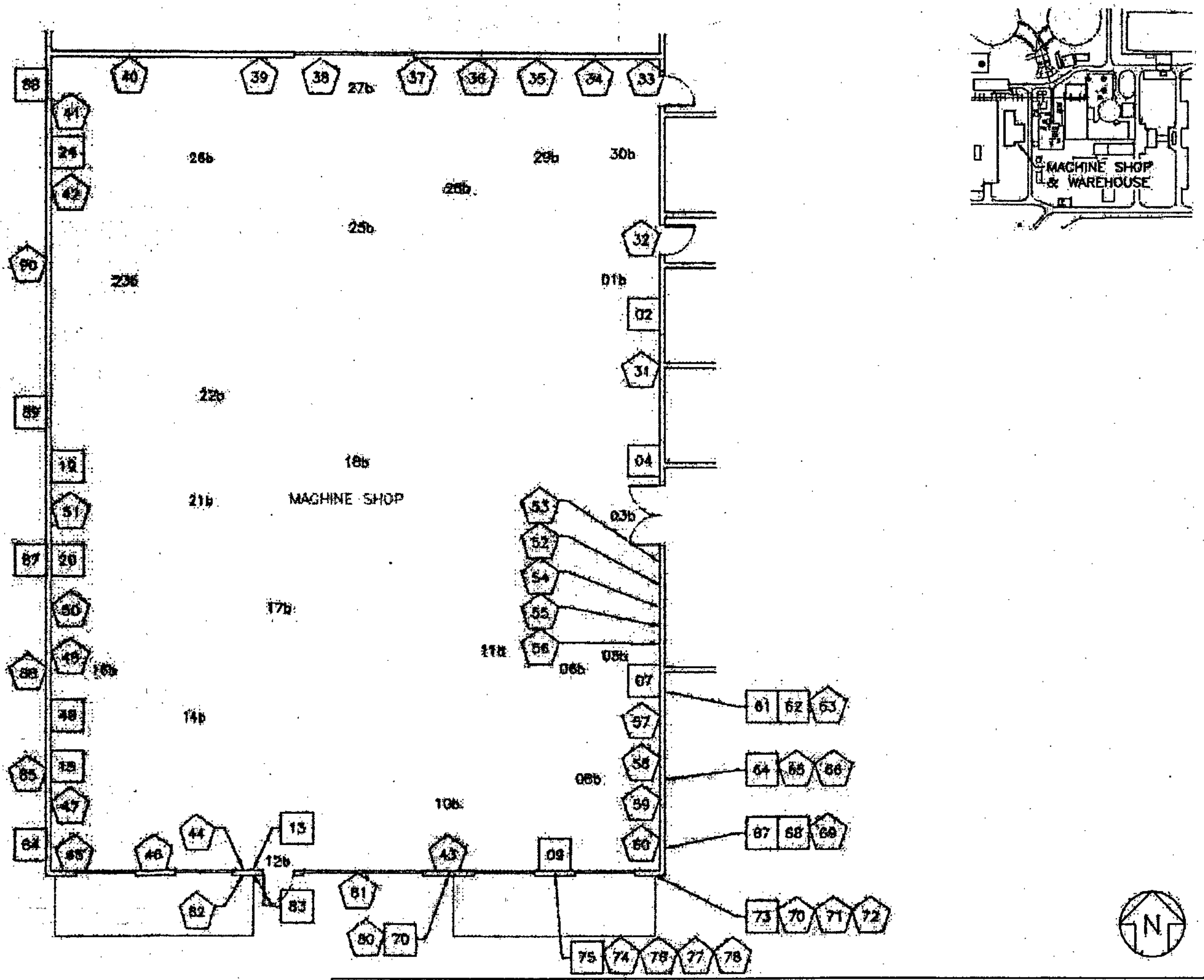
DWG: 851003.02



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-139
Switchyard Control Building Basement
851003

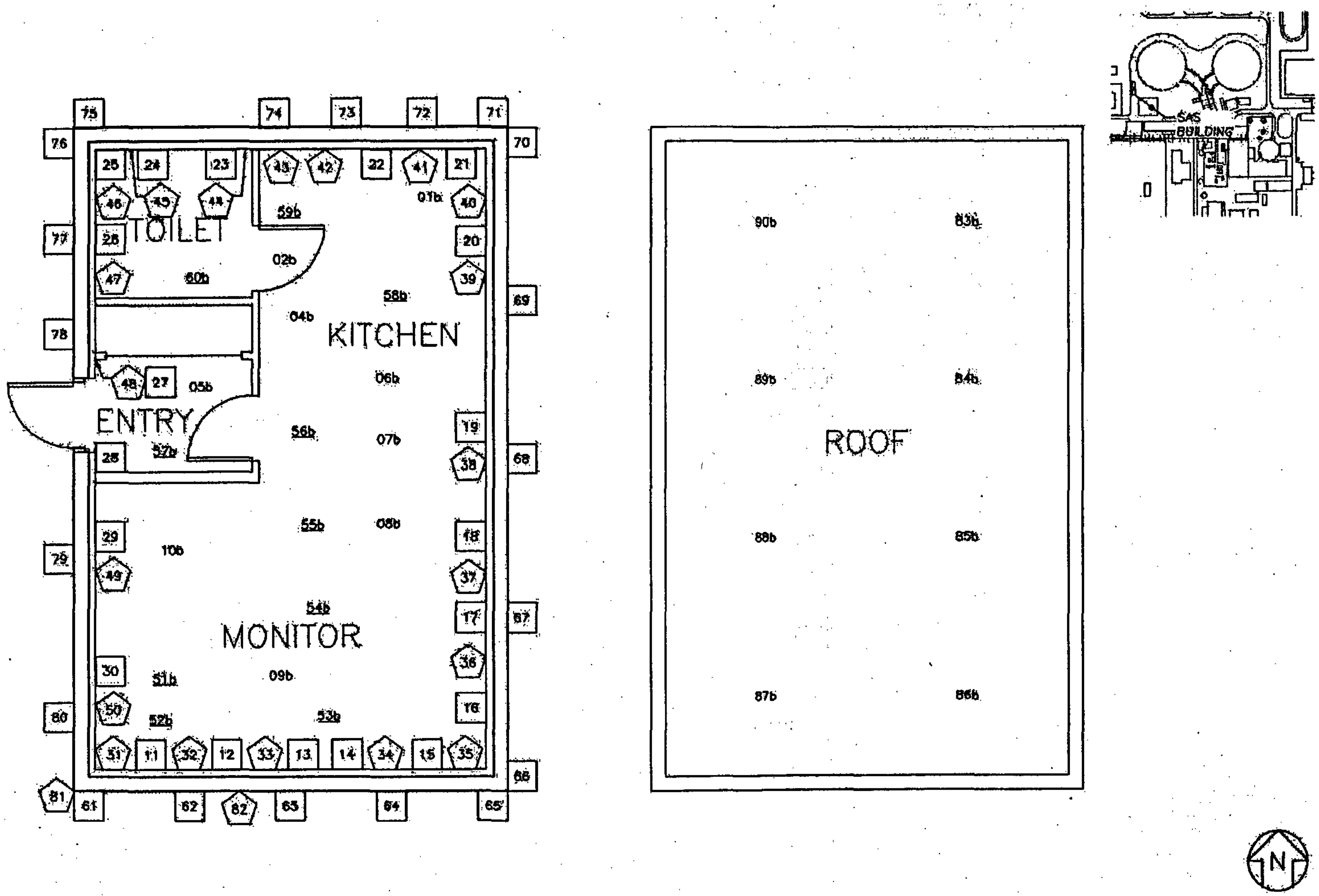
DWG: 851003.01



SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Figure 2-140
Machine Shop
852000

DWG: 852000.01



SMUD
SACRAMENTO MUNICIPAL
UTILITY DISTRICT

Figure 2-141
SAS Building & Roof
856000

DWG: 856000-01

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3.0 IDENTIFICATION OF REMAINING DECOMMISSIONING ACTIVITIES

3.1 Introduction

In accordance with 10 CFR 50.82 (a)(9)(ii)(B), the License Termination Plan (LTP) must identify the major remaining dismantlement and decontamination activities. This chapter was written following the guidance of NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," [Reference 3-1] and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," [Reference 3-2] and will discuss those dismantlement activities as of December 31, 2005. Information is presented to demonstrate that these activities will be performed in accordance with 10 CFR 50 and will not be inimical to the common defense and security or to the health and safety of the public pursuant to 10 CFR 50.82(a)(10). Information that demonstrates that these activities will not have a significant effect on the quality of the environment is provided in LTP Chapter 8, Supplement to the Environmental Report.

The information includes those areas and equipment that need further remediation and an estimate of radiological conditions that may be encountered. Included are estimates of associated occupational radiation dose and projected volumes of radioactive waste.

Sacramento Municipal Utility District's (District's) primary goals are to decommission the Rancho Seco Nuclear Generating Station (Rancho Seco) safely and to maintain the continued safe storage of spent fuel in an Independent Spent Fuel Storage Installation (ISFSI). The District will decontaminate and dismantle Rancho Seco in accordance with the DECON alternative, as described in NUREG-0586, "Final Generic Environmental Impact Statement" (FGEIS) [Reference 3-3]. Completion of the DECON option is contingent upon access to one or more low-level waste (LLW) disposal sites. Currently, Rancho Seco has access to the disposal facilities of Energy*Solutions*[1]. These facilities currently only accept Class A waste. Completion of the second phase of site release will require access to a LLW disposal facility, acceptable to the District, which can accept Class B and C waste.

The District is currently conducting decontamination and dismantlement (D&D) activities at the Rancho Seco site in accordance with the Rancho Seco Post Shutdown Decommissioning Activities Report (PSDAR) [Reference 3-4]. Decommissioning activities are being coordinated with the appropriate Federal and State regulatory agencies in accordance with plant administrative procedures. All special nuclear material (spent fuel) is located at the ISFSI. By the end of the second quarter of 2006, it is expected that all greater than Class C (GTCC) waste material will also be located at the ISFSI.

Decommissioning activities at Rancho Seco are conducted in accordance with the Rancho Seco Defueled Safety Analysis Report (DSAR) [Reference 3-5], Permanently Defueled Technical Specifications [Reference 3-6], Rancho Seco Quality Assurance Program (QAP) [Reference 3-7], existing 10 CFR Part 50 license, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). If an activity requires prior Nuclear Regulatory Commission (NRC) approval under 10 CFR 50.59(c)(2) or a change to the Rancho Seco Permanently Defueled Technical Specifications or license, a submittal shall be made to the NRC for review and approval before implementation of the activity in question.

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[1] Energy*Solutions* was previously Envirocare of Utah

Decommissioning activities are conducted in accordance with the Rancho Seco Radiation Protection Program, the Off-Site Dose Calculation Manual (ODCM), Safety Program, and the Radwaste Manual. Such activities are and shall be conducted in accordance with these established programs that are frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during former plant operations. Decommissioning activity radiological risk is bounded by previously analyzed radiological risk for former operating activities that occurred during major maintenance and outage evolutions.

The activities described in Section 3.3, Future Decommissioning Activities, include activities up to the future release of the site. This section provides an overview and describes the major remaining components of contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations.

Information related to the remaining D&D tasks is also provided. This information includes an estimate of the quantity of radioactive material to be released in accordance with 10 CFR 20.2001, a description of proposed control mechanisms to ensure areas are not re-contaminated, estimates of occupational exposures, and characterization of radiological conditions to be encountered and the types and quantities of radioactive waste. This information supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify inspection or technical resources needed during the remaining dismantlement activities. Many of these dismantlement tasks require coordination with other federal, state or local regulatory agencies or groups.

The dismantlement activities described in Section 3.3 provide the NRC the information to support site release and future license termination pursuant to 10 CFR 50.82(a)(11)(i). Therefore, this section was written to clearly indicate each dismantlement activity that remains to be completed prior to qualifying for license termination. The final state of the Industrial Area will be a partially abandoned facility (as defined in Chapter 1 of this LTP) with portions, other than the power block, available for reuse. The impacts of decommissioning activities performed will be to reduce residual radioactivity to a level that permits release of the property for beneficial reuse by the District for industrial purposes.

3.2 Completed Decommissioning Activities and Tasks

3.2.1 Spent Fuel Storage

The District signed the contract in 1992 for the design, licensing and fabrication of a transportable storage system. In 1995 the ISFSI was constructed and fabrication of the cask and associated equipment began. However, in 1996, quality issues throughout the dry storage industry and vendor bankruptcy forced work to be stopped. In 1997, a new supplier resumed the design and license work.

The transportable storage system consists of a transportation cask, twenty-one dry storage canisters, twenty-two horizontal storage modules and a multi-axle trailer. The cask serves for on-site transfer and off-site transportation overpack for the canisters. The canisters hold the spent fuel in a structural array and are then seal-welded. The horizontal storage modules are thick reinforced concrete storage bunkers used to store the canisters. The twenty-second module will provide storage for GTCC waste from reactor vessel internals.

Fuel movement began in May of 2001 and was completed in August of 2002. All spent fuel is currently stored in the ISFSI under a separate Part 72 license.

3.2.2 Spent Fuel Pool Activities

The eleven spent fuel racks were removed from the pool during the first quarter of 2003 and shipped to Energy*Solutions* for direct disposal. The process for removal and disposal began with vacuuming the debris from each cell, followed by radiological survey for hot spots and further rack decontamination during removal from the pool. A vacuuming unit coupled with high-loading filters was used to collect the loose debris from the racks.

Upon removal each rack was placed on the cask wash-down platform, where a thorough decontamination and survey of each cell and outer surface was performed. The racks were then removed from the wash-down platform and staged for drying, followed with a coating of spray adhesive, and wrapped in 12-mil plastic. The 12-mil plastic served for contamination control during packaging, which occurred outside on the plant turbine deck.

The racks were then moved to a laydown area outside the fuel building, down-ended, and placed in a watertight shipping bag. The final step involved re-rigging the rack for placement in a large metal strong-tight container. Each rack had less than an A2 quantity of radioactivity that allowed use of the strong-tight container. Radiological surveys were performed to ensure DOT radiation limits were met, communications applied, and the package placed on the transport vehicle for disposal. Each transport package contained only one rack and was transported by highway for disposal at Energy*Solutions*.

Dose rates on the rack exteriors ranged from 2 mrem/h on the top, to 15 mrem/h in the middle, and 50 to 80 mrem/h on the bottom. Hotspots within the cells ranged from 1 to 4 rem/h. The hotspots were easily removed through decontamination using high-pressure washing except for the 4 rem/h hotspot, which was found between the cells and was found to be mobile. After making several attempts to remove the hot spot, ready-mix grout was poured into the cell matrix, to fix the hotspot in place. The introduction of grout lowered the measured dose rate to less than 80 mrem/h.

Prior to decontamination, loose surface contamination ranged from 300,000 to 500,000 dpm/100 cm^2 beta-gamma, and less than 20 dpm/100 cm^2 alpha. Post-decontamination levels were found to be no more than 30,000 dpm/100 cm^2 beta-gamma.

Following successful completion of the rack project, the remaining pool water was drained, and the pool walls and floor were pressure washed. Loose contamination levels on the floor and walls after washdown ranged from 1,000 to 3,000 dpm/100 cm^2. The remaining water and wash water was sent to a holding tank for processing. Work then commenced on removal of the pool liner plate, which was constructed of ¼ inch thick stainless steel and was connected to the concrete wall with numerous embedded supports positioned at 6-foot centers in the horizontal and vertical direction. Various methods for removal were considered including plasma cutting and machine cutting the plates. Plasma cutting would have involved a tremendous effort to construct an enclosed area to control smoke and potential hazardous fumes (chromium +6) generated by the flame cut. Although a slow process, machine cutting was selected because it generated little secondary waste, involved no industrial hygiene concerns and was proven to be capable of performing the cuts.

The machine cutting was performed with a carbide bit installed on a hydraulically operated milling unit track which could be positioned horizontally or vertically depending on the cut to be made. The carbide bit travels along the milling unit track machining the stainless steel as it travels. The milling unit was affixed to the wall by use of fasteners installed with a stud gun.

The milled sections of the liner plate were loaded into top-loading 20 foot Seavans and shipped for direct disposal. Dose rates on a loaded Seavan were no greater than 0.2 mrem/h.

Once the liner plate was removed, the underlying concrete structure and subsurface soil was sampled to ascertain if pool water leakage (known to have occurred) would require their excavation and packaging for disposition. No significant activity was found below the concrete floor.

3.2.3 System Removal

System removal began in 1997 with secondary system components. As experience was gained in removal, decontamination and shipping, removal rates increased and staff was added to begin full decommissioning. In 2000 system removal moved to the Tank Farm and the Auxiliary Building, both containing significantly more contaminated systems. After an initial building decontamination, work moved to the Reactor Building in 2002. Most contaminated systems had been removed by mid 2004 with the exception of large components. Remaining radioactive liquids were processed in temporary systems.

3.2.4 Large Component Removal

3.2.4.1 Reactor Coolant Pumps

Main reactor coolant piping and the reactor coolant pumps were removed and shipped in 2002 with the exception of piping sections attached to the reactor vessel remaining in the primary shield walls. Primary piping was cut with machine tooling into short sections that could be filled with other piping and placed into standard shipping containers. The four reactor coolant pumps were removed and packaged for shipment to Energy*Solutions* in two rail cars. Packaging included welded covers on piping connections and stuffing boxes, paint for contamination control, and heavy bags for final packaging. The pumps were blocked and braced in the rail cars with heavy cables and steel cradles.

3.2.4.2 Reactor Head

A major work activity during 2003 involved the disposition of the Reactor Head. This Babcock and Wilcox design consisted of sixty-nine Control Rod Drive Motors, each weighing approximately 1,000 pounds; a Service Structure weighing 35,000 pounds and the Reactor Head itself, weighing 160,000 pounds.

This work began with removal of the Service Structure, which was removed from the Reactor Head after flame cutting the lower shroud. The Service Structure was removed from the Refueling Cavity and taken to an adjacent work area where it was segmented. These sections were packaged into a 20-foot Seavan, which was subsequently sent to a processor for dispositioning.

The next step was to remove the Control Rod Drive Mechanisms (CRDMs) from the Reactor Head. The CRDMs were grouped by their applicable function during plant operation and consisted of safeties, control and power shaping rods. There was very little radiological data associated with the CRDMs and leadscrews, which connected to the control rods, thus the dismantlement crew proceeded very carefully during removal of the CRDMs.

The CRDMs were removed by cutting the nozzles just below the mounting flange by use of a machine tool. Once cut, the CRDM was lifted from the cavity, surveyed and placed in a processing area where it was segmented into box-sized lengths for disposition.

The first CRDM removed was from the "safety" group and surveys indicated low dose rates. The survey along the length of the lead screw indicated 50 to 60 mrem/h gamma while the tip of the lead screw was 40 mrem/hr gamma. There was little fluctuation in dose rates as the different CRDM groups were removed and surveyed. All were packaged within a metal container and sent for direct disposal after segmentation.

The Reactor Head was segmented with use of a diamond wire cable supplied by the segmentation vendor. The five segmented sections included three sections of the flange and two sections of the top portion of the Head, cut just off-center through a clear path around the remaining portions of the CRDM nozzles.

The reactor head showed no signs of corrosion including the extended area around the CRDM nozzles.

3.2.4.3 Pressurizer

The Pressurizer, a 45-foot long, 150-ton component, was disposed of at Energy*Solutions* in May of 2004. Preparation for this project included removal of piping systems with subsequent plugging of the penetrations. Contracts were established for rigging and removal of the Pressurizer from the Reactor Building and for railroad transport to Energy*Solutions*. Exterior dose rates were 0.2 mrem/h or less except for a hot spot at the Pressurizer bottom where the surge line exits the vessel. To ensure 49 CFR 173.441 radiation limits were met, a carbon steel shielding cover was placed over the surge line and welded to the exterior of the vessel reducing the contact dose rate to less than 200 mrem/h. To prepare the vessel for contamination control while handling onsite, a polymer-based latex paint was applied to the exterior rendering loose contaminant levels to less than 1,000 dpm/100 cm^2. The Pressurizer was shipped as a surface contaminated object within a soft-sided strong tight container

3.2.4.4 Steam Generators

The Rancho Seco Steam Generators are of Babcock & Wilcox (B&W) design and commonly known as Once-Through Steam Generators (OTSGs). The B&W design consists of two such steam generators, each approximately 80 feet in height, 12 feet in diameter, and over 550 tons in weight. The OTSGs were too large to ship to Energy*Solutions* in their intact state due to rail clearances with respect to the length of the generator and certain radii of track along the required route to the disposal facility. Rancho Seco cut the OTSGs in the latitudinal direction at approximately the halfway point and capped the cuts with large steel plates to meet rail requirements and enable the OTSGs to be shipped directly for disposal to Energy*Solutions*.

Rancho Seco staff evaluated each section of the OTSG as it's own package and included other documentation to submit with the request for the DOT Exemption which was approved by the DOT in May 2004. The first OTSG was segmented and removed from the Reactor Building in the last quarter of 2004 and loaded onto railcars. Blocking and bracing work was completed and shipment of the OTSG sections was performed in December 2004. The second OTSG was similarly prepared and shipped in January 2005.

3.2.4.5 Outside Tanks

Two large stainless-steel tanks and two lined carbon-steel outdoor tanks were dismantled, packaged and shipped for direct disposal in 2003. The four tanks included the:

1. Borated Water Storage Tank (BWST),
2. Demineralized Reactor Coolant Storage Tank (DRCST),
3. The A Regenerant Hold-Up Tank, and
4. The B Regenerant Hold-Up Tank (RHUT).

The BWST and DRCST contained water for reactor coolant makeup and filling the reactor cavity during outages. The RHUTs held water collected for discharge.

The process for tank removal was the same for all four – layout of cut locations, lead paint abatement of these locations (if required), plasma arc segmentation of the stainless-steel sections or cutting torch segmentation of the carbon steel sections and packaging the sections in open-top 20 foot Seavans. The original plan for the RHUTs was to attempt free release, however residual activity prevented this. The BWST and DRCST presented minor contamination control challenges regarding radioactivity within the tanks - up to 400,000 dpm/100 cm^2 beta-gamma was discovered on the surfaces of the inner walls and floors. A wash-down of the interior was conducted prior to segmentation and the wash water with gross contamination was sent to a holding tank for processing.

The inside of the RHUTs were lined with a rubber barrier, which was removed and placed within a Seavan with segmented sections of the tank. The barrier was only slightly contaminated and the desire was that the inner tank would be free of detectable contamination and could thus, be free released. Small amounts of radioactive contamination were discovered in many areas inside the tank and the decision was made not to pursue free release.

The BWST and the DRCST each weighed 112,000 pounds while the A RHUT weighed 38,000 pounds and the B RHUT weighed 56,000 pounds. The segmented waste from these tanks was packaged into eight open-top 20-foot Seavans; each Seavan contained approximately 300 cubic feet of waste and was shipped for disposal at Energy*Solutions*.

3.2.5 Underground Pipe Removal

Underground pipe of highly contaminated systems has been removed except for a small portion remaining to be removed in 2006. These systems include the Decay Heat System, Borated Water System, Radioactive Waste System and the Spent Fuel Cooling System. Other minimally contaminated systems were also removed, or sampled and surveyed to ensure that they could remain. These include the Component Cooling Water System, Auxiliary Feedwater System, Main Condensate and Make-up System, and portions of the Clean Drain System used for radioactive effluents. Portions of non-impacted systems in close proximity to the target piping were also removed. The radioactive discharge line from the RHUTs to the Retention Basins is scheduled to be removed in the spring of 2007.

3.2.6 Non-Radiological Decommissioning Activities

Non-radiological decommissioning activities include the removal of temporary buildings such as wooden or metal structures after being cleared by Radiation Protection. Underground storage tanks for diesel fuel oil were removed and the remaining lines cleaned. Asbestos was removed from the cooling towers, the roofs of permanent buildings and other miscellaneous locations. Removal of non-essential materials and equipment and general cleanup of the site was also performed. The electrical generator was sold and removed. The Switchyard remains in operation by the newly constructed Cosumnes Power Plant (CPP) being operated on a non-impacted portion of the 2,480-acre Rancho Seco site.

3.3 Future Decommissioning Activities

Table 3-1 lists the current schedule for the remaining decommissioning activities. The following sections describe those activities.

Table 3-1

Schedule of Remaining Major Activities

Activity	Start Date	Finish Date
Reactor Vessel Internals Removal	January 2005	June 2006
Rector Vessel Removal	May 2006	December 2006
Underground Piping	June 2006	May 2007
Reactor Building Internal Structures Removal	December 2006	January 2008
Embedded Piping Decontamination	October 2005	June 2007
Auxiliary Building Decontamination	October 2005	October 2007
Spent Fuel Pool Decontamination	April 2006	April 2007
Wastewater Systems Decontamination	October 2005	October 2007
Reactor Building Decontamination	January 2008	June 2008

3.3.1 Remaining Component Removal

3.3.1.1 Reactor Vessel Internals

The Reactor Vessel Internals project is currently in progress and is expected to be completed in the second quarter of 2006. The final activation analysis and radiological characterization of the Vessel and Internals was completed in June of 2003. As of that date the GTCC waste totaled approximately 50,000 curies with 28,000 curies attributable to Co-60.

The GTCC waste weighs approximately 25,000 pounds and has been packaged into a single canister and will be stored within the onsite ISFSI alongside the spent fuel under the separate Part 72 license. It is scheduled to be transported from the reactor cavity to the ISFSI in the fuel cask by June of 2006.

Class A pieces are being shipped in boxes and liners to Energy*Solutions*. Class B and C waste, approximately 16,000 curies (as of January 1, 2006), will be stored in liners in the Interim Onsite Storage Building (IOSB) under the Part 50 license until acceptable disposal is arranged.

With the exception of the plenum, the vessel internals are being cut or disassembled underwater with mechanical milling or cutting devices designed to minimize the production of fine material that could be dispersed in the water or air.

3.3.1.2 Reactor Vessel

The Reactor Vessel is scheduled to be segmented and shipped beginning in the third and fourth quarters of 2006 once the vessel internals project is complete, the cavity drained, and the cavity and vessel are cleaned. In preparation, shield blocks around the flange will be removed and the mirror insulation on the vessel will be removed. Once the vessel is removed the remaining primary piping segments will be removed from the shield wall penetrations.

The vessel will be segmented using a robotic arm and a high-pressure water/abrasive cutting head. Pieces will be sized to allow standard packaging with the exception of the cylinder pieces opposite the core that will require special packaging. The fuel region of the vessel will be cut into six pieces vertically and be placed into two boxes, grouted and transported by rail.

3.3.1.3 Remaining Underground Pipe

Portions of underground pipe (approximately 140 linear feet) from the Borated Water System and the Decay Heat System in the area in front of the Reactor Building equipment hatch remain to be removed. They are scheduled to be removed during the summer of 2006. The radioactive discharge line from the RHUTs to the Retention Basins is scheduled to be removed in the spring of 2007.

3.3.1.4 Reactor Building Exhaust Ventilation

The only remaining radiological exhaust system, the Reactor Building Exhaust System, is currently in use to exhaust and filter the Reactor Building atmosphere. This system will remain until significant activities in the Reactor Building are complete.

3.3.2 Reactor Building Internal Structure Removal

Once the Reactor Vessel is removed, work is to begin on the removal of almost all concrete and internal structures in the Reactor Building. Removal of the concrete to the liner plate should minimize the need for decontamination and simplify the final status survey (FSS). Only the building liner should remain inside the structure and possibly a grade level platform just inside the equipment hatch. The structure removal includes the activated steel and concrete around the vessel, and the polar crane.

3.3.3 Decontamination of Structures and Systems

Decontamination methods include wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. Selection of the preferred method is based on the specific situation. Other decontamination technologies will be considered and utilized, as appropriate. Approved administrative procedures and processes control decontamination. These controls ensure that wastewater is collected for processing by liquid waste processing systems. Airborne contamination control and waste processing systems are used as necessary to control and monitor releases. Decontamination methods are further discussed in Chapter 4 of this LTP.

3.3.3.1 Reactor Building

Some liner decontamination is expected to be required once internal structures are removed. Cleaning, up to and including paint removal, will be done as necessary to meet the derived concentration guideline level (DCGL).

3.3.3.2 Auxiliary Building

Extensive decontamination is planned for rooms below grade level in the Auxiliary Building. Many of the rooms were exposed to leaking or spraying water systems and decontamination is expected to include extensive surface removal including core boring and sawing. It may be necessary to remove floors or sumps if contamination extends through the concrete. These rooms are currently undergoing removal of obstacles that will interfere with 100% scanning surveys.

3.3.3.3 Spent Fuel Pool

Significant decontamination is required for the Spent Fuel Pool. The pool liner has been removed and one interior wall where significant pool liner leakage has occurred is scheduled for removal. Once the wall is removed decontamination of remaining wall and floor surfaces will occur as well as the cleaning of the embedded leak chases and through-wall pipes.

3.3.3.4 Turbine Building

The Turbine Building has only minor contamination levels with little decontamination planned with the exception of selected floor drain piping segments and sumps.

3.3.3.5 Embedded Pipe Systems

Embedded pipe systems are located in all of the impacted buildings listed above. Most embedded system piping is for floor drains. Cleaning is in progress with an initial high-pressure wash to remove debris followed by an abrasive grit blast process as required. Once cleaned to acceptable limits most embedded piping will be grouted to mitigate reuse or transport of remaining residual activity.

3.3.3.6 Wastewater Systems

While most wastewater piping that will remain is believed to be below DCGL levels, the Retention Basins and associated bottom drains will require remediation and/or partial removal. The radioactive discharge line from the RHUTs to the Retention Basins is expected to be removed. Storm drains that lead directly offsite and storm drains that collect system drainage and lead to the outfall should require no remediation. Some system piping that leads to the storm drains is currently being removed in lieu of extensive surveys. Cleaning is currently underway on the oil/water separator. Oil and sludge will be removed and a FSS will be performed.

3.3.4 Non-Radiological Activities

3.3.4.1 Outbuilding Demolition

The demolition of temporary outbuildings continues. The remaining concrete pads will be surveyed as a part of the FSS process.

3.3.4.2 Site Grading

Once Phase I site release is obtained, low areas will be filled and graded for drainage. These areas include the cooling tower basins and canal, the spray ponds and the below grade portion of the Turbine Building including the circulating water lines. Other grading and landscaping may occur.

3.3.5 Control Mechanisms to Ensure No Recontamination

Due to the large scope of remaining structures and systems to be decontaminated and the need for some FSS activities to be performed in parallel with dismantlement activities, a systematic approach to controlling areas is established. Upon commencement of the FSS for survey areas within the Restricted Area where there is a potential for re-contamination, implementation of one or more of the following control measures will be required:

- Personnel training,
- Installation of barriers to control access to surveyed areas,
- Installation of barriers to prevent the migration of contamination from adjacent areas,
- Installation of postings requiring personnel to perform contamination monitoring prior to surveyed area access,
- Locking entrances to surveyed areas of the facility,
- Installation of tamper-evident labels or seals, or
- Upon completion of FSS, the area is placed under periodic routine survey by Radiation Protection to ensure no re-contamination occurs. If re-contamination is identified, an investigation will be initiated that would result in corrective actions up to and including re-performance of the FSS for that area.

3.3.6 Deferred Activities

3.3.6.1 Storage of Class B and C Waste

It is the decision of District management that acceptable waste disposal options for Class B and C waste do not exist at this time. As a result, the waste will be stored in the IOSB until such time as an acceptable waste disposal site is available. Once a solution is available, waste will be shipped and the building will be decontaminated as required.

3.3.6.2 Final Status Survey of IOSB

Once the IOSB is decontaminated a FSS will be performed in accordance with this LTP and a final release from the Part 50 license will be requested. The time frame for that request is currently scheduled for 2028, but depends of factors outside of District control. The request is expected to occur prior to the decommissioning of the ISFSI.

3.4 Radiological Impacts of Decontamination and Dismantlement Activities

3.4.1 Occupational Exposure

Figure 3-1 provides Rancho Seco cumulative site dose and estimates for the decommissioning project. These estimates were developed to provide site management ALARA goals. The goals are verified by summation of actual site dose, as determined by appropriate dosimetry. ALARA estimates are a compilation of work plan (radiation work permit) estimates for the period. This information is in addition to information gathered for reporting of yearly site dose in accordance with the Rancho Seco Quality Manual (RSQM), Appendix A. The annual report of occupational dose meets the guidance of NRC Regulatory Guide 1.16, "Reporting of Operating Information-Appendix A Technical Specifications," [Reference 3-8]. The total nuclear worker exposure during decommissioning is currently estimated to be less than 200 person-rem. This estimate is significantly below the 1,215 person-rem estimate of the FGEIS for immediate dismantlement and below the ten-year SAFSTOR estimate of 664 person-rem.

3.4.2 Public Exposure

Continued application of Rancho Seco's current and future Radiation Protection, Radioactive Waste, Radiological Effluent Technical Specification and Radiological Environmental Monitoring Programs assures public protection in accordance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I. Section 8.6.2 of this LTP contains an evaluation of estimated public exposure as a result of decommissioning activities including the transportation of radioactive waste as compared to the FGEIS.

3.4.3 Estimate of Quantity of Radioactive Material to be Shipped for Disposal or Processing

Rancho Seco has shipped for radioactive disposal approximately 5,560 cubic meters (196,325 cubic feet) through December 31, 2005. The estimate of remaining waste is 11,730 cubic meters (414,206 cubic feet), most of which is very low activity concrete debris from the Reactor Building interior. This volume of waste is bounded by NUREG-0586, (FGEIS) volume for the reference pressurized water reactor of 18,343 cubic meters (647,700 cubic feet).

3.4.4 Solid Waste Activity and Volume

Rancho Seco's Annual Radioactive Effluent Release Report, submitted in accordance with the RSQM and 10 CFR 50.36(a), includes a report on solid waste activity and volumes. This report is submitted before May 1st each year. A summary of solid waste disposal for 1997 through 2005 is provided in Table 3-2. Future updates may be obtained from Rancho Seco for inspection.

Table 3-2
Solid Waste Effluent Release Report Summary

Waste Type	Source	Year	Volume (m^3)	Total Curies	Principle Radionuclides
a.	Spent Resins, filter sludges, evaporator bottoms, etc.	1997	0	0	Co-60, Sr-90, Cs-137, Cs-134, Fe-55, Ni-63, H-3, C-14
		1998	0	0	
		1999	46.5	0.867	
		2000	37.7	1.40	
		2001	26.1	0.03	
		2002	21.1	4.79	
		2003	3.01	0.082	
		2004	0	0	
		2005	0	0	
b.	DAW, contaminated equipment, etc.	1997	0	0	Co-60, Cs-137, Fe-55, Ni-63, C-14, Sr-90, H-3
		1998	196	0.025	
		1999	434	0.89	
		2000	498	3.12	
		2001	422	3.99	
		2002	946	53.7	
		2003	710	13.2	
		2004	799	124	
		2005	784	131	
c.	Irradiated components, control rods, etc.	1997-2002	0	0	Co-60, Ni-63, Fe-55, Eu-152, Eu-154
		2003	0.166	0.599	
		2004	3.17	0.0007	
		2005	0	0	
d.	Other (primary metals, valves, piping)	1997	0	0	Co-60, Fe-55, Ni-63, Cs-137, C-14, Sr-90, H-3
		1998	0	0	
		1999	0.89	0.05	
		2000	11.2	0.147	
		2001	116	1.02	
		2002	246	6.31	
		2003	132	2.35	
		2004	117	7.13	
		2005	6.64	0.008	

3.4.5 Liquid Waste Activity and Volume

Rancho Seco also reports, in accordance with the RSQM, Appendix A, the Annual Radioactive Effluent Release Report, which includes data on liquid waste.. The set of data provided in Table 3-3 provides a compilation of this information. A summary of the liquid waste effluent release reports for 1997 through 2004 is provided below. Liquid effluent release data was not available for 2005 at the time of this LTP submittal; future updates may be obtained from Rancho Seco for inspection. Releases for 2005 and 2006 are expected to be similar to 2004 releases, a small fraction of the limits. Liquid radioactive releases are expected to cease in 2007.

Table 3-3

Liquid Waste Effluent Releases

Year	Tritium Release (Ci)	Dissolved and Entrained Gas Release (Ci)	Alpha Release (Ci)	Other Fission and Activation Release (Ci)	Volume (Liters)	Volume of Dilution Water (Liters)	Max. Dose Commitment - Whole Body (mrem)	Max Dose Commitment - Organ (mrem)
1997	8.15E -03	0.00	0.00	8.07E-05	2.02E+06	1.40E+10	0.026	0.051
1998	5.50E+00	0.00	0.00	9.11E-05	2.27E+07	1.53E+10	0.094	0.147
1999	5.53E-01	0.00	0.00	5.65E-04	1.06E+06	1.69E+10	0.128	0.266
2000	2.64E+00	0.00	0.00	5.13E-04	3.61E+06	1.64E+10	0.139	0.276
2001	3.52E+00	0.00	0.00	6.50E-03	9.82E+06	1.58E+10	0.065	0.093
2002	1.15E+01	0.00	1.52E-05	1.36E-03	1.37E+07	1.68E+10	0.258	0.372
2003	1.82E+01	0.00	2.53E-05	5.55E-03	1.70E+07	1.57E+10	0.666	1.26
2004	8.73E-03	0.00	0.00	3.95E-05	4.28E+06	1.76E+10	0.010	0.023

3.4.6 Gaseous Waste Activity and Volume

Rancho Seco also reports, in accordance with the RSQM, Appendix A, the Annual Radioactive Effluent Release Report, which includes data on gaseous waste. The set of data provided in Table 3-4 provides a compilation of this information. A summary of the liquid waste effluent release reports for 1997 through 2004 is provided below. After all spent fuel was moved to the ISFSI for storage, no source for gases exists. Particulate matter is the only expected material to be released in gaseous effluents after 2002. Gaseous effluent release data was not available for 2005 at the time of this LTP submission; future updates may be obtained from Rancho Seco for inspection. Releases for 2005 and 2006 are expected to be similar to 2004 releases, a small fraction of the limits. Radioactive gaseous releases are expected to cease in early 2008.

Table 3-4
Gaseous Waste Effluent Releases

Year	Fission and Activation Gas Release (Ci)	Iodines (Ci)	Particulates (Ci)	Tritium (Ci)	Whole Body Dose, β (mrads)	Whole Body Dose, γ (mrads)	Organ Dose (mrem)
1997	0.00	0.00	0.00	1.61E+00	0.00	0.00	5.22E-02
1998	0.00	0.00	0.00	2.87E+00	0.00	0.00	9.33E-02
1999	2.84E-02	0.00	0.00	1.94E+00	1.76E-04	1.55E-06	6.59E-02
2000	0.00	0.00	0.00	1.74E+00	0.00	0.00	5.65E-02
2001	0.00	0.00	0.00	9.04E-01	0.00	0.00	2.93E-02
2002	0.00	0.00	1.07E-05	1.41E+00	0.00	0.00	1.11E-01
2003	0.00	0.00	9.91E-06	3.50E-01	0.00	0.00	6.98E-02
2004	0.00	0.00	2.13E-06	4.10E-02	0.00	0.00	1.42E-02

3.5 Site Description after License Release

Currently, no permanent buildings or structures on site are scheduled for demolition. The switchyard is in use for the Cosumnes Power Plant, the IOSB will be used for storage of Class B and C waste, the Administration Building is used as an Emergency Operation Facility for the District in case Sacramento facilities are unavailable and the Secondary Alarm Station is used by ISFSI security personnel. Various other buildings may be used for office space or maintenance activities.

The District may at some future date decide to demolish or refurbish any of the buildings or structures onsite. However, the impacted structures (Reactor Building, Spent Fuel Building and the Auxiliary Building) are unlikely to be reused or demolished in the near future.

Many possible uses for the site or portions of the site have been considered but it will remain District property and the site's most likely use would be for future electric generation due to switchyard access and water availability.

3.6 Coordination with Outside Entities

The decommissioning and termination of Rancho Seco's 10 CFR Part 50 license involves, among others, the US NRC, the US Department of Transportation, US Department of Energy, the State of California (Cal/OSHA, Cal/EPA, State Water Resources Board, Central Valley Regional Water Quality Control Board), Sacramento County and the local fire district. Chapter 8 of this LTP discusses some of the related requirements.

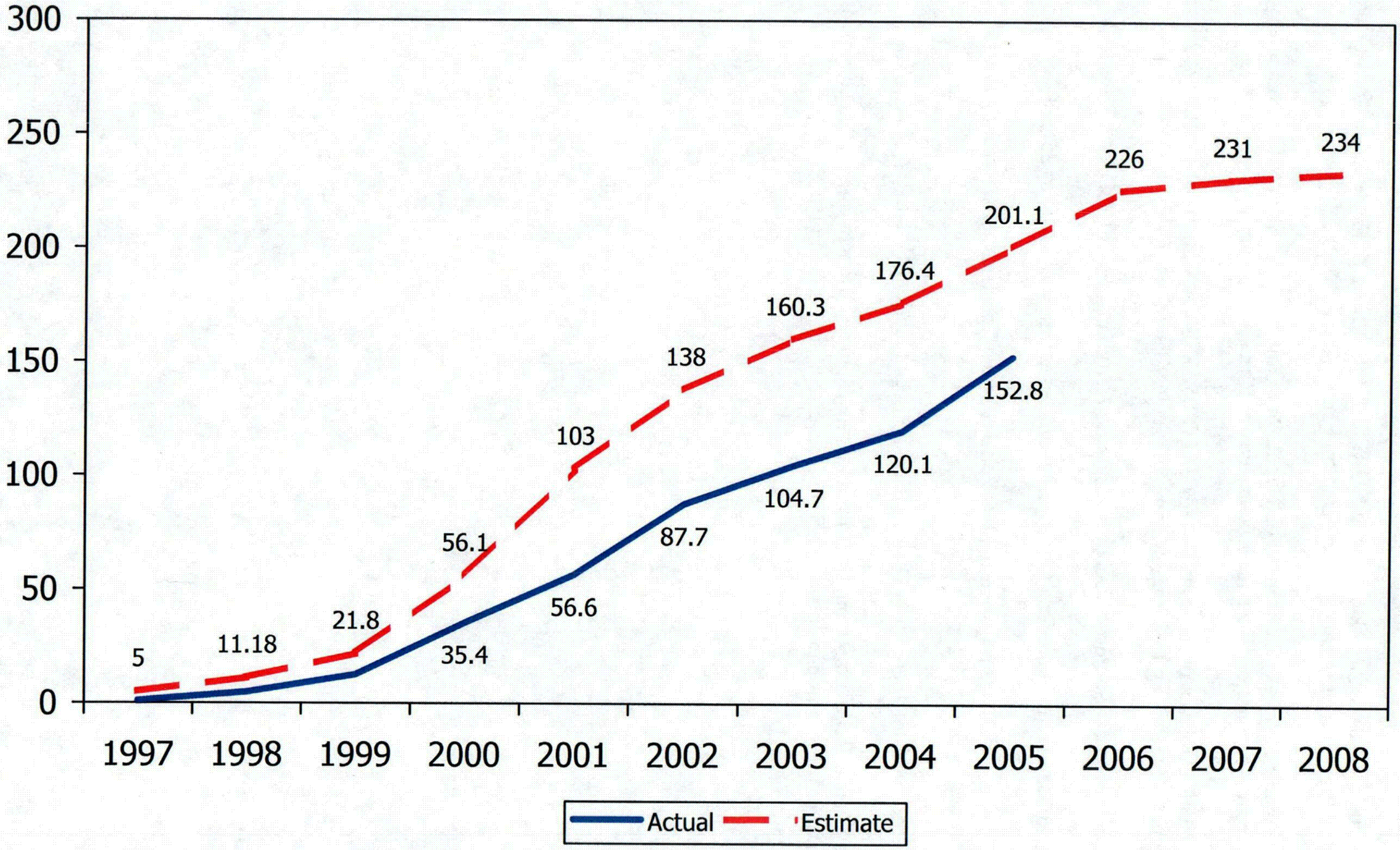

Figure 3-1
Dose-Estimate and Actual (Person-Rem)

3.7 References

3-1 U.S. Nuclear Regulatory Commission NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" (April 2000)

3-2 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (January 1999)

3-3 U.S. Nuclear Regulatory Commission NUREG-0586, "Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities" (August 1998)

3-4 Rancho Seco Post Shutdown Decommissioning Activities Report (PSDAR), Revision 4,

3-5 Rancho Seco Defueled Safety Analysis Report, Amendment 6, October 25, 2004

3-6 Rancho Seco Unit 1, Permanently Defueled Technical Specifications, Amendment 130, September 29, 2005

3-7 Sacramento Municipal Utility District, Rancho Seco Quality Manual

3-8 U.S. Nuclear Regulatory Commission Regulatory Guide 1.16, Revision 4, "Reporting of Operating Information-Appendix A Technical Specifications, August 1975

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APPENDICES

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4.0 SITE REMEDIATION PLAN

4.1 Remediation Actions and ALARA Evaluations

This chapter of the LTP describes various remediation actions that may be used during the decommissioning of Rancho Seco. In addition, the methods used to reduce residual contamination to levels that comply with the NRC's annual dose limit of 25 mrem plus ALARA are described. Finally, the Radiation Protection Program requirements for the remediation are described.

4.2 Remediation Actions

Remediation actions are performed throughout the decommissioning process. The remediation action taken is dependent on the material contaminated. The principal materials that may be subjected to remediation are hardened structural surfaces and soils. Appendix 4-A of this LTP chapter describes the equipment, personnel, and waste costs used to generate a unit cost basis for the remediation actions discussed below.

4.2.1 Structures

Following the removal of equipment and components, structures will be surveyed as necessary and contaminated materials will be remediated or removed and disposed of as radioactive waste. Contaminated structural surfaces will be remediated to a level that will meet the established radiological criteria provided in Chapter 5 of this LTP.

Remediation techniques that may be used for the structural surfaces include washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting. Washing, wiping, abrasive blasting, vacuuming and pressure washing techniques may be used for both metal and concrete surfaces. Scabbling and chipping are mechanical surface removal methods that are intended for concrete surfaces. Activated concrete removal may include using machines with hydraulic-assisted, remote-operated, articulating tools. These machines have the ability to exchange scabbling, shear, chisel and other tool heads.

4.2.1.1 Scabbling and Shaving

The principal remediation method expected to be used for removing contaminants from concrete surfaces is scabbling and shaving. Scabbling is a surface removal process that uses pneumatically operated air pistons with tungsten-carbide tips that fracture the concrete surface to a nominal depth of 0.25 inches at a rate of about 20 ft^2 per hour. The scabbling pistons (feet) are contained in a close-capture enclosure that is connected by hoses to a sealed vacuum and collector system. Shaving uses a series of diamond cutting wheels on a spindle, and performs at similar rates to scabbling. The wheels are also contained in a close-capture enclosure similar to scabbling equipment. The fractured media and dusts from both methods are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute HEPA (high efficiency particulate air filter) filtration devices. Dust and generated debris are collected and controlled during the operation.

4.2.1.2 Needle Guns

A second form of scabbling is accomplished using needle guns. The needle gun is a pneumatic air-operated tool containing a series of tungsten carbide or hardened steel rods enclosed in a housing. The rods are connected to an air-driven piston to abrade and fracture the media surface. The media removal depth is a function of the residence time of the rods over the surface. Typically, one to two millimeters are removed per pass. Generated debris collection, transport and dust control are accomplished in the same manner as for scabbling. Use of needle guns for removal and chipping of media is usually reserved for areas not accessible to normal scabbling operations. These include, but are not limited to inside corners, cracks, joints and crevices. Needle gunning techniques can also be applied to painted and oxidized surfaces.

4.2.1.3 Chipping

Chipping includes the use of pneumatically operated chisels and similar tools coupled to vacuum-assisted collection devices. Chipping activities are usually reserved for cracks and crevices but may also be used in lieu of concrete saws to remove pedestal bases or similar equipment platforms. This action is also a form of scabbling.

4.2.1.4 Sponge and Abrasive Blasting

Sponge and abrasive blasting are similar techniques that use media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide, and walnut hulls. Sponge blasting is less aggressive incorporating a foam media that, upon impact and compression, absorbs contaminants. The medium is collected by vacuum and the contaminants washed from the medium for reuse.

Abrasive blasting is more aggressive than sponge blasting but less aggressive than scabbling. Both operations use intermediate air pressures. Sponge and abrasive blasting are intended for the removal of surface films and paints. Abrasive blasting is evaluated as a remediation action and the cost is comparable to sponge blasting with an abrasive media.

4.2.1.5 Pressure Washing

Pressure washing uses a hydrolazer-type nozzle of intermediate water pressure to direct a jet of pressurized water that removes surficial materials from the suspect surface. A header may be used to minimize over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing.

4.2.1.6 Washing and Wiping

Washing and wiping techniques are actions that are normally performed during the course of remediation activities and will not always be evaluated as a separate ALARA action. When washing and wiping techniques are used as the sole means to reduce residual contamination below DCGL levels, ALARA evaluations are performed. Washing and wiping techniques used as a housekeeping or good practice measure will not be evaluated. Examples of washing and wiping activities for which ALARA evaluations would be performed include:

- Decontamination of stairs and rails,
- Decontamination of structural materials, metals or media for which decontamination reagents may be required, or
- Structure areas that do not provide sufficient access for utilization of other decontamination equipment such as pressure washing.

4.2.1.7 Grit Blasting

Most contaminated piping will be removed and disposed of as radioactive waste. Any remaining contaminated piping buried or embedded in concrete may be remediated using methods such as grit blasting. Grit blasting uses grit media such as garnet or sand under intermediate air pressure directed through a nozzle that is pulled through the closed piping at a fixed rate. The grit blasting action removes the interior surface layer of the piping. A HEPA vacuum system maintains the sections being cleaned under negative pressure and collects the media for reuse or disposal. The final system pass is performed with clean grit to remove any residual contamination.

4.2.1.8 Removal of Activated/Contaminated Concrete

Removal of concrete may be accomplished using a machine mounted, remote-operated articulating arm with exchangeable actuated hammer and bucket (sawing, impact hammering and expansion fracturing may also be employed). As concrete is fractured and rebar exposed, the metal is cut using flame cutting (oxygen-acetylene or other) equipment. Bulk concrete such as walls or floors may be removed as intact sections after sawing with blades, wires or other cutting methods. Removal may also be accomplished by demolition using power impact tools or explosives.

The debris media are transferred into containers for later disposal. Dusts, fumes and generated debris are collected locally or in bulk room exhaust and as necessary, controlled using temporary enclosures coupled with close-capture HEPA filtration systems and controlled water misting. Any remaining loose media are removed by pressure washing or dry vacuuming using a HEPA filter equipped wet-dry vacuum.

4.2.2 Soil

Soil contamination above the site specific DCGL will be removed and disposed of as radioactive waste. Operational constraints and dust control will be addressed in site excavation and soil control procedures. In addition, work package instructions for remediation of soil may include additional constraints and mitigation or control methods. The site characterization process established the location, depth and extent of soil contamination. As needed, additional investigations will be performed to ensure that any changing soil contamination profile during the remediation actions is adequately identified and addressed. It should also be noted that soil remediation volume estimates in the LTP may vary from section to section, as appropriate, depending on their use, e.g., decommissioning cost estimates, ALARA evaluations, or dose assessment. Section 5.4.1.2 of this LTP discusses soil sampling and survey methods.

Soil remediation equipment will include, but not be limited to, back and track hoe excavators. As practical, when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique may be used. This simple methodology minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. Remediation of soils will include the use of established Excavation Safety and Environmental Control procedures. Additionally, soil handling procedures and work package instructions will augment the above guidance and procedural requirements to ensure adequate erosion, sediment, and air emission controls during soil remediation.

4.3 Remediation Activities Impact on the Radiation Protection Program

The Radiation Protection Program approved for decommissioning is similar to the program in place during commercial power operation. During power operations, contaminated structures, systems and components were decontaminated in order to perform maintenance or repair actions. The techniques used during operations are the same or similar to the techniques used during decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas.

Decommissioning does not present any new challenge to the Radiation Protection Program above those encountered during normal plant operation and refueling. Decommissioning planning allows radiation protection personnel to focus on each area of the site and plan each activity well before execution of the remediation technique.

Low levels of surface contamination are expected to be remediated by washing and wiping. These techniques have been used throughout the operational history of the facility. Water washing with detergent has been the method of choice for large area decontamination. Wiping with detergent soaked or oil-impregnated media has been used on small items, overhead spaces and small hand tools to remove surface contaminants. These same techniques will be applied to remediation of lightly contaminated structure surfaces during remediation actions.

Intermediate levels of contamination and contamination on the internal surfaces of piping or components have been subjected to high-pressure washing, hydrolazing or grit blasting in the past. The refueling cavity has been decontaminated by both pressure washing and hydrolazing. Pipes, surfaces and drain lines have been cleaned and hot spots removed using hydrolazing, sponge blasting or grit blasting. Small tools, hoses and cables have been pressure washed in a self-contained glove box to remove surface contamination. These methods will be used to reduce contamination on moderately contaminated exterior surfaces as well as internal surfaces of pipes during decommissioning.

Scabbling or other surface removal techniques will reduce high levels of contamination, including that present on contaminated concrete. Concrete cutting or surface scabbling has been used at Rancho Seco in the past during or prior to installation of new equipment or structures both outside and inside the RCA.

Abrasive water jet cutting will be used to section the reactor vessel and mechanical cutting was be used to section reactor internals. Abrasive water jet cutting uses actions similar to hydrolazing and grit blasting that have been used at the site in the past. Mechanical cutting was used at this facility during past operations. The current Radiation Protection Program provides adequate controls for these actions.

The decommissioning organization is experienced in and capable of applying these remediation techniques on contaminated systems, structures or components during decommissioning. The Radiation Protection Program is adequate to safely control the radiological aspects of this work. Because the activities expected during decommissioning are the same or similar to those encountered during operations, as described above, no changes to the program are necessary in order to ensure the health and safety of the workers and the public.

4.4 ALARA Evaluation

As described in Chapter 6 of this LTP, dose assessment scenarios were evaluated for the residual contamination that could remain on structural surfaces and soils. The ALARA analysis is based on the same industrial worker, industrial worker building occupancy and containment building renovation/demolition scenarios used for RESRAD and RESRAD-BUILD derivation of single nuclide DCGLs.

4.4.1 Dose Models

To calculate the cost and benefit of averted dose for the ALARA calculation, certain parameters such as size of contaminated area and population density are required. This information was developed as a part of the dose models described in Chapter 6 and the Final Status Survey Program in Chapter 5 and is summarized below.

4.4.1.1 Industrial Worker Scenario for Surface and Subsurface Soil Exposure

The average member of the critical group is defined as a District employee or contractor who is allowed occupational access to areas of the site (which were classified as impacted prior to license termination) over the course of his/her employment. The assumption is made that occupancy would be limited to a 50-workweek year (2,000 hours per year). It was further assumed that the industrial worker would spend 50 percent of his/her time indoors and 50 percent outdoors while onsite.

The drinking water pathway is not suppressed – there are currently four potable water wells existing on the 2,480-acre site. Three of these wells are upgradient of the impacted area; however, the fourth well is in the northern portion of the impacted area and is used for potable water purposes.

4.4.1.2 Industrial Worker Scenario for Building Occupancy Exposure

The average member of the critical group is defined as a District employee or contractor who is allowed occupational access to structures (which were classified as impacted prior to license termination) of the site over the course of his/her employment. The occupancy assumed is the 45 hours per week used in NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination from Decommissioning – Parameter Analysis," [Reference 4-1].

The building occupancy survey unit floor area size is 137 m^2 based on the probabilistic sensitivity analysis derivation found in LTP Section 6.7.3. ALARA cost analyses are based on an assumption that only the 137 m^2 floor area requires remediation. This is conservative since including the walls would increase remediation cost without significantly increasing the benefit of averted dose.

4.4.2 Methods for ALARA Evaluation

The ALARA evaluations were performed in accordance with the guidance in Appendix N to NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria," [Reference 4-2]. The principal equations used for the calculations are presented in Section 4.5. The evaluation determines if the benefit of the dose averted by the remediation is greater or less than the cost of the remediation. When the benefit is greater than the cost, additional remediation is required. Conversely when the benefit is less than the cost, additional remediation is not required.

4.4.3 Remediation Methods and Cost

For the Rancho Seco facility the remediation techniques examined are scabbling, pressure water washing, wet and dry wiping, grit blasting for embedded and buried piping, grit blasting of surfaces and soil excavation. The principal remediation method expected to be used is scabbling, which is intended to include needle guns and chipping. The total cost of each remediation method is provided in Appendix 4-A. The cost inputs are defined in Section 4.5.1, Calculation of Total Cost.

4.4.3.1 Concrete Surfaces

The characterization data for concrete surfaces at the Rancho Seco facility indicates that a major fraction of the contamination occurs in the top ten millimeters of the concrete. The ALARA evaluation was performed by bounding the cost estimate for a scabbled depth of 0.125 and 0.25 inches. For each evaluation the same manpower cost is used. However, the manpower and equipment costs for the lower bounding depth do not include compressor and consumable supply costs which adds some conservatism to the cost estimate, i.e., biases the cost low. The major variables for the bounding conditions are the costs associated with manpower and waste disposal.

4.4.3.2 Structure Activated Concrete

Concrete activation is associated with the containment building. Characterization of the reactor bioshield and loop area concrete has provided information regarding the identification, concentration, and distribution of the radionuclides. In addition to the observed concrete activation products, the concrete surfaces in the containment structure are radioactively contaminated by the deposition and transport of fluids and airborne distribution that occurred during plant operation. Based upon the difficulty that these activated and contaminated characteristics have raised in demonstrating compliance with the dose criteria in 10 CFR 20, Subpart E at other commercial reactor decommissioning projects; Rancho Seco has decided to remove and dispose of all containment building interior concrete without having performed an ALARA analysis.

4.4.4 Remediation Cost Basis

The cost of remediation depends on several factors such as those listed below. This section describes the attributes of each remediation method that affect cost. The detailed cost estimates for each method are provided in Appendix 4-A.

- Depth of contaminants;
- Surface area(s) of contamination relative to total;
- Types of surfaces: vertical walls, overhead surfaces, media condition;
- Consumable items and equipment parts;
- Cleaning rate and efficiency (decontamination factor);
- Work crew size;

- Support activities such as, waste packaging and transfer, set up time and interfering activities for other tasks; and
- Waste volume.

4.4.4.1 Scabbling

NUREG/CR-5884, Volume 2, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," [Reference 4-3] states that scabbling can be effectively performed on smooth concrete surfaces to a depth of 0.125 inches at a rate of 115 ft^2 per hour. The scabbling pistons (feet) are contained in a close-capture enclosure that is connected by hoses to a sealed vacuum and collector system. The waste media and dust are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute HEPA filtration devices. Dust and generated debris are collected and controlled during the operation.

The unit cost is presented in Table 4-2. Scabbling the room assumes that 100% of the concrete surface contains contamination at levels equal to the DCGL and that 12.5% of this residual activity is removed by each pass and that it takes eight passes to effectively remove all the residual activity. The debris is vacuumed into collectors that are transferred to containers for truck or rail shipments. For the evaluation, the truck container is assumed to carry 13.5 m^3 of concrete per shipment based on the NUREG-1757, Volume 2 guidance contained in Table 4-1.

Based on evaluation of concrete core samples, scabbling is expected to be the principal method used for remediation of concrete surfaces. The cost elements used to derive the unit costs for the ALARA evaluation are listed in Appendix 4-A. The methods for calculating total cost are provided in Section 4.5.1.

4.4.4.2 Pressure Water Washing

The unit costs provided in Table 4-2 for pressure water washing were established by assuming that 20,312 m^2 of the site structures' surface area is pressure washed using the surface area example of NUREG/CR-5884, Volume 1, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," [Reference 4-4], Table 3.22. This information was used to provide a cost per square meter factor. Appendix 4-A provides the cost details. The equipment consists of a hydrolazer and when used, a header assembly. The hydrolazer type nozzle directs the jet of pressurized water that removes surficial materials from the concrete. The header minimizes over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing. The cleaning speed is approximately 240 ft^2 (22.3 m^2) per hour and the process generates about 5.4 liters of liquid per square meter as discussed in NUREG/CR-5884, Volume 2. The contamination reduction rates are dependent on the media in which the contaminants are fixed, the composition of the contaminants, cleaning reagents used and water jet pressure. Mitigation of loose contaminants is high. Reduction of hard-to-remove surface contamination is approximately 25% for the jet pressure and cleaning speed used. The use of reagents and slower speeds can provide better contamination reduction rates but at proportionally higher costs. The formula associated with the cost elements is provided in Section 4.5.1 and the cost elements are provided in Appendix 4-A.

4.4.4.3 Wet and Dry Wiping

The unit costs provided in Table 4-2 for washing and wiping assume the same 20,312 m^2 of the site structures' surface area as discussed in Section 4.4.4.2 is washed and wiped. The information is used to develop a cost per square meter. Appendix 4-A provides the detailed costs. Wet wiping consists of using a cleaning reagent and wipes on surfaces that cannot be otherwise cleaned or decontaminated. Dry wiping includes the use of oil-impregnated media to pick up and hold contaminants. The cleaning rate of these actions is estimated at 2.8 m^2/hr (~2 min per ft^2), based on industry experience such as that described in the Maine Yankee License Termination Plan [Reference 4-5]. This action is labor intensive. The action is effective for the removal of loose contaminants and reduction of surface contaminants, especially when cleaning reagents are used. Waste generation is about 0.005 m^3 per hour (NUREG-5884, Volume 2). Decontamination factors vary and are dependent on factors such as the reagents that are used, the level of wiping effort and the chemical and physical composition of the contaminant. The contamination reduction efficiency used for wet and dry wiping is 20 percent. Removal of loose contaminants, oil and grease is very effective (100 percent). The formula associated with the cost elements is provided in Section 4.5.1. Appendix 4-A lists the cost elements used for the evaluation.

4.4.4.4 Grit Blasting (Embedded Piping)

The cost for grit blasting was established by assuming that 5,354 linear feet, which is the estimated total of embedded piping to remain at Rancho Seco, is decontaminated. For the evaluation, the entire interior surface is assumed to require decontamination and the internal diameter is assumed at 4 inches (typical drain line dimensions). The grit blasting system is comprised of a hopper assembly that delivers a grit medium (garnet or sand) at intermediate air pressures through a nozzle that is pulled at a fixed rate (~1 ft/min) through the piping. A HEPA vacuum system maintains the piping system under a negative pressure and collects the grit for reuse (cyclone separator) or disposal. Usually several passes are required to effectively clean the piping to acceptable residual radioactivity levels. The contamination reduction efficiency used for grit blasting is 95 percent. This reduction rate can vary depending on radial bends in piping, reduction and expansion fittings, pipe material composition, physical condition and the plate-out mechanisms associated with the contaminants and effluents. The final pass is made with clean grit to mitigate the possibility of loose residual contaminants associated with previous cleaning passes. Grit decontamination factors are related to pressure, nozzle size, grit media and the number of passes made. A nominal grit usage rate of one pound per linear foot is used in the calculation. This cost unit information is provided as cost per linear foot factor and is also converted to m^2 for evaluation. Appendix 4-A provides the cost details used to derive unit cost. The formula associated with the cost elements is provided in Section 4.5.1.

4.4.4.5 Sponge and Abrasive Blasting

Sponge and abrasive blasting uses media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide and walnut hulls. The operation uses intermediate air pressures as that described for grit blasting. The operation uses a closed-capture system and air filtration system to mitigate loose and airborne radioactivity. The system includes a cyclone or similar separation system to collect the generated media. The operation is intended for removal of surficial films. The removal efficiency and depth are a function of the surface, abrasive mix, air pressure, grit media, and speed or number of passes performed over the suspect surface. Surface cleaning rates are about 30 ft^2/hr (2.8 m^2/hr). For the rate given, the removal depth using aluminum oxide grit will range from less than 1 to as much as 3 millimeters. Abrasive blasting techniques are often used for film and paint removal and are less aggressive than scabbling.

4.4.4.6 Soil Excavation

The unit costs provided in Table 4-2 for soil excavation were established by assuming 52,972 ft^3 (1,500.0 m^3) of soil is excavated from the site. This information was used to generate a cost per cubic meter for soil remediation. The equipment consists of an excavator that first moves the soil at the contaminated depth interface into a container or if necessary, a pile that is scooped into a staged shipping container. When filled, the container is moved from the excavation area with a forklift. Contamination reduction is assumed at 95%. The operation is performed using two equipment operators and two laborers. Costs for radiation protection support activities and supervision are also included. The formula associated with the cost elements is provided in Section 4.5.1 and the cost elements are provided in Appendix 4-A.

4.5 Unit Cost Estimates

In order to effectively perform ALARA evaluations and remediation actions, unit cost values are required. These values are used to perform the NUREG-1757, Volume 2 cost-benefit analysis. Table 4-2 lists the unit costs of the remediation methods anticipated to be used at Rancho Seco.

4.5.1 Calculation of Total Cost

In order to evaluate the cost of remediation actions NUREG-1757, Volume 2, Appendix N provides the elements necessary to derive the costs that are compared to the benefits. The total cost, $Cost_T$, which is balanced against the benefits, has several components defined as follows in Appendix N, Equation (N-3):

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{other}$$

Equation 4-1

where:

$Cost_R$ = monetary cost of the remediation action (may include "mobilization" costs);

$Cost_{WD}$ = monetary cost for transport and disposal of the waste generated by the action;

$Cost_{ACC}$ = monetary cost of worker accidents during the remediation action;

$Cost_{TF}$ = monetary cost of traffic fatalities during transporting of the waste;

$Cost_{WDose}$ = monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility;

$Cost_{PDose}$ = monetary cost of the dose to the public from excavation, transport, and disposal of the waste; and

$Cost_{other}$ = other costs as appropriate for the particular situation.

4.5.1.1 Transport and Disposal of the Waste

In accordance with the guidance provided in NUREG-1757, Volume 2, the cost of waste transport and disposal, $Cost_{WD}$, may be evaluated according to Equation 4-2 below:

$$Cost_{WD} = V_A \times Cost_V$$

Equation 4-2

where:

V_A = volume of waste produced, remediated in units of m^3; and

$Cost_V$ = cost of waste disposal per unit volume, including transportation cost, in units of $\$/m^3$.

4.5.1.2 Nonradiological Risks

Also in accordance with the guidance provided in NUREG-1757, Volume 2, the cost of nonradiological workplace accidents, $Cost_{ACC}$, may be evaluated using Equation 4-3 below:

$$Cost_{ACC} = \$3,000,000 \times F_W \times T_A$$

Equation 4-3

where:

\$3,000,000 = monetary value of a fatality equivalent to \$2,000/person-rem (see pages 11-12 of NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," [Reference 4-6]);

F_W = workplace fatality rate in fatalities/hour worked; and

T_A = worker time required for remediation in units of worker-hours.

4.5.1.3 Transportation Risks

Also, the cost of traffic fatalities incurred during the transportation of waste, $Cost_{TF}$, may be calculated using Equation 4-4 below:

$$Cost_{TF} = \$3,000,000 \times \left(\frac{V_A}{V_{SHIP}}\right) \times F_T \times D_T$$

Equation 4-4

where:

V_A = volume of waste produced in units of m^3,

V_{SHIP} = volume of a truck shipment in m^3,

F_T = fatality rate per truck-kilometer traveled in units of fatalities/truck-km, and

D_T = distance traveled in km.

The actual parameters will depend on Rancho Seco's planned method of waste transport. This may include a mix of trucking and rail transport to get the waste to the disposal site. In these cases, the cost would be equivalent to the total fatalities likely from the rail transport and the limited trucking, not just the trucking alone.

4.5.1.4 Worker Dose Estimates

The cost of the remediation worker dose, $Cost_{WDose}$, can be calculated as shown in Equation 4-5 below:

$$Cost_{WDose} = \$2,000 \times D_R \times T$$

Equation 4-5

where:

D_R = total effective dose equivalent (TEDE) rate to remediation workers in units of rems/hr, and

T = time worked (site labor) to remediate the area in units of person-hour.

4.5.1.5 Default Parameter Values

In accordance with the guidance provided in NUREG-1757, Volume 2, parameter values found acceptable by the NRC for performing the calculations provided in Equations 4-2 through 4-5 and the source of the parameter values, are provided in Table 4-1.

Table 4-1
Acceptable Parameter Values for Use in ALARA Analyses

Parameter	Parameter Value	Reference Source
Workplace accident fatality rate, F_W	4.2E-08/hr	NUREG–1496, Volume 1 [Reference 4-7] and NUREG–1496, Volume 2 [Reference 4-8], Appendix B, Table A.1
Transportation fatal accident rate, F_T	Trucks: 3.8E-08/km	NUREG–1496, Volume 2, Appendix B, Table A.1
Dollars/person-rem	$2,000	NUREG/BR–0058 [Reference 4-9], Section 4.3.5
Monetary discount rate, r	0.07/y for the first 100 years and 0.03/y thereafter, or 0.07 for buildings and 0.03 for soil	NUREG/BR–0058, Section 4.3.5
Number of years of exposure, N	Buildings: 70 years Soil: 1000 years	NUREG–1496, Volume 2, Appendix B, Table A.1
Population density, P_D	Building: 0.09 person/m^2 Land: 0.0004 person/m^2	NUREG–1496, Volume 2, Appendix B, Table A.1
Excavation, monitoring, packaging, and handling soil	Soil: 1.62 person-hours/m^3 of soil	NUREG–1496, Volume 2, Appendix B, Table A.1
Waste shipment volume, V_{SHIP}	Truck: 13.6 m^3/shipment	NUREG–1496, Volume 2, Appendix B, Table A.1

4.5.2 Calculation of Benefits

In order to evaluate the benefits of remediation actions NUREG-1757, Volume 2, Appendix N provides the elements necessary to derive the benefits that are compared to the total cost. As discussed in Section 4.4.1, calculation of the benefits of remediation actions is based on an industrial worker scenario for surface and subsurface soil exposure and for building occupancy exposure. The benefit from collective averted dose, B_{AD}, is calculated by determining the present worth of the future collective averted dose and multiplying it by a factor to convert the dose to monetary value:

$$B_{AD} = \$2{,}000 \times PW(AD_{collective})$$

Equation 4-6

where:

B_{AD} = benefit from an avoided dose for a remediation action, in current U.S. dollars;

\$2,000 = value in dollars of a person-rem averted (see NUREG/BR–0058); and

$PW(AD_{collective})$ = present worth of a future collective averted dose.

A value acceptable to the NRC for a collective dose is \$2,000 per person-rem averted, discounted for a dose averted in the future (see Section 4.3.5 of NUREG/BR–0058, Revision 4). For doses averted within the first 100 years (applicable to structural surfaces), a discount rate of 7 percent was used. For doses averted beyond 100 years (applicable to surface and subsurface soil), a 3 percent discount rate was used.

The present worth of the future collective averted dose can be estimated from Equation 4-7, for relatively simple situations:

$$PW(AD_{collective}) = P_D \times A \times 0.025 \times F \times \frac{Conc}{DCGL_W} \times \frac{1-e^{-(r+\lambda)N}}{r+\lambda}$$

Equation 4-7

where:

P_D = population density for the critical group scenario in people/m^2;

A = area being evaluated in square meters (m^2);

0.025 = annual dose to an average member of the critical group from residual radioactivity at the Derived Concentration Guideline Level ($DCGL_W$) concentration in rem/y;

F = effectiveness, or fraction of the residual radioactivity removed by the remediation action;

$Conc$ = average concentration of residual radioactivity in the area being evaluated in units of activity per unit area for buildings or activity per unit volume for soils;

$DCGL_W$ = derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 25 mrem/y to the average member of the critical group, in the same units as "*Conc*";

r = monetary discount rate in units per year;

λ = radiological decay constant for the radionuclide in units per year; and

N = number of years over which the collective dose will be calculated.

The present worth of the benefit calculated by Equation 4-7, above, assumes that the peak dose occurs in the first year. The $DCGL_W$ used is the single nuclide $DCGL_W$ derived in LTP Chapter 6 to show compliance with the 25 mrem/y dose limit. The population density, P_D, is based on the dose scenario used to demonstrate compliance with the dose limit. The factor at the far right of the equation, which includes the exponential terms, accounts for both the present worth of the monetary value and radiological decay.

4.5.3 Residual Radioactivity Levels that are ALARA

NUREG-1757, Volume 2, Appendix N, also provides the guidance necessary to determine if residual levels of radioactivity are ALARA. The residual radioactivity level that is ALARA is the concentration, *Conc*, at which the benefit from removal equals the cost of removal. If the total cost, $Cost_T$, is set equal to the present worth of the collective dose averted in Equation 4-7, the ratio of the concentration, *Conc*, to the $DCGL_W$ can be determined from Equation 4-8 below (derivation shown in NUREG-1757, Volume 2, Section N.5).

$$\frac{Conc}{DCGL_W}=\frac{Cost_T}{\$2{,}000\times P_D\times 0.025\times F\times A}\times\frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

Equation 4-8

All the terms in Equation 4-8 are as defined previously.

Equation 4-9 may be derived from Equation 4-8 to perform the ALARA evaluation in the presence of multiple radionuclides as follows:

$$\frac{Conc}{DCGL_W}=\frac{Cost_T}{\$2{,}000\times P_D\times 0.025\times F\times A}\times\left[\frac{\frac{r+\lambda}{1-e^{-(r+\lambda)N}}}{1}\right]$$

The right term of the above equation is then multiplied by 1 as follows:

$$\frac{Conc}{DCGL_W}=\frac{Cost_T}{\$2{,}000\times P_D\times 0.025\times F\times A}\times\left[\frac{\frac{r+\lambda}{1-e^{-(r+\lambda)N}}}{1}\right]\times\left[\frac{\frac{1-e^{-(r+\lambda)N}}{r+\lambda}}{\frac{1-e^{-(r+\lambda)N}}{r+\lambda}}\right]$$

Equation 4-8 is then expressed as:

$$\frac{Conc}{DCGL_W}=\frac{Cost_T}{\$2{,}000\times P_D\times 0.025\times F\times A\times\left[\frac{1-e^{-(r+\lambda)N}}{r+\lambda}\right]}$$

For multiple radionuclides the denominator must be summed over all radionuclides as shown below:

$$\frac{Conc}{DCGL_W}=\frac{Cost_T}{\sum_i^n \$2{,}000\times P_D\times 0.025\times Df_i\times F\times A\times\left[\frac{1-e^{-(r+\lambda_i)N}}{r+\lambda_i}\right]}$$

Equation 4-9

where:

i = radionuclide "*i*",
n = total of all radionuclides, and
Df_i = dose fraction of radionuclide "*i*"

and:

$$Df_i=\frac{\frac{nf_i}{DCGL_{W_i}}}{\sum_i^n \frac{nf_i}{DCGL_{W_i}}}$$

where:

nf_i = nuclide fraction of the mixture radionuclide

4.6 Radionuclides Considered for ALARA Calculations

As discussed in LTP Chapter 6, Section 6.4.1, the site-specific suite of radionuclides identified for use at Rancho Seco contains 26 radionuclides. Only six of these radionuclides have been identified above minimum detectable concentration (MDC) levels in soil samples while 21 have been identified at least one time in structural surface samples. For purposes of the ALARA calculations, only Co-60 and Cs-137 were used along with their associated DCGL values (adjusted DCGL value for Co-60 and surrogate DCGL value for Cs-137) and nuclide fractions. Cs-137 was used as a surrogate radionuclide for the other 19 radionuclides with the surrogate DCGL value used to classify survey units. Rancho Seco Decommissioning Technical Basis Document DTBD-05-015, "Rancho Seco Nuclear Generating Station Structure Nuclide Fraction and DCGLs," [Reference 4-10] provides the bases for the structural surface nuclide fractions and the surrogate DCGL value. Rancho Seco DTBD-05-014, "Rancho Seco Nuclear Generating Station Surface Soil Nuclide Fraction and DCGL," [Reference 4-11] provides the bases for the surface soil nuclide fractions and the surrogate DCGL value.

4.7 ALARA Calculation Results

The final ALARA calculations were performed by comparing the total remediation cost to the benefit of averted dose using Equation 4-9. The calculations were described in detail in Sections 4.5.1, 4.5.2 and 4.5.3. The results for each remediation method, for both the Industrial Worker (for soils) and (Industrial Worker) Building Occupancy scenarios, are provided in Table 4-2. Since the *Conc/DCGL$_W$* values are greater than 1 for all remediation methods and scenarios, no remediation below the NRC 25 mrem/y dose limit is required.

Table 4-2

ALARA Evaluation Results

Remediation Action	Unit Costs ($ per ft, m^2 or m^3)	*Conc/DCGL$_W$* Ratio
Pressure Washing and Vacuuming	15.31	1.31
Wiping/Washing	58.87	6.31
Concrete Scabbling(Upper Bound)	67.02	5.75
Concrete Scabbling (Lower Bound)	33.36	5.72
Grit Blasting Surfaces (Upper Bound)	96.88	2.19
Grit Blasting Surfaces (Lower Bound)	80.58	1.82
Grit Blasting Embedded/Buried Piping	27.39	42.77
Soil Excavation	2,679.82	1142.00

4.8 References

4-1 U.S. Nuclear Regulatory Commission NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination from Decommissioning – Parameter Analysis, Draft Report for Comment," October 1999

4-2 U.S. Nuclear Regulatory Commission, NUREG-1757, Volume 2, Final Report, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria," September 2003

4-3 U.S. Nuclear Regulatory Commission, NUREG/CR-5884, Volume 2, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," Draft Report for Comment, October 1993

4-4 U.S. Nuclear Regulatory Commission, NUREG/CR-5884, Volume 1, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," Draft Report for Comment, October 1993

4-5 "License Termination Plan," submitted by Maine Yankee Atomic Power Company, Revision 3, October 15, 2002

4-6 U.S. Nuclear Regulatory Commission, NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," December 1995

4-7 U.S. Nuclear Regulatory Commission, NUREG–1496, Volume 1, Final Report, "Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," July 1997

4-8 U.S. Nuclear Regulatory Commission, NUREG–1496, Volume 2, Final Report, "Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," July 1997

4-9 U.S. Nuclear Regulatory Commission, NUREG/BR–0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," September 2004

4-10 Rancho Seco Decommissioning Technical Basis Document DTBD-05-015, Revision 0, "Rancho Seco Nuclear Generating Station Structure Nuclide Fraction and DCGLs"

4-11 Rancho Seco Decommissioning Technical Basis Document DTBD-05-014, Revision 0, "Rancho Seco Nuclear Generating Station Surface Soil Nuclide Fraction and DCGL"

A.1 General

This Appendix provides the unit cost values used to develop the total cost C_T as defined in Section 4.5.1.

Remediation Activity Rates

Remediation activity rates were provided based on previous experience, from published literature, or from groups or vendors currently performing these or similar activities. Past operational experience was also used in developing the rates.

Contingency

A contingency of 0.25 was added to the manpower hours. Scabbling (the primary activity) was bounded using cost and manpower associated with the volume of concrete (disposal cost) for remediation of 0.125 inches versus using a compressor, consumable materials and the volume of concrete (disposal cost) for remediation of 0.25 inches of concrete.

Equipment

Equipment costs were developed based on the cost of buying specific equipment and whenever possible prorating the cost over the task activities. Rental rates are also included for specific equipment such as forklifts and excavators. Consumable supplies and parts were included in the cost for equipment. Shipping containers were included with shipment costs.

Mobilization and Demobilization Costs

Costs were conservatively included for delivery and pick up of equipment. Anticipated costs to stage and move equipment from location to location were also included.

Waste Disposal Cost

Disposal costs for generated waste were based on an average total disposal cost of $2,500/m^3. This average cost includes packaging, transportation and disposal fees. The transportation component of this average cost is based on the average transportation cost of using either rail or highway hauling from the Rancho Seco site to Clive, Utah (Energy*Solutions*[1] site). The details of the average total disposal cost of $2,500 are considered proprietary values defined by negotiated contract.

The Clive, Utah round trip distance from the Rancho Seco site by highway is 1,223 miles (1,968 km). The distance for rail shipments is considerably further than that for highway shipments because of the route rail shipments must follow. The highway shipment distance of 1,968 km (D_T) was used as a conservative value for the calculation of C_T since it results in a lower transportation cost.

The volume for highway hauling (V_{SHIP}) used for the calculation of C_T was 13.6 m^3 as specified in Table 4-1. The distance and haul volume are used for determining transport accident cost in

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[1] Energy*Solutions* was previously Envirocare of Utah

accordance with NUREG-1757, Volume 2 and Section 4.5.1.3. The impact to total cost of this item is minimal.

Worker Accident Costs

To determine worker accident cost in accordance with NUREG-1757, Volume 2 and Section 4.5.1.2, the same hours input for labor cost were used for worker accident cost.

Worker Dose

Costs associated with worker dose are a function of the hours worked and the workers' radiation exposure for the task. A value of 2 mrem/hr per work crew or 3 mrem/hr per work crew for work crew dose depending upon remediation action was based on the assumed dose rate used for worker dose calculations in NUREG/CR-5884 Volume 2.

Labor Costs

The individual cost for the applicable disciplines, e.g., laborer, equipment operator, health physics technicians, were developed into an hourly crew rate for the task and based on guidance provided by NUREG/CR-5884, Volumes 1 and 2. Manpower costs assumptions were also based on NUREG/CR-5884. The NUREG/CR-5884 manpower cost assumptions are based on 1993 dollars and were not escalated to the projected time remediation activities would occur for the following C_T calculations. This is considered a conservative approach because escalating manpower costs or using current contracted rates would only raise C_T, thus raising the $Conc/DCGL_W$ ratio. It is important to note that the total work hours for a normal day were used and not adjusted for personnel breaks, ALARA meetings or ingress and egress from an area.

Unit Cost

The sum of all the cost elements was divided by the applicable unit (m^2, m^3 or linear feet) to provide a unit cost for the activity. Other cost units for cost per hour or linear foot were also developed in the same fashion. The tables to follow provide the crew cost per hour but do not provide the individual hourly rates for individual disciplines. These values are however included in the supporting calculation.

A.2 Pressure Water Washing And Vacuuming

Area Evaluated For Unit Cost Determination:	20,312 m^2 (218,636 ft^2) per NUREG/CR-5884, Volume 1, Table 3.22
Crew Composition	2 Laborers, 1 Craft, 0.5 HP Technician and 0.5 Crew Leader per NUREG/CR-5884, Volume 2, Appendix C
Hourly Crew Cost:	\$148.27 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	22.3 m^2/h (240 ft^2/hr) per NUREG/CR-5884, Volume 2, Appendix C

Contamination Removed:	25% based on industry experience
Hours:	1,139 [(20,312 m^2/22.3 m^2/h)(1.25 contingency)]
Mobilization Costs	\$600
Labor Cost:	\$168,815
Equipment Costs:	\$3,480 per NUREG/CR-5884, Volume 2, Appendix C
Liquid Processing Costs:	\$34,276 [(\$1.00/g)(1.35g/m^2)(20,312 m^2)(1.25 liquid contingency)]
Waste Disposal Cost ($Cost_{WD}$):	\$101,500 [Solids estimated at 0.002 m^3/m^2 = 40.6 m^3 (\$ 2,500/m^3)]
Worker Accident Cost ($Cost_{ACC}$):	\$574 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	\$670 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	\$1,139 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.002 rem/crew-hour using Equation 4-5
Total Costs ($Cost_T$):	\$311,053
Cost ($Cost_T$) per m^2:	\$15.31

A.3 <u>Washing and Wiping Remediation Actions</u>

Area Evaluated For Unit Cost Determination:	20,312 m^2 (218,636 ft^2) per NUREG/CR-5884, Volume 1, Table 3.22
Crew Composition	2 Laborers, 0.5 HP Technician and 0.5 Crew Leader
Hourly Crew Cost:	\$98.57 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	2.8 m^2/h based on industry experience
Contamination Removed:	20% based on industry experience
Hours:	9,975 [(20,312 m^2/2.8 m^2//h) + 4 h/40 h set up)(1.25 contingency)]
Mobilization Costs	\$600

Labor Cost:	\$983,200
Equipment Costs:	\$21,571 based on industry experience
Waste Generation:	68.9 m^3 (3.39E-03 m^3/m^2) based on industry experience
Waste Disposal Cost ($Cost_{WD}$):	\$172,250 (\$ 2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	\$3,770 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	\$1,137 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	\$13,300 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.002 rem/crew-hour using Equation 4-5
Total Costs ($Cost_T$):	\$1,195,828
Cost ($Cost_T$) per m^2:	\$58.87

A.4.a Scabbling Remediation Action (Bounding Condition 0.635 cm (0.25 in) Concrete)*

Area Evaluated For Unit Cost Determination:	2,007 m^2 (21,598 ft^2) per NUREG/CR-5884, Volume 1, Table 3.22
Crew Composition	3 Laborers, 0.25 HP Technician and 0.25 Crew Leader per NUREG/CR-5884, Volume 2, Appendix C
Hourly Crew Cost:	\$102.02 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	4.65 m^2/h per NUREG/CR-5884, Volume 2, Appendix C removal rate is 9.29 m^2/h per pass with two passes required to remove 0.25 inches
Contamination Removed:	25% per NUREG/CR-5884, Volume 2, Appendix C removal of 12.5% per pass
Hours:	540 [(2,007 m^2/4.65 m^2//h)(1.25 contingency)]
Mobilization Costs	\$7,100 based on industry experience
Labor Cost:	\$55,041

Equipment Costs:	\$39,549.60 (\$73.24/hr) based on current industry experience*
Waste Generation:	12.7 m^3 = (2,007 m^2)(6.35E-3 m)
Waste Disposal Cost ($Cost_{WD}$):	\$31,750 (\$2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	\$238 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	\$210 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	\$617 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.003 rem/crew-hour using Equation 4-5
Total Costs ($Cost_T$):	\$134,505
Cost ($Cost_T$) per m^2:	\$67.02*

*Bounding condition includes cost for air compressor, consumables at 10% of the base equipment costs and the waste volume of 0.25 inch (0.635 cm) concrete depth.

A.4.b Scabbling Remediation Action (Bounding Condition 0.32 cm (0.125 in) Concrete)*

Area Evaluated For Unit Cost Determination:	2,007 m^2 (21,598 ft^2) per NUREG/CR-5884, Volume 1, Table 3.22
Crew Composition	3 Laborers, 0.25 HP Technician and 0.25 Crew Leader per NUREG/CR-5884, Volume 2, Appendix C
Hourly Crew Cost:	\$102.02 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	9.29 m^2/h per NUREG/CR-5884, Volume 2, Appendix C removal rate is 9.29 m^2/h per pass with one pass required to remove 0.125 inches
Contamination Removed:	12.5% per NUREG/CR-5884, Volume 2, Appendix C removal of 12.5% per pass
Hours:	270 [(2,007 m^2/9.29 m^2//h)(1.25 contingency)]
Mobilization Costs	\$7,100 based on industry experience
Labor Cost:	\$27,550

Equipment Cost:	\$15,827.40 (\$58.62/hr) based on current industry experience*
Waste Generation:	6.38 m^3 = (2,007 m^2)(3.18E-3 m)*
Waste Disposal Cost ($Cost_{WD}$):	\$15,950 (\$2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	\$119 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	\$105 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	\$309 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.003 rem/crew-hour using Equation 4-5
Total Costs ($Cost_T$):	\$66,961
Cost ($Cost_T$) per m^2:	\$33.36

*Bounding condition uses: (1) base equipment cost, (2) assumes an on-site air compressor, (3) no added consumables, and (4) the waste volume is relative to 0.125 inches (0.32 cm) depth of concrete, i.e., one-half of that assumed in A.4.a.

A.5 <u>Grit Blasting (Embedded/Buried Piping) Remediation Action</u>

Length Evaluated For Unit Cost Determination:	5,354 linear feet (LF) – total of embedded piping to remain at Rancho Seco
Crew Composition	2 Laborers, 1 Craft, 0.5 HP Technician and 0.5 Crew Leader based on NUREG/CR-5884, Volume 2, Appendix C crew for pressure washing
Hourly Crew Cost:	\$148.27 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	60 LF/h based on recent industry experience
Hours:	112 [(5,354 LF/60 LF/hr)(1.25 contingency multiplier)]
Mobilization Costs	\$4,000 based on recent industry experience
Labor Cost:	\$16,538
Equipment Costs:	\$123,311 based on recent industry experience

Waste Generation:	1.05 m^3 = (5,354 LF x 1.96E-04 m^3/LF at ~ 1.0 lb. per linear foot)
Waste Disposal Cost ($Cost_{WD}$):	\$2,625 (\$ 2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	\$56 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	\$17 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	\$112 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.002 rem/crew-hour for pressure washing using Equation 4-5
Total Costs ($Cost_T$):	\$146,659
Cost ($Cost_T$) per linear foot:	\$27.39

A.6.a Grit Blasting (Surfaces) Remediation Action (Bounding Condition 1.25 Contingency)

Area Evaluated For Unit Cost Determination:	2,007 m^2 (21,598 ft^2) per NUREG/CR-5884, Volume 1, Table 3.22 for scabbling evaluation
Crew Composition:	3 Laborers, 0.25 HP Technician and 0.25 Crew Leader as in the NUREG/CR-5884, Volume 2, Appendix C, evaluation for scabbling remediation action
Hourly Crew Cost:	\$102.02 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	2.79 m^2/hr based on recent industry experience
Hours:	899 [(2,007/2.79 m^2/h) x 1.25 contingency]
Mobilization Costs	\$7,339 [(2,007/2.79 m^2/h) x 0.10 set up x \$102.02/crew hour]
Labor Cost:	\$91,736
Equipment Costs:	\$51,315 based on recent industry experience
Grit/Consumables	\$17,984 based on recent industry experience
Waste Generation:	9.59 m^3 = (2,007 m^2 x 3.0E-03 m + 3.57 m^3 for grit)

Waste Disposal Cost ($Cost_{WD}$):	\$23,975 (\$2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	\$397 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	\$158 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	\$1,541 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.003 rem/crew-hour for scabbling using Equation 4-5
Total Costs ($Cost_T$):	\$194,445
Cost ($Cost_T$) per m^2	\$96.88

A.6.b <u>Grit Blasting (Surfaces) Remediation Action (Bounding Condition No Contingency)</u>

Area Evaluated For Unit Cost Determination:	2,007 m^2 (21,598 ft^2) per NUREG/CR-5884, Volume 1, Table 3.22 for scabbling evaluation
Crew Composition:	3 Laborers, 0.25 HP Technician and 0.25 Crew Leader as in the NUREG/CR-5884, Volume 2, Appendix C, evaluation for scabbling remediation action
Hourly Crew Cost:	\$102.02 per NUREG/CR-5884, Volume 2, Appendix C without escalation from 1993 labor costs
Cleaning Rate:	2.79 m^2/hr based on recent industry experience
Hours:	719 (2,007 m^2/2.79 m^2/hr)
Mobilization Costs	\$7,339 [(2,007/2.79 m^2/h) x 0.10 set up x \$102.02/crew hour]
Labor Cost:	\$73,389
Equipment Costs:	\$37,320 based on recent industry experience
Grit/Consumables	\$17,984 based on recent industry experience
Waste Generation:	9.59 m^3 = (2,007 m^2 x 3.0E-03 m + 3.57 m^3 for grit)
Waste Disposal Cost ($Cost_{WD}$):	\$23,975 (\$2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	\$317 per Equation 4-3

Transportation Accident Cost ($Cost_{TF}$):	$158 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	$1,233 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.003 rem/crew-hour for scabbling using Equation 4-5
Total Costs ($Cost_T$):	$161,715
Cost ($Cost_T$) per m^2	$80.58

A.7 Soil Excavation Remediation Action

Volume Evaluated For Unit Cost Determination:	1,500 m^3 (52,972 ft^3) based on top 15 cm of soil removed from a 10,000 m^2 area
Crew Composition	2 Laborers, 2 Craft, 0.25 HP Technician and 0.25 Crew Leader based on recent industry experience
Hourly Crew Cost:	$175.06 per NUREG/CR-5884, Volume 2, Table B.1 labor costs without escalation from 1993 labor costs
Cleaning Rate:	3.06 m^3/h based on recent industry experience
Hours:	980 [(1,500 m^3/3.06 m^3/h)(2.0 contingency multiplier for restaging and articulation)]
Mobilization Costs	$700 based on recent industry experience
Labor Cost:	$171,627
Equipment Costs:	$71,228 (consumables $9,291)
Waste Generation:	1,500 m^3 based on volume of soil removed
Waste Disposal Cost ($Cost_{WD}$):	$3,750,00 ($2,500/m^3)
Worker Accident Cost ($Cost_{ACC}$):	$556 per Equation 4-3
Transportation Accident Cost ($Cost_{TF}$):	$24,745 per Equation 4-4
Worker Dose Cost ($Cost_{WDose}$):	$871 at NUREG/CR-5884, Volume 2, Appendix C, dose rate of 0.002 rem/crew-hour for pressure washing using Equation 4-5
Total Costs ($Cost_T$):	$4,019,727

Cost ($Cost_T$) per m^3: **$2,679.82**

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5.0 FINAL STATUS SURVEY PLAN

5.1 Introduction

The Rancho Seco Nuclear Generating Station (Rancho Seco) Final Status Survey (FSS) Plan has been prepared using the applicable regulatory and industry guidance. This plan will be used to develop site procedures and work instructions to perform the FSS of the Rancho Seco site.

5.1.1 Purpose

The FSS Plan describes the final survey process used to demonstrate that the Rancho Seco facility and site comply with radiological criteria for unrestricted use specified in 10 CFR 20.1402, i.e., annual dose limit of 25 millirem plus ALARA for all dose pathways. Nuclear Regulatory Commission (NRC) regulations applicable to radiation surveys are found in 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b).

5.1.2 Scope

The Sacramento Municipal Utility District (District) intends to release site land from the 10 CFR Part 50 license using a phased approach. Phase I includes the majority of the site land and remaining structures (approximately 2,468 of the total 2,480 acres) scheduled for release after all demolition, remediation and FSS activities associated with plant operation are complete. Phase II of site release includes the approximately two acre Interim On-Site Storage Building (IOSB) following class B and C packaged radioactive waste removal, any required facility remediation and FSS. Once both these phases are complete the Rancho Seco site license under 10 CFR Part 50 will be terminated. An approximate 10 acre Independent Spent Fuel Storage Installation (ISFSI) located on the site is licensed under 10 CFR Part 72 and is not subject to the conditions of this License Termination Plan (LTP).

This FSS Plan addresses requirements applicable to Phase I of site release and may also be used during Phase II to release the IOSB following class B and C packaged radioactive waste removal. This Plan addresses only facilities and land areas that are identified as contaminated or potentially contaminated (impacted) resulting from activities associated with commercial nuclear plant operation.

5.1.3 Final Status Survey Preparation and Implementation Overview

The FSS Plan contained in this chapter will be used as the basis for developing FSS procedures and applying existing procedures to the FSS process. Section 5.1.4 contains a list of regulatory documents used as guidance in preparing the FSS Plan. Figure 5-1 provides an overview of the FSS process. Quality Assurance requirements are outlined in Section 5.8 and apply to activities associated with decommissioning and FSS.

An FSS Package will be produced for each survey area; this survey package is a collection of documentation detailing survey design, survey implementation and data evaluation for a final status survey of an area. The sections below describe specific elements of the FSS organization, preparation and implementation. All processes associated with final status surveys will be conducted in accordance with approved site procedures.

5.1.3.1 FSS Organization

The general FSS organization will consist of supervision, technical specialists, work planning coordinators, field coordinators, data analysts, and technicians. Since the FSS organization has not been fully implemented at the time of LTP development, it is expected that specific job titles may vary over the period of project execution. These titles are used within this document to describe various functional areas of responsibility and do not necessarily refer to specific job titles. Refer to Section 5.8.2.1 and RSAP-1901 which outline the responsibilities and functions of the FSS organization.

5.1.3.2 Survey Preparation

Survey preparation is the first step in the final status survey process and occurs after any necessary remediation is completed. In areas where remediation is required, a remediation survey or equivalent evaluation will be performed to confirm that remediation was successful prior to initiating FSS activities. Remediation surveys, turnover surveys, or equivalent evaluation, for areas not requiring remediation, may be performed using the same process and controls as a FSS so that data from these surveys may be used as part of the FSS data. In order for survey data to be used for FSS, it is intended that it should have been designed and collected in compliance with approved procedures and Sections 5.3 through 5.5 and the area controlled in accordance with approved procedures and Section 5.2.4 (Any surveys performed prior to the approval of the LTP are understood to have been performed "at risk". Survey design and the data collected would be carefully evaluated to ensure the intent of the LTP and associated procedures were met before using the data). Following turnover/remediation surveys or post-remediation evaluation, the FSS is performed. Areas to be surveyed are isolated and/or controlled to ensure that radioactive material is not reintroduced into the area from ongoing activities nearby and to maintain the "as left" condition of the area. Section 5.2 addresses specific survey preparation requirements and considerations.

Tools, equipment, and materials not needed to support survey activities are removed, unless authorized by the Dismantlement Superintendent (Radiological). Routine access, material storage, and worker transit through the area are not allowed, unless authorized by the FSS Supervisor. However, survey areas may, with proper approval, be used for staging of materials and equipment providing; 1) the staging does not interfere with performance of surveys, and 2) the material or equipment is free of surface contamination or radioactive materials, and 3) the safety of survey personnel is not jeopardized. An inspection of the area is conducted by FSS personnel to ensure that work is complete and the area is ready for final status survey. Approved procedures provide isolation and control measures until the area is released for unrestricted use.

5.1.3.3 Survey Design

The survey design process establishes the methods and performance criteria used to conduct the survey. Survey design assumptions are documented in "Survey Packages" in accordance with approved procedures. The site land, structures, and systems (embedded and buried piping/conduit are the principal potentially contaminated systems that will remain after decommissioning) are organized into survey areas and classified by contamination potential as Class 1, Class 2, Class 3, or non-impacted in accordance with Section 5.2.2.

Survey unit size is based on the assumptions in the dose assessment models in accordance with the guidance provided in NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning

Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report," [Reference 5-1]. The percent coverage for scan surveys is determined in accordance with Section 5.3.2. The number and location of structure surface measurements (and structure volumetric samples) and soil samples are established in accordance with Sections 5.3.3 and 5.3.5. Investigation levels are also established in accordance with Section 5.3.6.

A survey map is prepared for each survey unit and a reference grid is superimposed on the map to allow use of an (x, y) coordinate system. Random numbers between 0 and 1 are generated, which are then multiplied by the maximum x and y axis values of the sample grid. This provides coordinates for each random sample location, or a random start location for systematic grid, as appropriate. The measurement/sample locations are plotted on the map.

Each measurement/sample location is assigned a unique identification code, which identifies the measurement/sample by survey unit, and sequential number. The appropriate instruments and detectors, instrument operating modes and survey methods to be used to collect and analyze data are also specified.

Replicate measurements are performed as part of the quality process established to identify, assess, and control errors and uncertainty associated with sampling, survey, or analytical activities. This quality control process, described in Section 5.8.2 (FSS QAPP), provides assurance that the survey data meets the accuracy and reliability requirements necessary to support the decision to release or not release a survey unit.

Written survey instructions that incorporate the requirements set forth in the survey design are completed. Direction is provided, as applicable to survey design, for selection of instruments, count times, instrument modes, survey methods, required documentation, alarm/investigation setpoints, alarm actions, background requirements and other appropriate instructions. In conjunction with the survey instructions, survey data forms may be prepared to assist in survey documentation. Alternatively, electronic data recording systems may be utilized. The survey design is reviewed and quality verification steps applied to ensure that appropriate instruments, survey methods and sample locations have been properly identified.

5.1.3.4 Survey Data Collection

After preparation of a survey package, the FSS data are collected. Trained and qualified personnel will perform the necessary measurements using calibrated instruments in accordance with approved procedures and instructions contained in the survey package. Section 5.5 addresses FSS data collection requirements.

Survey areas and/or locations are identified by gridding, markings, or flags as appropriate. A FSS Field Coordinator performs a pre-survey briefing with the survey technicians during which the survey instructions are reviewed and additional survey unit considerations are discussed (e.g., safety). The technicians gather instruments and equipment as indicated and perform surveys in accordance with the appropriate procedures and survey package specifications. Technicians are responsible for documenting survey results and maintaining custody of samples and instrumentation. At the completion of surveys, technicians return instruments and prepare samples for analysis. Survey instruments provided to the technicians are prepared in accordance with approved site procedures and the survey instructions. Instrument calibration and performance checks are performed in accordance with applicable procedures. Data are reviewed to flag any measurements that exceed investigation criteria so that appropriate investigation surveys and remediation can be performed as necessary.

Following completion of a FSS, if a survey unit has been designated to receive a Quality Control (QC) survey (replicate surveys, sample recounts, etc.), a QC survey package is developed and implemented. QC measurement results are compared to the original measurement results. If QC results do not reach the same conclusion as the original survey, an investigation is performed. Section 5.8 provides additional detail regarding QC survey requirements.

5.1.3.5 Data Assessment

Survey data assessment is performed to verify that the data are sufficient to demonstrate that the survey unit meets the unrestricted use criterion. Statistical analyses are performed on the data and compared to pre-determined investigation levels (see Section 5.3.6). Depending on the results of the data assessment and any required investigation, the survey unit may either be released or require further remediation, reclassification, and/or resurvey. Assumptions and requirements in the survey package are reviewed for applicability and completeness; additional data needs are identified during this review. Specific data assessment requirements are contained in Section 5.6.

A review is performed of survey data and sample counting reports to verify completeness, legibility and compliance with survey design and associated instructions. As directed by FSS supervision, the following types of activities may be performed:

- Convert data to reporting units,
- Calculate mean, median and range of the data set,
- Review the data for outliers,
- Calculate the standard deviation of the data set,
- Calculate minimum detectable concentration (MDC) for each survey type performed, and
- Create posting, frequency or quantile plots for visual interpretation of data.

Computer programs may be utilized for these activities. FSS personnel include data quality verifications in their evaluations of statistical calculations; integrity and usefulness of the data set and the need for further data or investigation are also included in the evaluations. The results of the data evaluation are documented and filed in the survey package.

5.1.3.6 Final Status Survey Package Completion

Survey results are documented by survey unit in corresponding survey packages. Each FSS Package may contain the data from the several survey units that are contained in a given survey area. The data are reviewed, analyzed, and processed and the results documented in the FSS Package. This documentation file provides a record of the information necessary to support the decision to release the survey units for unrestricted use. An FSS Report will be prepared to provide the necessary data and analyses from survey packages for submittal to the NRC. Section 5.7 addresses reporting of survey results and conclusions.

5.1.4 Regulatory Requirements and Industry Guidance

This FSS Plan has been developed using the guidance contained in the following documents:

- NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," [Reference 5-2],
- NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," [Reference 5-3],
- NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," [Reference 5-4],
- NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," [Reference 5-5],
- NUREG–1757, Vol. 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report," and
- Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," (January 1999) [Reference 5-6].

Other documents used in the preparation of this plan are listed in the References section (see Section 5.9).

The District anticipates the NRC may choose to conduct confirmatory measurements during Rancho Seco FSS activities. The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the FSS and associated documentation demonstrate that the site is suitable for release in accordance with the criteria established in 10 CFR Part 20, subpart E.

5.2 Development of Survey Plan

5.2.1 Radiological Status

The following sections provide a summary of site characterization and dose modeling results applicable to development of the Rancho Seco FSS Plan.

5.2.1.1 Identification of Radiological Contaminants

A site-specific suite of radionuclides potentially present at Rancho Seco has been developed. This suite contains 26 radionuclides that are potentially present in Rancho Seco environs, structures and systems/components. Development of this site-specific suite of radionuclides is described in detail in the LTP Chapter 6, Compliance with the Radiological Criteria for License Termination, Section 6.4.

The District has conducted extensive radiological characterization of the site property to identify and document residual contamination resulting from nuclear plant operation. The effort included reviews of historical information as well as physical measurements of onsite soils, structures, systems and groundwater during scoping and characterization surveys. The LTP Chapter 2, Site Characterization, contains a detailed discussion of this effort.

5.2.1.2 Dose Modeling Summary

Dose models were based on NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning," [Reference 5-7] and RESRAD Version 6.22 and

RESRAD-BUILD Version 3.22 (RESRAD Version 6.3 and RESRAD-BUILD Version 3.3 for calculation revisions) were used to calculate single nuclide Derived Concentration Guideline Levels (DCGLs) for the Rancho Seco site. These dose models translate residual radioactivity levels into potential radiation doses to the public and are defined by three factors: (1) exposure scenario, (2) exposure pathways, and (3) exposed critical group. The scenarios presented in NUREG/CR-5512 address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. These scenarios also identify the critical group. The "critical group" is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of the particular land and structure use scenario.

As discussed in LTP Chapter 6, Section 6.5, the District has no plans to release any of the District-owned and District-controlled 2,480 acre site for ownership by members of the public. The site continues to be an important electrical generation and distribution center for the District. Therefore, an industrial worker scenario was selected for developing site-specific soil DCGLs and an industrial worker building occupancy scenario was selected for developing site-specific structural surface DCGLs. Table 5-1 provides a list of significant radionuclides that may be present in onsite soils and their corresponding single nuclide DCGL values derived in LTP Chapter 6. Table 5-2 provides a list of significant radionuclides that may be present on structural surfaces and their corresponding single nuclide DCGL values as derived in Chapter 6.

Table 5-1

Single Nuclide $DCGL_W$ Values for Detectable Radionuclides in Soil

Radionuclide	Peak of the Mean Dose (mrem/y per pCi/g)	$DCGL_W$ (pCi/g)
C-14	2.93E-06	8.33E+06
Co-60	1.93E+00	1.26E+01
Ni-63	1.60E-06	1.52E+07
Sr-90	3.76E-03	6.49E+03
Cs-134	1.09E+00	2.24E+01
Cs-137	4.62E-01	5.28E+01

Table 5-2

Calculated Structural Surface Single Nuclide DCFs and $DCGL_W$s

Radionuclide	Dose Conversion Factor (mrem/yr per dpm/100 cm^2)	$DCGL_W$ (dpm/100 cm^2)
H-3	7.94E-08	3.15E+08
C-14	2.92E-06	8.56E+06
Na-22	1.47E-03	1.70E+04
Fe-55	7.31E-07	3.42E+07
Ni-59	3.13E-07	7.99E+07
Co-60	1.64E-03	1.52E+04
Ni-63	8.20E-07	3.05E+07
Sr-90	2.07E-04	1.21E+05
Nb-94	1.09E-03	2.29E+04
Tc-99	2.13E-06	1.17E+07
Ag-108m	1.13E-03	2.21E+04
Sb-125	3.13E-04	7.99E+04
Cs-134	1.14E-03	2.19E+04
Cs-137	4.50E-04	5.56E+04
Pm-147	1.50E-06	1.67E+07
Eu-152	7.86E-04	3.18E+04
Eu-154	8.43E-04	2.97E+04
Eu-155	4.78E-05	5.23E+05
Np-237	1.05E-02	2.38E+03
Pu-238	7.30E-03	3.42E+03
Pu-239	8.19E-03	3.05E+03
Pu-240	8.19E-03	3.05E+03
Pu-241	1.37E-04	1.82E+05
Am-241	8.37E-03	2.99E+03
Pu-242	7.81E-03	3.20E+03
Cm-244	4.15E-03	6.02E+03

Other specialized DCGL values have also been developed in LTP Chapter 6. These include DCGL values for bulk material, containment building interior surfaces, buried piping and embedded piping. Their use is described in subsequent sections of this LTP Chapter.

5.2.1.3 Surrogate Ratio DCGLs

As a general rule, surrogate ratio DCGLs are developed and applied to land areas and materials with volumetric residual radioactivity where fairly constant radionuclide concentration ratios can be demonstrated to exist. They are derived using pre-remediation site characterization data collected prior to the FSS. The established ratio among the radionuclide concentrations allows the concentration of every radionuclide to be expressed in terms of any one of them.

Likewise, a surrogate ratio DCGL allows the DCGLs specific to hard-to-detect radionuclides in a mixture to be expressed in terms of a single radionuclide that is more readily measured. The measured radionuclide is called the surrogate radionuclide. Cs-137 is expected to be the surrogate radionuclide.

A sufficient number of measurements, representative of the area of interest, are taken to establish a consistent ratio of radionuclide concentrations. The number of measurements needed to determine the ratio is based on the chemical, physical and radiological characteristics of the radionuclides and the site. Measurements from different media types will not be mixed to derive the ratio. The surrogate ratio is acceptable if the mean values for individual samples for a given media are within two standard deviations of the overall mean value for the media.

Once an appropriate surrogate ratio is determined, the DCGL of the measured radionuclide is modified to account for the represented radionuclide according to the following Equation 5-1 (MARSSIM Equation 4-1):

$$DCGL_{SR} = DCGL_{Sur} \times \frac{DCGL_{\mathrm{Re}\,p}}{\left[\left(C_{\mathrm{Re}\,p}/C_{Sur}\right)\left(DCGL_{Sur}\right)\right] + DCGL_{\mathrm{Re}\,p}}$$

Equation 5-1

where:

$DCGL_{SR}$ = modified DCGL for surrogate ratio,

$DCGL_{Sur}$ = DCGL for surrogate radionuclide,

$DCGL_{Rep}$ = DCGL for represented radionuclide,

C_{Rep} = Concentration of represented radionuclide, and

C_{Sur} = Concentration of surrogate radionuclide.

When a surrogate ratio is established using data collected prior to remediation, post-remediation or FSS measurements will be reviewed to ensure that the established ratios are still appropriate. The surrogate ratio DCGL will be evaluated using the Rancho Seco DQOs and modified, if necessary. Professional judgment is used to determine consistency.

5.2.1.4 Gross Activity DCGLs

As a rule, gross activity DCGLs are developed and applied to structures and plant systems with surface residual radioactivity where multiple radionuclides are present at concentrations that exceed 10 percent of their respective DCGLs. The gross activity DCGL is determined in a manner similar to surrogate DCGLs taking into account nuclide detectability to enable field measurement of gross activity, rather than the determination of individual radionuclide activity, for comparison to the radionuclide specific DCGL. The gross activity DCGL, or $DCGL_{GA}$, for surfaces with multiple radionuclides is calculated using the following Equation 5-2 (MARSSIM, Equation 4-4):

$$DCGL_{GA} = \frac{1}{\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \dots \frac{f_n}{DCGL_n}}$$

Equation 5-2

where:

f_n = fraction of the total activity contributed by radionuclide n, and

$DCGL_n$ = DCGL for radionuclide n.

Different radionuclides or radionuclide combinations may exist on different portions of the site and require the calculation of one or more site-specific gross activity DCGLs. Gross activity DCGLs are calculated using the relative nuclide fractions determined from samples of building surface or plant system material, as appropriate, prior to remediation. For areas where the radionuclide distribution has not been determined, the most conservative distribution resulting in the lowest DCGL of those specified areas will be used. The distributions are based on the radionuclides identified in composite samples collected from the specific areas prior to FSS. If new radionuclide distribution data are obtained and determined to be more appropriate for use, the DCGL may be re-evaluated and altered during the course of the FSS, however the single nuclide DCGLs will not be revised without NRC approval.

5.2.2 Classification of Areas

Prior to beginning the final status survey, a thorough characterization of the radiological status and history of the site was performed. Additional data may be collected and evaluated throughout the decommissioning. The methods and results from site characterization are described in Chapter 2 of this LTP. Based on the characterization results, the structures and open land areas were classified following the guidance in Appendix A of NUREG-1757, Volume 2 and Section 4.4 of NUREG 1575. Area classification ensures that the number of measurements, and the scan coverage, are commensurate with the potential for residual contamination to exceed the unrestricted use criteria.

Initial classification of site areas is based on historical information and site scoping and characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance or any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the final status survey as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Once the FSS of a given survey unit begins, the basis for any reclassification will be documented, requiring a redesign of the survey unit package and the initiation of a new survey using the redesigned survey unit package. If during the conduct of a FSS, sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit, the FSS may be terminated without completing the survey unit package.

5.2.2.1 Non-Impacted Areas

Non-impacted areas have no reasonable potential for residual contamination because there was no demonstrable impact from site operations. These areas are not required to be surveyed beyond what has already been completed as a part of the HSA as described in the LTP Chapter 2, Section 2.2, or scoping or site characterization surveys performed to confirm the area's non-impacted classification. Rancho Seco will continue to implement a Radiological Environmental

Monitoring Program (REMP) through decommissioning and license termination. The REMP program is focused upon the collection of radiological data from offsite, non-impacted areas. Impacted areas are shown on Figure 2-2, Impacted Areas, in Chapter 2 of this LTP. The remaining areas of the 2,480 acre site listed in Table 5-3 are Non-Impacted.

5.2.2.2 Impacted Areas

Impacted areas may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. The definitions provided below are from NUREG-1757, Volume 2, Page A2.

- **Class 1 Areas:** Class 1 areas are impacted areas that are expected to have concentrations of residual radioactivity that exceed the $DCGL_W$ ($DCGL_W$ is defined in the Glossary of this LTP)[1],
- **Class 2 Areas:** Class 2 areas are impacted areas that are not likely to have concentrations of residual radioactivity that exceed the $DCGL_W$, and
- **Class 3 Areas:** Class 3 areas are impacted areas that have a low probability of containing residual radioactivity.

If the available information is not sufficient to designate an area as a particular class, the area will either be classified as Class 1 or be further characterized. Areas that are considered to be on the borderline between classes will receive the more restrictive classification.

5.2.2.3 Initial Classification of Structural Surfaces, Land, Embedded Piping, and Buried Piping

Based on more than 24,000 measurements made during the site characterization and the information evaluated as part of the HSA, all land areas, structural surfaces, and piping to remain after decommissioning were assigned an initial classification.

Characterization was performed and reported by initial survey unit designation. The area designations developed for the characterization process were used, for the most part, to delineate and classify areas for final status survey. This allows characterization data to be efficiently used for final survey area classification and for estimating the sigma value for sample size determination. For operational efficiency, each of the final survey areas listed in Table 5-3 may be subdivided into multiple areas. Smaller survey areas may be necessary to enhance the efficiency of data collection, processing, and review and serve to better support the decommissioning schedule. The classification of all subdivided survey areas will be the same as indicated in Tables 5-4A, 5-4B, 5-4C, 5-4D and 5-4E, unless reclassified in accordance with this LTP. No individual survey unit will have more than one classification.

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[1] The $_W$ In $DCGL_W$ refers to the Wilcoxon Rank Sum test per MARSSIM (NUREG-1575, page 2-3) and generally represents the uniform level of residual contamination that results in the dose limit, regardless of the statistical test used. Rancho Seco intends to use the Sign Test and will still use the term $DCGL_W$ to denote contamination limits, see Section 5.6.1.3.

Table 5-3
Area Designations

Area 1 (100000)	Non-Impacted*
Area 2 (200000)	Non-Impacted
Area 3 (300000)	Non-Impacted
Area 4 (400000)	Non-Impacted
Area 5 (500000)	Non-Impacted*
Area 6 (600000)	Non-Impacted
Area 7 (700000)	Non-Impacted
Area 8 (800000)	Impacted
Area 9 (900000)	Non-Impacted

*Areas 1 and 5 contain impacted survey units within them

Table 5-4A

Survey Unit Classification – General Open Land Areas

Survey Unit ID #	Survey Area	Sigma pCi/g, Cs-137	Classification	Mean Cs-137, pCi/g	Maximum Cs-137, pCi/g	Approx. Survey Area Size, m^2
100000	Plant Effluent Water Course	14.7	Class 2	9.22	42.2	42,315
100001,2	Area Around Effluent Path	0.14	Non-Impacted	0.349	0.483	N/A
200000	South Plant Outfall	0.15	Class 3	0.129	0.301	159,328
300000	South Non-Impacted Area	0.19	Non-Impacted	0.323	0.653	N/A
400000	South East Non-Impacted Area	0.17	Non-Impacted	0.344	0.465	N/A
500000	North East Non-Impacted Area (excluding parking lot and warehouse)	0.16	Non-Impacted	0.145	0.255	N/A
600000	North Non-Impacted Area	0.10	Non-Impacted	0.164	0.293	N/A
700000	West Non-Impacted Area (excluding ISFSI)	0.12	Non-Impacted	0.202	0.332	N/A

N/A – Not applicable

Table 5-4B
Survey Unit Classification – Site Surface Soils

Survey Unit ID #	Survey Area	Sigma pCi/g, Cs-137	Classification	Mean pCi/g, Cs-137	Maximum pCi/g, Cs-137	Approx. Survey Area Size, m^2
800100	Industrial Area Soils (4 quadrants)	0.042	Class 3	0.062	0.179	141,567
	Includes the following units as identified in the HSA:					
800001	Folsom Canal Intake	0.012	Class 3	0.073	0.091	4,645
800002	Helicopter Landing Pad	0.027	Class 3	0.065	0.127	4,506
800003	South Scrap Yard	0.103	Class 3	0.121	0.121	1,710
800004	Central N-S Transit/South	0.010	Class 3	0.028	0.048	200
800005	South E-W Transit Zone	0.042	Class 3	0.088	0.154	3,820
800006	Storm Drain Buffer Zone	0.047	Class 3	0.085	0.179	15,794
800007	West Industrial Area	0.034	Class 3	0.077	0.144	65,776
800008	South Support Structures Yard	0.041	Class 3	0.083	0.083	7,250
800009	South East Industrial Area	0.019	Class 3	0.080	0.099	10,765
800012	Industrial Area Waste Storage Buffer	0.010	Class 3	0.056	0.071	6,215
800013	Central Industrial Area	0.030	Class 3	0.070	0.149	14,676
800014	North Industrial Area Boundary	0.007	Class 3	0.060	0.068	6,410
	Following to be managed on unit specific basis					
501004	Extended Parking/Storage Area	0.058	Class 3	0.088	0.232	42,735
803002	Quonset-Hut Yard	0.030	Class 3	0.077	0.146	13,075
806000	East/West Spray Ponds	0.026	Class 3	0.069	0.167	21,600
810001	Tank Farm	10.7	Class 1	379.0	1040.0	5,000
826015	Spent Fuel/Diesel Gen room Gap	0.006	Class 2	0.041	0.047	25
834002	Rail Line	0.018	Class 3	0.073	0.114	6,410
837000	RHUT Area	9.83	Class 1	4.10	31.1	929
839000	Transformer Yard	0.432	Class 3	0.266	0.913	1,858
843002	Barrel Farm Buffer	1.460	Class 2	0.750	4.250	929
843003	Barrel Farm Berms	N/A	N/A	N/A	N/A	3,486
848000	Retention Basins Buffer	0.047	Class 3	0.086	0.200	17,615
851000	Switchgear Yard	0.008	Class 3	0.056	0.072	32,970

Table 5-4C
Survey Unit Classification – Paved Surfaces and Foundation Pads

Survey Unit ID #	Survey Area	Pavement/Pad		Mean Direct Pavement/Pad dpm/100 cm^2	Maximum Pavement/Pad dpm/100 cm^2	Approx. Survey Area Size, m^2
		Sigma dpm/100 cm^2	Classification*			
800200	Industrial Area Pavement	513	Class 3	2,630	5,262	73,231
	(Includes the following units as identified in the HSA):					
800001,2	Helicopter Pad Area	207	Class 3	2,713	3,158	7,573
800003	South Scrap Yard	255	Class 3	3,172	3,518	6,650
800004	Central N-S Transit/South	102	Class 3	1,613	1,719	5,350
800005	South E-W Transit/East	243	Class 3	841	1,344	5,470
800007	West Industrial Area	496	Class 3	2,977	5,262	5,759
800008	South Support Structures Yard	478	Class 3	2,397	3,517	18,820
		Incorporated in 800008				
818001	Electrical Fab Shop Pad	152	Class 3	1,392	1,647	233
820001	L&D Building Pad	163	Class 3	2,593	2,979	103
827001	Tool Room Pad	136	Class 3	1,245	1,593	605
828001	GRS Warehouse Pad	141	Class 3	1,337	1,892	446
838001	Fab Shop Pad	169	Class 3	1,413	1,772	1,025
842000	Warehouse C Pad	235	Class 3	2,207	2,468	235
800009	South East Industrial Area	440	Class 3	2,424	3,425	3,635
800010	IA Central Yard	745	Class 3	2,261	3,397	6,725
		Incorporated in 800010				
839000	Transformer Pads	944	Class 3	1,086	3,930	100
800012	IA Waste Storage Buffer	N/A	N/A	N/A	N/A	4,935
800013	Central IA Area	496	Class 3	2,593	4,403	6,689
800014	North IA Boundary	147	Class 3	2,696	3,055	1,625
Following to be managed on unit specific basis						
501003	Upper/Outer Yard	212	Class 3	2,419	2,805	2,375
501004	Extended Parking/Laydown	255	Class 3	2,522	3,251	28,150
501005	Access Road	200	Class 3	2,000	2,240	7,250
800011	IA Central E-W Corridor	1,256	Class 2	3,664	7,175	8,065

Table 5-4C

Survey Unit Classification – Paved Surfaces and Foundation Pads

Survey Unit ID #	Survey Area	Pavement/Pad		Mean Direct Pavement/Pad dpm/100 cm^2	Maximum Pavement/Pad dpm/100 cm^2	Approx. Survey Area Size, m^2
		Sigma dpm/100 cm^2	Classification*			
		Incorporated in 800011				
853001	NPS Fab Shop Pad	829	Class 3	395	2,093	929
803001	Q-Hut Pad	234	Class 3	3,446	3,821	150
803002	Q-Hut Yard	N/A	N/A	N/A	N/A	2,250
806000	Area Around Spray Ponds	207	Class 3	2,680	3,251	18,820
808003	Cooling Tower Basin Buffer	3,896	Class 2	2,894	14,025	37,996
809001	Sewer Plant	158	Class 3	1,892	2,169	227
819000	Bulk Waste Bld	7,356	Class 2	2,497	69,870	475
823000	Intake Pmp Structure	667	Class 3	260	1,375	25
824000	PCW Intake	397	Class 2	3,941	5,175	5
826025	N Laydown	250	Class 3	2,718	3,207	175
834000	Rail Line	1470	Class 1	3,653	8,567**	~100
836001	Aux Boiler Pad	1,109	Class 2	5,382	9,513	112
843002	Barrel Farm	3,637	Class 1	9,393	14,574	1,170
848000	Retention Basins	1,294	Class 1,2,3	11,361	13,950	3,750
851000	Switchgear Yard	397	Class 3	2,751	3,696	225

* Areas with more than one classification represent the range of classification throughout the area. Only a single class will apply to a survey unit.

**Contaminated asphalt at 43 pCi/g discovered.

N/A – Not applicable

Table 5-4D

Survey Area Characterization-Structures

Survey Unit/ Area ID #	Survey Area	Interior				Exterior				
		Sigma dpm/100 cm^2	Class	Mean Dir. Beta dpm/100 cm^2	Maximum Dir. Beta dpm/100 cm^2	Sigma dpm/ 100 cm^2	Class	Mean Dir. Beta dpm/100 cm^2	Maximum Dir. Beta dpm/100 cm^2	Approx. Survey Area Size, m^2
501001	Receiving Warehouse	397	3	1,734	2,386	65	3	1,212	1,364	1,860
501002	Hazmat Warehouse	533	3	1,913	2,642	147	3	1,468	1,826	1,420
804001	PAP Building	500	3	2,011	3,153	196	3	2,142	2,778	3,375
805001	Admin. Building	451	3	1,903	2,734	712	3	2,017	4,387	3,750
808001	E/W Cool Twr Basin	N/A	N/A	N/A	N/A	685	2	4,952	6,289	189
811000	Reactor Bldg. –27'	2,593,910	1	1,535,383	8,134,000	N/A	N/A	N/A	N/A	2,268
"	Reactor Bldg. Grade	238,479	1	201,670	370,000	N/A	N/A	N/A	N/A	780
"	Reactor Bldg. +40'	67,358	1	51,521	99,150	N/A	N/A	N/A	N/A	780
"	Reactor Bldg. +60'	22,086	1	20,110	46,660	N/A	N/A	N/A	N/A	2,100
"	Reactor Bldg. Ext/Roof	N/A	N/A	N/A	N/A	119	3	1,364	1,571	8,483
812000	Spent Fuel Bldg pool	56,500,000	1	16,900,000	200,000,000	N/A	N/A	N/A	N/A	195
"	Spent Fuel Bldg. +40'	4,631	2	5,942	19,358	N/A	N/A	N/A	N/A	1,078
"	Spent Fuel Bld Exterior	N/A	N/A	N/A	N/A	747	2	1,935	4,996	~660
"	Spent Fuel Bldg. roof	N/A	N/A	N/A	N/A	408	3	1,729	2,229	480
813000	Auxiliary Bldg. –47'	740,452	1	320,071	5,720,000	N/A	N/A	N/A	N/A	1,639
"	Auxiliary Bldg. –29'	1,200,000	1	544,756	11,370,277	N/A	N/A	N/A	N/A	4,391
"	Auxiliary Bldg. –20'	920,181	1,2	247,831	10,080,000	N/A	N/A	N/A	N/A	8,518
"	Auxiliary Bldg. Grade	1,046,734	1,2	373,758	5,800,000	N/A	N/A	N/A	N/A	2,927
"	Auxiliary Bldg. +20'	309,414	1,2	85,408	1,900,000	N/A	N/A	N/A	N/A	2,162
"	Auxiliary Bldg. +40'	3,627	1,2	3,288	24,781	N/A	N/A	N/A	N/A	1,572
"	Auxiliary Bldg. roof	N/A	N/A	N/A	N/A	136	3	1,984	2,250	1,955
"	Auxiliary Bld exterior	N/A	N/A	N/A	N/A	342	3	1,897	2,990	~950
814000	T&R Bldg.	500	3	1,680	2,528	435	3	1,865	2,995	11,400
815000	Nucl Svc Elect Bldg.	353	3	1,636	2,131	261	3	1,913	2,669	4,650
816000	Cent Alarm Sta Bldg	364	3	2,066	3,327	234	3	2,327	2,789	1,210

Table 5-4D
Survey Area Characterization-Structures

Survey Unit/ Area ID #	Survey Area	Interior				Exterior				
		Sigma dpm/100 cm^2	Class	Mean Dir. Beta dpm/100 cm^2	Maximum Dir. Beta dpm/100 cm^2	Sigma dpm/ 100 cm^2	Class	Mean Dir. Beta dpm/100 cm^2	Maximum Dir. Beta dpm/100 cm^2	Approx. Survey Area Size, m^2
817000	TDI Diesel Gen Bldg	647	3	2,343	4,066	141	3	1,859	2,256	4,500
821000	Water Treat Bld	266	3	2,343	2,897	511	3	2,968	3,816	826
822000	Chlorine Bld	438	3	828	1,723	1,032	3	1,517	3,832	1,055
826000	Turbine Bldg-7'	5,990	1,2,3	3,077	24,900	N/A	N/A	N/A	N/A	974
"	Turbine Bldg Grade	1,316	1,2,3	2,305	6,980	N/A	N/A	N/A	N/A	4,190
"	Turbine Bldg Mezz	402	3	1,566	2,626	N/A	N/A	N/A	N/A	2,605
"	Turbine Bld +40'& Ext	277	3	2,843	3,615	1,723	3	1,984	10,312	144
831000	Microwave Bld	1,639	3	1,568	6,344	5,210	2	2,875	13,253	350
833000	Warehouse B	807	3	635	3,751	10,064	2	3,749	34,785	6,345
840000	Warehouse A	495	3	1,941	3,397	511	3	2,310	3,838	6,875
848000	Retention Basin	601	1	45,485	45,910	N/A	N/A	N/A	N/A	21,368
850000	Solidification Pad/Wall	N/A	N/A	N/A	N/A	57,832	1	22,653	322,600	360
851000	Switchyard Control Bld	342	3	1,663	2,376	196	3	1,397	1,843	2,975
852000	Machine Shop	288	3	1,973	2,408	217	3	2,087	2,620	2,900
856000	Sec Alarm Station	1,205	3	300	2,636	4,317	3	293	9230	350

Table 5-4E
Survey Area Characterization – Remaining Buried and Embedded Pipe

Survey Unit/Area ID #	Description	Sigma dpm/100 cm^2	Mean Direct dpm/100 cm^2	Maximum Direct dpm/100 cm^2	Classification	Internal Surface Area m^2
899002	Aux Feedwater Piping	207	368	634	Class 3	<29
899005	Clean Drain System, Storm Drain Non-Discharge	392	196	380	Class 3	770
899006	Component Cooling Water System Piping	4,174	780	10,482	Class 3	15
899007	Clean Drain System Piping - Turbine	93,519	56,208	680,000	Class 1,2	310
899008	Clean Drain System Piping - Sewer	4	6	10	NI	N/A
899009	Clean Drain System Piping – Storm Drain/Liquid Disch	431	2,590	3,158	1*	809
899010	Diesel Fuel Oil System Piping	4,984	-8,894	186	NI	66
899011	Decay Heat System Piping	630,324	480,508	3,412,000	Class 1	17
899017	Fire Protection Water System Piping	815	35	1,154	NI	N/A
899025	Instrument Air System Piping	3,793	-5,367	1,520	NI	N/A
899028	Main Condenser Makeup	1,089	594	3,542	Class 3	34
899029	Main Circulating Water System Piping	99	475	614	Class 3	1,296
899032	Nitrogen Gas System Piping	9,677	19,100	33,200	Class 2	8
899034	Nuclear Service Raw Water System Piping	157	28	413	NI	N/A
899035	Nuclear Service Water Piping	125	-59	174	Class 3	37
899036	Plant Cooling Water System Piping	153	-9	310	NI	N/A
899040	Reactor Coolant Drain System	TBD	TBD	TBD	Class 1	2
899042	Radwaste System Piping	66,200,000	49,700,000	211,000,000	Class 1	53
899043	Service Air System Piping	964	-5	1,740	Class 3	287
899044	Spent Fuel Cooling System Piping	4,730,000	5,190,000	16,500,000	Class 1	11
899045	Site Reservoir System	232	6	441	NI	N/A
899047	Service Water System Piping	1,992	187	2,700	Class 3	1000
899050	Waste Gas System Piping	2,977	528	3,665	Class 3	<9
899051	Carbon Dioxide System	8,930	8,585	23,654	Class 2	4
899052	Acid Waste System	13,000	2,450,000	74,600,000	Class 1	26

NI – Non-Impacted

*Pipe sediment activity 186 pCi/g Cs-137, 23.5 pCi/g Co-60

5.2.2.4 Changes in Classification

Initial classification of site areas is based on historical information, scoping surveys and site characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance and any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the FSS as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Areas within initial survey units may be upgraded in classification due to future requirements for laydown and storage areas during demolition activities or incorrect initial classification. If during the conduct of a FSS sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit in accordance with Section 5.3.6, the survey may be terminated without completing the survey unit package.

5.2.3 Establishing Survey Units

The survey units listed in Tables 5-4A, 5-4B, 5-4C, 5-4D and 5-4E are areas that have similar characteristics and contamination levels. Survey units are assigned only one classification. The site and facility are surveyed, evaluated, and released on a survey unit basis.

5.2.3.1 Survey Unit Size

Survey unit sizes will be selected based on area classification, survey execution logistics, and applicable regulatory guidance documents. NUREG-1757, Volume 2, Appendix A, provides suggested sizes for survey units based on the guidance contained in MARSSIM.

Typical survey unit sizes for structural surfaces and open land area soil are listed below in Table 5-5; these are consistent with NUREG-1757, Volume 2 guidance. Use of 319 m^2 for structures meets the 25 mrem/y criterion at the surface DCGL as described in DTBD 06-002, "Use of a Survey Unit Size of 319 m^2 for Class One Structure Surveys at Rancho Seco Nuclear Generating Station," [Reference 5-8]. Class 1 and 2 areas provided in Tables 5-4A, 5-4B, 5-4C, 5-4D and 5-4E may be further subdivided into smaller areas to meet the guidelines present in Table 5-5. If larger survey unit areas are used, a technical evaluation will be presented in the FSS Package for the specific survey unit justifying the survey unit size.

Table 5-5
Suggested Survey Unit Areas

Class	Structural Surfaces	Open Land Area Soil
1	up to 319 m^2	up to 2,000 m^2
2	319 to 1000 m^2	2,000 to 10,000 m^2
3	no limit	no limit

5.2.3.2 Reference Coordinate System for Open Land Areas (Reference Grid)

A reference coordinate system is used for impacted areas to facilitate the identification of survey units within the survey area. The reference coordinate system is basically an X-Y plot of the site area referenced to the state of California Mercator projections as shown in Figure 5-2. Once the reference point is established, grids may be overlaid parallel to lines of latitude and longitude.

5.2.4 Access Control Measures

5.2.4.1 Turnover

Due to the large scope of decommissioning activities, it is anticipated that some surveys will be performed in parallel with dismantlement activities. This will require a systematic approach to turnover of areas be established. Prior to acceptance of a survey unit for FSS, the following conditions must be satisfied in accordance with applicable procedures. These include:

a) Decommissioning activities having the potential to contaminate a survey unit shall be complete or measures taken to eliminate such potential.

b) Tools and equipment not required for the survey must be removed, and housekeeping and cleanup shall be complete.

c) Decontamination activities in the area shall be complete.

d) Access control or other measures to prevent recontamination must be implemented.

e) Turnover or remediation surveys may be performed and documented to the same standards as final status surveys so that data can be used for the FSS.

5.2.4.2 Walkdown

The principal objective of the walkdown is to assess the physical scope of the survey unit. The walkdown ensures that the area has been left in the necessary configuration for FSS or that any further work has been identified. The walkdown provides detailed physical information for survey design. Details such as structural interferences or sources needing special survey techniques can be determined. Specific requirements will be identified for accessing the survey area and obtaining support functions necessary to conduct final status surveys, such as excavation shoring, interference removal, dewatering, etc. Industrial safety and environmental concerns will also be identified during this walkdown.

5.2.4.3 Transfer of Control

Once a walkdown has been performed and the turnover requirements have been met, control of access to the area is transferred from the RP/Chemistry Department to the FSS group. Access control and isolation methods are described in the subsection below.

5.2.4.4 Isolation and Control Measures

Since all decommissioning activities will not be completed prior to the start of the FSS, measures will be implemented to protect survey areas from contamination during and subsequent to the FSS. Decommissioning activities creating a potential for the spread of contamination will be completed within each survey unit prior to the FSS. Additionally, decommissioning activities that create a potential for the spread of contamination to adjacent areas will be evaluated and controlled. Upon commencement of the FSS for survey units where there is a potential for re-contamination, implementation of one or more of the following control measures will be required:

- Personnel training,
- Installation of barriers to control access to surveyed areas,
- Installation of barriers to prevent the migration of contamination from adjacent or overhead areas from water runoff, etc.,
- Installation of postings requiring contamination monitoring prior to surveyed area access,
- Locking entrances to surveyed areas of the facility,
- Installation of tamper-evident devices at entrance points, or
- Routine surveys to monitor and verify adequacy of isolation and control measures.

Routine surveys will not be required for open land areas that are not normally occupied and are unlikely to be impacted by decommissioning activities. Post-FSS survey locations will be judgmentally selected for survey, based on technical or site-specific knowledge and current conditions present in or near the survey area. These surveys are primarily designed to detect the potential migration of contaminants from decommissioning activities taking place in adjacent areas.

5.3 Survey Design and Data Quality Objectives

This section describes the methods and data required to determine the number and location of measurements or samples in each survey unit and the coverage fraction for scan surveys. The design activities described in this section will be documented in a survey package for each survey unit. Survey design includes the following:

- Type I and II Errors,
- Scan Survey Coverage,
- Sample Size Determination,
- Instrumentation and Required MDCs,
- Reference Grid and Sample Location, and
- DCGL and DCGL_{EMC}.

5.3.1 Data Quality Objectives (DQOs)

The appropriate design for a given survey area is developed using the DQO process as outlined in MARSSIM, Appendix D. These seven steps are:

1) State the problem,
2) Identify the decision,
3) Identify inputs to the decision,
4) Define the study boundaries,
5) Develop a decision rule,

6) Specify limits on decision errors, and
7) Optimize the design for obtaining data.

The DQO process will be used for designing and conducting all final status surveys at Rancho Seco. Each survey package will contain the appropriate information, statistical parameters and contingencies to support the DQO process.

5.3.2 Scan Survey Coverage

The area covered by scan measurement is based on the survey unit classification as described in NUREG 1757 and as shown in Table 5-6 below. A 100% accessible area scan of Class 1 survey units will be required. The emphasis will be placed on scanning the higher risk areas of Class 2 survey units such as soils, floors and lower walls. Scanning percentage of Class 3 survey units will be performed on likely areas of contamination based on the judgment of the FSS engineer.

Table 5-6
Scan Measurements

	Class 1	Class 2*	Class 3
Scan Coverage	100%	10 – 100%	Judgmental, 1-10%

* For Class 2 Survey Units, the amount of scan coverage will be proportional to the potential for finding areas of elevated activity or areas close to the release criterion in accordance with MARSSIM Section 5.5.3. Accordingly, Rancho Seco will use the results of individual measurements collected during characterization to correlate this activity potential to scan coverage levels.

5.3.3 Sample Size Determination

NUREG-1727, Volume 2, Appendix A describes the process for determining the number of survey measurements necessary to ensure a data set sufficient for statistical analysis. The sample density for Class 1 structures will be maintained at the level of 1/7 m^2 for the larger survey unit size of 319 m^2 which results in a total of 46 samples for a relative shift of 3. Sample size is based on the relative shift, the Type I and II errors, sigma, and the specific statistical test used to evaluate the data.

Alternate processes may be used if such gain NRC and industry acceptance between the time this plan is adopted and the commencement of FSS activities. However, any new technologies must still meet the applicable requirements of this plan for calibration, detection limit, areal coverage, operator qualification, etc.

5.3.3.1 Determining Which Test Will Be Used

Appropriate tests will be used for the statistical evaluation of survey data. Tests such as the Sign test and Wilcoxon Rank Sum (WRS) test will be implemented using unity rules, surrogate methodologies, or combinations of unity rules and surrogate methodologies, as described in MARSSIM and NUREG-1505 chapters 11 and 12.

If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign test will be used. If background is a significant fraction of the DCGL, the Wilcoxon Rank Sum (WRS) test will be used.

5.3.3.2 Establishing Decision Errors

The probability of making decision errors is controlled by hypothesis testing. The survey results will be used to select between one condition of the environment (the null hypothesis) and an alternate condition (the alternative hypothesis). These hypotheses, chosen for MARSSIM Scenario A, are defined as follows:

Null Hypothesis (H_0): The survey unit does not meet the release criteria. Alternate Hypothesis (H_a): The survey unit does meet the release criteria.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criteria. It occurs when the null hypothesis is rejected when it is true. The probability of making this error is designated as "α". A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criteria. This occurs when the Null Hypothesis is accepted when it is not true. The probability of making this error is designated as "β".

Appendix E of NUREG-1757, Volume 2 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the NUREG-1757, Volume 2 guidance, α will be set at 0.05. A β of 0.05 will initially be selected based on site specific considerations. The β may be modified, as necessary, after weighing the resulting change in the number of required survey measurements against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criteria.

5.3.3.3 Relative Shift

The relative shift (Δ / σ) is calculated. Delta (Δ) is equal to the $DCGL_W$ minus the Lower Boundary of the Gray Region (LBGR). Calculation of sigma's is discussed in Section 5.3.3.3.2 and initial values are provided in Table 5-4. The sigma's used for the relative shift calculation may be recalculated based on the most current data obtained from post-remediation or post-demolition surveys or from background reference areas, as appropriate. The LBGR is initially set at 0.5 times the $DCGL_W$, but may be adjusted to obtain an optimal value, of normally between 1 and 3 for the relative shift.

5.3.3.3.1 Lower Boundary of the Gray Region

The Lower Boundary of the Gray Region (LBGR) is the point at which the Type II (β) error applies. The default value of the LBGR is set initially at 0.5 times the DCGL. If the relative shift is greater than 3, then the number of data points, N, listed for the relative shift values of 3 from Table 5-5 or Table 5-3 in MARSSIM will normally be used as the minimum sample size. If the minimum sample size results in a sample density less than the required minimum density (see Section 5.2.3), the sample size will be increased accordingly.

5.3.3.3.2 Sigma

Sigma values (estimate of the standard deviation of the measured values in a survey unit, and/or reference area) were initially calculated from characterization data as listed in DTBD-06-001, "RSNGS Initial Classification of Survey Areas and Survey Design Sigma Values," [Reference 5-9]. These sigma values can be used in FSS design or more current post-remediation sigma values can be used. The use of the sigma values from the characterization data will be conservative for the sample size determination since the post-remediation sigma's are expected to be smaller. The sigma values for survey areas listed in Tables 5-4A, 5-4B, 5-4C, 5-4D and 5-4E which contain survey units with two different classifications (typically upper walls and ceiling being a Class lower than lower walls and floor of the same room), will be evaluated to ensure that the sigma conservatively represents the contaminant distribution of each associated survey unit; otherwise a specific sigma value will be developed.

5.3.3.3.3 Wilcoxon Rank Sum Test Sample Size

The number of data points, N, to be obtained from each reference area or survey unit are determined using Table 5-3 in MARSSIM. The table includes the recommended 20% adjustment to ensure an adequate sample size.

5.3.3.3.4 Sign Test Sample Size

The number of data points is determined from Table 5-5 in MARSSIM for application of the Sign Test. This table includes the recommended 20% adjustment to ensure an adequate sample size.

5.3.3.3.5 Elevated Measurement Comparison Sample Size Adjustment

If the scan MDC is greater than the $DCGL_W$, the sample size will be calculated using Equation 5-3 provided below. If N_{EMC} exceeds the statistically determined sample size (N), N_{EMC} will replace N.

$$N_{EMC} = A/A_{EMC}$$

Equation 5-3

where:

N_{EMC} = the elevated measurement comparison sample size,

A = the survey unit area, and

A_{EMC} = the area corresponding to the area factor calculated using the MDC_{scan} concentration.

5.3.4 Background Reference Area

Background reference area measurements are required when the WRS test is used, and background subtraction may be used with the Sign test, under certain conditions such as those described in Chapter 12 of NUREG-1505. Reference area measurements, if needed, will be collected using the methods and procedures required for Class 3 final survey units. For soil, reference areas will have a soil type as similar to the soil type in the survey unit as possible. When there is a reasonable choice of possible soil reference areas with similar soil types, consideration will be given to selecting reference areas that are most similar in terms of other physical, chemical, geological, and biological characteristics. For structure survey units that contain a variety of materials with markedly different backgrounds, a reference area will be selected that has similar materials. If one material is predominant or if there is not too great a variation in background among materials, a background from a reference area containing only a single material is appropriate when it is demonstrated that the selected reference area will not result in underestimating the residual radioactivity in the survey unit.

It is understood that background reference areas should have physical characteristics (including soil type and rock formation) similar to the site and shall not be contaminated by site activities. Non-impacted areas of the 2,480 acre site may be chosen to serve as background reference areas.

Should significant variations in background reference areas be encountered, appropriate evaluations will be performed to define the background concentration. As noted in NUREG-1757, Appendix A, Section A.3.4, the Kruskal-Wallis test can be conducted in such circumstances to determine that there are no significant differences in the mean background concentrations among potential reference areas. Rancho Seco will consider this and other statistical guidance in the evaluation of apparent significant variations in background reference areas.

If material background subtraction is performed, the sigma value used will take into account the variability of material background.

5.3.5 Reference Grid and Sample Location

Sample location is a function of the number of measurements required, the survey unit classification, and the contaminant variability.

5.3.5.1 Reference Grid

The reference grid is primarily used for reference purposes and is illustrated on sample maps. Physical marking of the reference grid lines in the survey unit will only be performed when necessary. For the sample grid in Class 1 and 2 survey units, a randomly selected sample start point will be identified and sample locations will be laid out in a square grid pattern at distance, L, from the start point in both the horizontal and vertical directions. The sample and reference grids are illustrated on sample maps and may be physically marked in the field. For Class 3 survey units, all sample locations are randomly selected, based on the reference grid. An example is shown in Figure 5-2. Global Positioning System (GPS) instruments may be used in open land areas to determine reference or sample grid locations within the survey area. Locations within a survey area may also be tied to a site USGS survey benchmark (The site drawings, including Figure 5-2, are based on the California Coordinate System 1927 which locates the center of the reactor building at 2,249,270 east and 242,040 north within Zone 2). Digital cameras may be employed to provide a record of survey location within the survey unit. When used, these photographic records will be linked to landmark and directional information to ensure reproducibility.

5.3.5.2 Measurement Locations

Measurement locations within the survey unit are clearly identified and documented for purposes of reproducibility. Actual measurement locations are identified by tags, labels, flags, stakes, paint marks, geopositioning units or photographic record. An identification code matches a survey location to a particular survey unit.

Sample points for Class 1 and Class 2 survey units are positioned in a systematic pattern or grid throughout the survey unit by first randomly selecting a start point coordinate. A random number generator is used to determine the start point of the square grid pattern. The grid spacing, L, is a function of the area of the survey unit as shown in Equation 5-4 below for a square grid:

$$L = \sqrt{\frac{A}{n}}$$

Equation 5-4

where:

A = the area of the survey unit, and

n = the number of sample points in the survey unit.

Sample points are located, L distance from the random start point in both the X and Y directions.

Random measurement patterns are used for Class 3 survey units. Sample location coordinates are randomly picked using a random number generator.

Measurement locations selected using either a random selection process or a randomly-started systematic pattern that do not fall within the survey unit or that cannot be surveyed due to site conditions are replaced with other measurement locations as determined by the FSS Engineer.

5.3.6 Investigation Levels and Elevated Areas Test

During survey unit measurements, levels of radioactivity may be identified that warrant investigation. Depending on the results of the investigation, the survey unit may require no action, remediation, and/or reclassification and resurvey. Investigation process and investigation levels are described below.

5.3.6.1 Investigation Process

During the survey process, locations with potential residual activity exceeding investigation levels are marked for further investigation. The elevated survey measurement is verified by resurvey. For Class 1 areas, size and average activity level in the elevated area is acceptable if it complies with the area factors and other criteria that may apply to evaluation of the DCGL for elevated measurements $\mathrm{DCGL_{EMC}}$. As discussed in Section 5.3.6.3 below, the $\mathrm{DCGL_{EMC}}$ is applicable only for Class 1 areas. If any location in a Class 2 area exceeds the DCGL, scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s) and the area evaluated for reclassification. If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area evaluated for reclassification and resurvey under the criteria of the new classification. All survey unit investigations will be conducted in accordance with the applicable FSS Data Quality Objectives (DQOs).

Investigations should address: (1) the assumptions made in the survey unit classification; (2) the most likely or known cause of the contamination; and (3) the effects of summing multiple areas with elevated activity within the survey unit. Depending on the results of the investigation, a portion of the survey unit may be reclassified or combined with an adjacent area with similar characteristics if there is sufficient justification. Either action would result in resurvey of the (new) area(s). The results of the investigation process are documented in the Survey Package. See also Section 5.6 for additional discussion regarding potential reclassification of the survey unit.

5.3.6.2 Investigation Levels

Technicians will respond to all instrument alarms while surveying. Upon receiving an alarm, the technician will stop and resurvey the last square meter of area to verify the alarm. Technicians are cautioned, in training, about the importance of the alarm verification survey and are given specific direction in the procedure as to survey extent and scan speed. If the alarm is verified, the technician will mark the area with a flag or other appropriate means. Each area marked will be addressed in an investigation survey instruction prepared for the survey unit. The instruction will specify the required actions, such as a re-scan of the area, direct measurements, field gamma spectroscopy measurement (as appropriate), and collection of a soil sample (for land surveys). Each investigation will be evaluated and reported in the survey unit Release Record. Investigation levels are shown in Table 5-7.

Table 5-7
Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	$>DCGL_{EMC}$	$>DCGL_{EMC}$
Class 2	$>DCGL_W$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_W$	$>DCGL_W$
Class 3	$>DCGL_W$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_W$	$>0.5\ DCGL_W$

The size and average activity level in the elevated area is determined to demonstrate compliance with the area factors. If any location in a Class 2 area exceeds the DCGL, scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s). If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area should be considered for reclassification.

5.3.6.3 Elevated Measurement Comparison

5.3.6.3.1 Open Land Areas and Structural Surfaces

The elevated measurement comparison is applied to Class 1 survey units when one or more verified scan or static measurement exceeds the investigation level. As stated in MARSSIM, the EMC is intended to flag potential failures in the remediation process and should not be considered the primary means to identify whether or not a survey unit meets the release criterion. The EMC provides assurance that unusually large measurements receive the proper attention and that any area having the potential for significant dose contribution is identified. Locations identified by scan methodology or soil sample analyses measurements with levels of residual radioactivity which exceed the $DCGL_{EMC}$ are subject to additional surveys to determine compliance with the elevated measurement criteria. The size of the area containing the elevated residual radioactivity and the average level of residual activity within the area are determined. The average level of activity is compared to the $DCGL_W$ based on the <u>actual</u> area of elevated activity. An *a priori* DCGLEMC for the area between direct measurements (the likely size of an elevated area) is established during the survey design and is calculated as follows:

$$DCGL_{EMC} = Area\ Factor \times DCGL_W$$

Equation 5-5

The area factor is the multiple of the $DCGL_W$ that is permitted in the area of elevated residual radioactivity without remediation. The area factor is related to the size of the area over which the elevated activity is distributed. The actual area is generally bordered by levels of residual radioactivity below the $DCGL_W$ and its size is determined during the investigation process. Area factor calculations are described in LTP Section 6.7 and summarized in Tables 5-8 and 5-9. (As shown in Tables 5-8 and 5-9, Co-60 and Cs-137 are the limiting Area Factors considering that direct exposure is the primary dose concern for structures and soil. Therefore, these area factors will typically be used to evaluate elevated measurements in soil or on surfaces). Alternatively, Figures 6-7 through 6-9 in Chapter 6 of this LTP provide a graphical method for selecting applicable area factors. The actual area of elevated activity is determined by investigation surveys and the area factor is adjusted for the actual area of elevated activity. The product of the adjusted area factor and the $DCGL_W$ determines the $DCGL_{EMC}$. Additional measurements are made to determine the average activity of the elevated area, if necessary. If the $DCGL_{EMC}$ is exceeded, the area is remediated and resurveyed. The results of the elevated area investigations in a given survey unit that are below the $DCGL_{EMC}$ limit are evaluated using Equation 5-6 below. If more than one elevated area is identified in a given survey unit, the unity rule with Equation 5-6 is used to determine compliance. If the formula value is less than unity, no further elevated area testing is required and the EMC test is satisfied.

Table 5-8

Calculated Surface Soil Area Factors

Contaminated Area (m^2)	**Radionuclide Area Factor (unitless)**					
	C-14	**Co-60**	**Ni-63**	**Sr-90**	**Cs-134**	**Cs-137**
10,000	1.00	1.00	1.00	1.00	1.00	1.00
3,000	1.66	1.02	1.00	1.02	1.02	1.02
1,000	2.52	1.04	1.00	1.04	1.04	1.04
300	4.80	1.12	3.27	1.20	1.12	1.11
100	8.04	1.24	9.30	1.36	1.23	1.23
30	13.6	1.62	26.9	1.77	1.58	1.58
10	21.8	2.39	60.6	2.61	2.31	2.31
3	41.5	5.05	114	5.51	4.89	4.87
1	76.0	11.8	164	12.8	11.3	11.3

Table 5-9

Calculated Structural Surface Area Factor Values

Contaminated Area (m^2)	Radionuclide Area Factor (unitless)					
	Co-60	Cs-134	Cs-137	Pu-238	Pu-239	Pu-240
137	1	1	1	1	1	1
68	1.22	1.24	1.26	—	—	—
36	1.51	1.55	1.59	—	—	—
25	1.74	1.79	1.85	—	—	—
16	2.11	2.18	2.26	—	—	—
9	2.83	2.91	3.03	—	—	—
4	4.65	4.80	5.02	—	—	—
1	13.8	14.3	14.9	—	—	—
0.5	25.9	26.8	28.1	273	272	272
Contaminated Area (m^2)	**Radionuclide Area Factor (unitless)**					
	Pu-241	**Am-241**				
137	1	1				
68	—	—				
36	—	—				
25	—	—				
16	—	—				
9	—	—				
4	—	—				
1	—	—				
0.5	274	257				

Equation 5-5 applies to a single radionuclide contaminant. When multiple radionuclides are present, the calculation in Equation 5-5 is made with a unitized DCGL.

$$\frac{\delta}{DCGL_W} + \frac{(Conc_{AVE} - \delta)}{(Area\ Factor)(DCGL_W)} < 1$$

Equation 5-6

where:

δ = Estimate of average concentration of residual radioactivity and

$Conc_{AVE}$ = average concentration in elevated area.

If more than one elevated area exists in the survey unit, a separate term will be included for each in Equation 5-6 (refer to Section 5.6.2.2).

5.3.6.3.2 Embedded Piping

The dose model for embedded pipe used a default pipe length of 3 m. Area factors will vary with the internal dimensions of the pipe being surveyed. If it becomes necessary to apply the elevated measurement comparison process to embedded piping, the first step will be to determine the length of pipe in the room or area being surveyed. Then the interior surface area will be calculated for given pipe I.D. The specific physical dimensions of the pipe and intervening shielding will be input into MicroShield™ using the "Cylinder Surface, External Dose Point Geometry" to calculate the gamma dose rate at a point one meter from the floor or wall surface. Subsequent calculations will be performed while reducing the contaminated area size for each calculation until the dose rates bound the hot spot area of interest. The dose rate for the initial pipe area will be divided by the dose rate for the reduced area of pipe to calculate area factors for the given hot spot size. Equation 5-5 may then be applied to provide the $DCGL_{EMC}$.

5.3.6.4 Remediation and Reclassification

As shown in Table 5-10, Class 1 or Class 2 areas of elevated residual activity above the $DCGL_{EMC}$ are remediated to reduce the residual radioactivity to acceptable levels. Based on survey data, it may be necessary to remediate an entire survey unit or only a portion of it. If an individual survey measurement (scan or direct) in a Class 2 survey unit exceeds the $DCGL_W$, the survey unit or a portion of it may be reclassified to a Class 1 survey unit and the survey redesigned and re-performed accordingly. If an individual survey measurement in a Class 3 survey unit exceeds 0.5 $DCGL_W$, the survey unit, or portion of a survey unit, will be evaluated, and if necessary, reclassified to a Class 2 survey unit and the survey redesigned and re-performed accordingly.

Table 5-10

Investigation Actions for Individual Survey Unit Measurements

Area Classification	**Action if Investigation Results Exceed:**		
	$DCGL_{EMC}$	**$DCGL_W$**	**0.5 $DCGL_W$**
Class 1	Remediate and resurvey as necessary	Acceptable*	N/A
Class 2	Remediate, reclassify portions as necessary and investigate**	Reclassify portions as necessary and investigate**	N/A
Class 3	Remediate, reclassify portions as necessary and investigate**	Reclassify portions as necessary, increase scan coverage and investigate**	Reclassify portions as necessary and resurvey, increase scan coverage

*For individual measurements above DCGL, the Sign Test will be conducted on the survey unit and an EMC evaluation performed.

**Requires an investigation of the initial classification process and a survey unit evaluation of sufficient intensity to satisfy the requirements of new classification status.

5.3.6.5 Resurvey

Following an investigation, if a survey unit is reclassified to a more restrictive classification or if remediation activities were performed, a resurvey is performed in accordance with approved procedures. If a Class 2 area had contamination greater than the $DCGL_W$, it should be reclassified to a Class 1 area. If the average value of Class 2 direct survey measurements was less than the $DCGL_W$, the scan MDC was sensitive enough to detect the $DCGL_{EMC}$ and there were no areas greater than the $DCGL_{EMC}$, the survey redesign may be limited to obtaining a 100% scan without having to re-perform the static measurements or soil sample analyses. This condition assumes that the sample density meets the requirements for a Class 1 area.

5.4 Survey Methods and Instrumentation

5.4.1 Survey Measurement Methods

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable procedure. The techniques for performing survey measurements or collecting samples are specified in approved procedures. FSS measurements include surface scans, direct surface measurements, and gamma spectroscopy of volumetric materials. Methods not specifically described may also be used for final status surveys. If so, Rancho Seco will give the NRC 30 days notice to provide an opportunity to review the associated basis document.

On-site lab facilities are used for gamma spectroscopy, liquid scintillation and gas proportional counting in accordance with applicable procedures. Off-site facilities are used, as necessary. No matter which facilities are used, analytical methods will be administratively established to detect levels of radioactivity at 10% to 50% of the DCGL value.

5.4.1.1 Structures

Structures will receive scan surveys, direct measurements and, when necessary, volumetric sampling.

5.4.1.1.1 Scan Surveys

Scanning is performed in order to locate small, elevated areas of residual activity above the investigation level. Structures are scanned for beta-gamma radiation with appropriate instruments such as those listed in Table 5-11. The measurements will typically be performed at a distance of 1 cm or less from the surface and at a scan speed of 5 cm/sec for hand-held instruments. Adjustments to scan speed and distance may be made in accordance with approved procedures. *In situ* gamma spectroscopy may be effectively substituted for scanning surveys in accordance with DTBD-06-003, "Use of *In Situ* Gamma Spectroscopy for Final Status Surveys," [Reference 5-10].

5.4.1.1.2 Direct Measurements

Direct measurements are performed to detect surface activity levels. Direct measurements are conducted by placing the detector on or very near the surface to be counted and acquiring data over a pre-determined count time. A count time of one minute is typically used for surface measurements and generally provides detection levels well below the DCGL. (The count time may be varied provided the required detection level is achieved).

5.4.1.1.3 Concrete With Activated Radionuclides

Residual radioactivity within activated building materials will be measured volumetrically. Following remediation of activated concrete and rebar down to the activated concrete DCGL, the remaining surface will be volumetrically sampled by coring or other means. DCGLs for activated concrete have been determined and are provided in LTP Chapter 6, Table 6-10. Per

Table 5-12, the gamma spectroscopy instrumentation achieves an MDC much lower than the DCGL.

5.4.1.1.4 Volumetric Concrete Measurements

Volumetric sampling of contaminated concrete, as opposed to direct measurements may be necessary if the efficiency or uncertainty of the gross beta measurements are too high. Volumetric concrete samples will be analyzed by gamma spectroscopy. The results will either be evaluated by 1) calculating the derived total gross beta dpm/100 cm^2 in the sample and comparing the gross beta results directly to the gross beta DCGL or 2) by using the radionuclide specific results to derive the surface activity equivalent and determine compliance using the unity rule. Use of the unity rule will require the use of a surrogate calculation to account for the radionuclides in the mixture not identified by gamma spectroscopy. This will be accomplished using the nuclide mixtures listed in Tables 2-16 or 2-19 in Chapter 2 of this LTP, as appropriate.

Volumetric samples analyzed by gamma spectroscopy will detect the presence of radioactivity below the surface. Such sampling is typically performed following removal of paint and other surface coatings during remediation. After analysis, the data may be converted to equivalent surface activity for crack or rough surface analysis.

5.4.1.2 Soils

Soil will receive scan surveys at the coverage level described in Table 5-6 and volumetric samples will be taken at designated locations. Surface soil samples will normally be taken at a depth of 0 to15 cm. Areas of subsurface soil contamination may require sampling at a depth exceeding 15 cm. The possibility of sub-surface contamination will be considered during the survey design process and the survey design package will contain requirements for sampling soil below 15 cm. Samples will be collected and prepared in accordance with approved procedures.

5.4.1.2.1 Scans

Open land areas are scanned for gamma emitting nuclides. The gamma emitters are used as surrogates for the HTD radionuclides. Sodium iodide detectors are typically used for scanning. For detectors such as the SPA-3, the detector is held within 2.5 to 5 centimeters of the ground surface and is moved at a speed of 0.5 m/sec, traversing each square meter 5 times. The area covered by scan measurements is based on the survey unit classification as described in Section 5.3.2.

5.4.1.2.2 Volumetric Samples

Soil materials are analyzed by gamma spectroscopy. Soil samples of approximately 1,500 grams are normally collected from the surface layer (top 15 cm). If contamination below 15 cm is suspected, split spoon sampling or similar methods, will be used for the final survey. Sample preparation includes removing extraneous material, homogenizing, and drying the soil for gamma isotopic analysis. Separate containers are used for each sample and each container is moved through the analysis process following site procedures. Samples are split when required by the applicable quality control procedures.

If a survey area has already been excavated and remediated to the soil DCGL, this area will be treated as a surface soil, and the FSS will be performed on the excavated area. Soil samples will be collected to depths at which there is high confidence that deeper samples will not result in higher concentrations. Alternatively, a sodium-iodide detector or intrinsic germanium detector of sufficient sensitivity to detect DCGL concentrations may be utilized to identify the presence or absence of subsurface contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the DCGL, confirmatory investigation and analyses of soil samples of the suspect areas will be performed. All subsurface sampling will be performed in accordance with the guidance in Section G.2.1 of NUREG-1757, Volume 2. The sample size for subsurface samples will be determined using the same methods described for surface soil. Per NUREG-1757, Volume 2, scanning is not applicable to subsurface areas; however, Rancho Seco FSSs will employ scanning techniques commensurate with the survey unit classification. Scanning on subsurface soils, where accessible as an excavated surface, will demonstrate compliance with site release criteria.

5.4.2 Specific Survey Area Considerations

5.4.2.1 Pavement-Covered Areas

Survey of paved areas will be required along the roadways providing ingress and egress to the site. Evaluation has determined that paved roadways are Class 3 areas. The survey design of paved areas will be based on soil survey unit sizes since they are outdoor areas where the exposure scenario is most similar to direct radiation to surface soil. The applicable DCGL will be the soil DCGL. Scan and static gamma and beta-gamma surveys are made as determined by the survey unit design. If the potential exists for sub-surface contamination under pavement, either the pavement/asphalt will be removed prior to the FSS or samples/measurements obtained through the pavement. Paved areas may be separate survey units or they may be incorporated into surveys of other adjacent open land areas of like classification. Surveys of paved areas may include road right-of-ways to check for radioactivity relocated from water runoff. Right-of-ways may also be separate survey units.

5.4.2.2 Stored Excavated Soil

The primary method for evaluation of excavated soils originating from Class 1 and Class 2 areas will follow the guidance provided in MARSSIM for FSS of Class 1 areas. Excavated soil will be evaluated in accordance with Rancho Seco procedures to determine suitability for transport and final status evaluation. Prior to performing a FSS, excavated soil will be characterized to determine suitability for transport to an area dedicated for excavated soils. Soils that do not contain residual radioactivity greater than DCGL values will be relocated to an area dedicated for soil evaluation and graded to a maximum depth of one meter. A Class 1 final status survey will then be conducted with soil measurements averaged over the total depth of soil. Sample/measurement density will be equal to that needed for a surface soil survey of the same volume. Surface scanning and volumetric analyses will be directly compared with DCGL values. Any measurement location identifying residual radioactivity above the DCGL will be investigated and remediated as necessary. Controls will be instituted to prevent mixing of soils from different survey areas prior to evaluation. Soils satisfying the criteria for unrestricted release will be stockpiled for use as onsite backfill material. (Class 2 material could be used in either Class 1 or 2 areas and Class 1 material could only be used in Class 1 areas.)

The radiological evaluation of soils resulting from minor trenching and digging efforts in Class 3 defined areas (no reasonable potential for subsurface contamination) will be performed by characterization survey in accordance with site procedures. Excavated soils that demonstrate residual radioactivity consistent with Class 3 status will be released for use as onsite excavation backfill.

5.4.2.3 Embedded Piping and Buried Piping

Residual radioactivity on internal surfaces, such as floor drains, embedded piping, and buried piping may be inaccessible or difficult to measure directly using field survey detectors and established techniques. Where no remediation has occurred, inaccessible or difficult to measure internal surfaces are assumed to have the same level of residual radioactivity as that found on accessible internal surfaces. No special measurement methods are applied.

Where remediation has occurred, representative samples of the inaccessible internal surfaces are obtained, an assessment of pre-remediation survey data is performed, or other appropriate measures are taken (e.g., calibrated detectors extended into piping runs in a controlled manner) such that a reasonable approximation of the residual radioactivity on the inaccessible internal surfaces can be made. Accessible internal surfaces are surveyed the same as other structural surfaces. Scale and sediment samples may be obtained, if appropriate.

5.4.2.4 Cracks, Crevices, Wall-Floor Interfaces and Small Holes

Surface contamination on irregular structure surfaces (e.g., cracks, crevices, and holes) are difficult to survey directly. Where no remediation has occurred and residual activity has not been detected above background, these surface blemishes may be assumed to have the same level of residual activity as that found on adjacent surfaces. The accessible surfaces are surveyed in the same manner as other structural surfaces and no special corrections or adjustments have to be made.

In situations where remediation has taken place or where residual activity has been detected above background, a representative sample of the contamination within the crack or crevice may be obtained or an adjustment for instrument efficiency may be made if justifiable. If an instrument efficiency adjustment cannot be justified based on the depth of contamination or other geometry factors, volumetric samples will be collected. The total dpm/100 cm^2 contained in the volumetric sample that is attributable to the beta emitting radionuclides used to determine the DCGL will be compared directly to the concrete gross activity DCGL. As an alternative, radionuclide specific analysis, coupled with application of the unity rule may be used.

Volumetric samples analyzed by gamma spectroscopy will detect the presence of radioactivity below the surface. Such sampling is typically performed following removal of paint and other surface coatings during remediation. After analysis, the data may be converted to equivalent surface activity.

The accessible surfaces are surveyed in the same manner as other structure surfaces except that they are included in areas receiving judgmental scans when scanning is performed over less than 100% of the area.

5.4.2.5 Paint Covered Surfaces

Final status surveys will consider the effect of painted surfaces on instrument efficiency in accordance with DTBD-05-010, "Beta Detection Including Beta Energy and Source Efficiency," [Reference 5-11]. Gross measurements will not be used in areas covered by thick paints or coatings. The surfaces will be volumetrically sampled or the coating will be removed prior to survey. No special consideration must be given to wall or ceiling areas painted before plant startup and which have not been subjected to repeated exposure to materials that would have penetrated the painted surface.

5.4.2.6 Reactor Building Liner

Because concrete from the shield walls and floors is being removed, FSS surveys of the interior of the reactor building will be limited to direct measurements and scans of the steel liner. Instrument efficiency will be adjusted following the method described in NUREG-1507 to account for the effects of surface coatings, if necessary. Remaining concrete or structural features will be surveyed as described above.

5.4.3 Instrumentation

Radiation detection and measurement instrumentation for the FSS is selected to provide both reliable operation and adequate sensitivity to detect the radionuclides identified at the site at levels sufficiently below the DCGL. Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field. The instrumentation will, to the extent practicable, use data logging with bar code scanning capability.

Commercially available portable and laboratory instruments and detectors are typically used to perform the three basic survey measurements: 1) surface scanning; 2) direct surface contamination measurements; and 3) spectroscopy of soil and other bulk materials, such as concrete.

Radiation Protection procedures and Decommissioning Survey Implementing Procedures (DSIPs) control the issuance, use, and calibration of instrumentation. Records supporting the instrumentation program are maintained in accordance with site document control procedures.

5.4.3.1 Instrument Selection

Radiation detection and measurement instrumentation is selected based on the type and quantity of radiation to be measured. The instruments used for direct measurements are capable of detecting the radiation of concern to a Minimum Detectable Concentration (MDC) of between 10% and 50% of the applicable DCGL. The use of 10% to 50% of the DCGL is an administrative limit only. Any value below the DCGL is acceptable in Class 1 or 2 survey units. MDCs of less than 50% of the DCGL allow detection of residual activity in Class 3 survey units at an investigation level of 0.5 times the DCGL. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the $DCGL_{EMC}$. Instrumentation currently proposed for use in the FSS is listed in Table 5-11. Instrument MDCs are discussed in Section 5.4.3.4 and nominal MDC values are also listed in Table 5-12.

Other measurement instruments or techniques may be utilized. The acceptability of alternate instruments or technologies for use in the FSS Program would be justified in a technical basis evaluation document. Technical basis evaluations for alternate final status survey instruments or techniques will be provided for NRC review 30 days prior to use. An instrument technical analysis will include the following:

- Description of the conditions under which the method would be used;
- Description of the measurement method, instrumentation and criteria;

- Justification that the technique would provide the required sensitivity for the given survey unit classification in accordance with Table 5-10; and
- Demonstration that the instrument provides sufficient sensitivity for measurement below the release criteria with Type I error equivalent to 5% or less.

Table 5-11

Typical FSS Survey Instrumentation

Measurement Type	Detector Type	Effective Detector Area and Window Density	Instrument and Model	Detector Model
Alpha Scan	Gas-flow proportional	126 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68
Alpha Scan	α Scintillation	0.8 mg/cm^2 - 1.2 mg/cm^2 ZnS(Ag) 125 cm^2	Ludlum 2350-1	Ludlum 43-90
Alpha Static	α Scintillation	0.8 mg/cm^2 - 1.2 mg/cm^2 ZnS(Ag) 125 cm^2	Ludlum 2350-1	Ludlum 43-90
Beta Scan	β Scintillation	1.2 mg/cm^2 0.01" Plastic Scintillation 125 cm^2	Ludlum 2350-1	Ludlum 44-116
Beta Static	β Scintillation	1.2 mg/cm^2 0.01" Plastic Scintillation 125 cm^2	Ludlum 2350-1	Ludlum 44-116
Beta Scan	G-M	15.5 cm^2 1.7 mg/cm^2	Ludlum 2350-1	Ludlum 44-40-2
Beta Static	G-M	15.5 cm^2 1.7 mg/cm^2	Ludlum 2350-1	Ludlum 44-40-2
Beta Scan	G-M	15.5 cm^2 1.7 mg/cm^2	Ludlum 2350-1	Ludlum 44-9
Beta Static	G-M	15.5 cm^2 1.7 mg/cm^2	Ludlum 2350-1	Ludlum 44-9
Beta-Gamma Scan	Gas-flow proportional	31 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-51
Beta-Gamma Static	Gas-flow proportional	31 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-51
Beta-Gamma Scan	Gas-flow proportional	55 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-116-1
Beta-Gamma Static	Gas-flow proportional	55 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-116-1

Table 5-11
Typical FSS Survey Instrumentation

Measurement Type	**Detector Type**	**Effective Detector Area and Window Density**	**Instrument and Model**	**Detector Model**
Beta-Gamma Scan	Gas-flow proportional	126 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68
Beta-Gamma Scan	Gas-flow proportional	584 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-37
Beta-Gamma Static	Gas-flow proportional	584 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-37
Gamma Scan	Scintillation	2" diameter x 2" length NaI	Ludlum 2350-1	Ludlum 44-10 or Eberline SPA-3
Static Surface Contamination	Gas-flow proportional	126 cm^2 0.8 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-68
Static Surface Contamination	Scintillation	126 cm^2 1.2 mg/cm^2 Aluminized Mylar	Ludlum 2350-1	Ludlum 43-90 Ludlum 44-116
Soil, Structure Surfaces and Bulk Material	High-purity Germanium	N/A	Canberra Lab or *In Situ* Detector	N/A
Gamma Pipe Scans and Directs	CsI NaI NaI NaI Gas-Flow Gas-Flow Gas-Flow	0.75" x 0.75" 2" x 2" 3" x 3" 0.75" x 3" 181 cm^2 122 cm^2 60 cm^2	Ludlum 2350-1	Ludlum 44-159 Ludlum 44-157 Ludlum 44-162 Bicron 1062000 Ludlum 43-98 Ludlum 43-111 Ludlum 43-94

Table 5-12
Typical FSS Detection Sensitivities

Instruments and Detectors[a]	Radiation	Background Count Time (minutes)	Background (cpm)	Instrument Efficiency[b] (ε_t)	Count Time (minutes)	Static MDC[c] (dpm/100 cm^2)	Scan MDC
Model 43-68	Alpha	10	1	0.074[e]	5.0	26	N/A[f]
Model 43-68	Beta-Gamma	1	300	0.146[g]	1.0	454	1082[d]
Model 44-116	Beta	1	300	0.162[h]	1.0	413	1063[d]
Model 43-90	Alpha	10	3	0.077[e]	5.0	39	N/A[f]
Model 43-116-1	Beta-Gamma	1	200	0.099[i]	1.0	1,262	5,547[d]
Model 43-51	Beta-Gamma	1	37	0.071[i]	1.0	1,395	4,734[d]
Model 43-37	Beta-Gamma	1	1,200	0.138[i]	1.0	204	635[d]
Model 44-9	Beta-Gamma	1	36	0.215[h]	1.0	926	2,719[d]
Model 44-40-2	Beta-Gamma	1	27	0.204[h]	1.0	858	2,481[d]
Model 44-10	Gamma	1	8,000	N/A	0.02	N/A	5.2 pCi/g[j]
Model SPA-3	Gamma	1	8,000	N/A	0.02	N/A	5.2 pCi/g[j]
HPGe	Gamma	Up to 60	N/A	0.40 relative	10-60	005 pCi/g volumetric	2000 to 4000 dpm/100 cm^2 0.15-0.30 pCi/g[k] vol.
Canberra Inspector 1000[o]	Gamma	Up to 60	N/A	0.085 relative	1-60	N/A	N/A
Beckman Liquid Scintillation	H-3	30 & 60	40 dpm	0.40	30 & 60	800 pCi/L	N/A
Tennelec Low Bkg Counter	Alpha Beta	10 10	0.1 1.0	0.41 0.48	1-10	<11 < 16	N/A N/A
Pipe Detectors:							
Model 44-159[l]	Gamma	1	677	0.024	1	5,200	N/A[n]
Model 44-157[l]	Gamma	1	6,300	0.224	1	1,445	N/A[n]
Model 44-162[l]	Gamma	1	16,000	0.568	1	1,041	N/A[n]
Model 1062000[l]	Gamma	1	1,250	0.050	1	3,321	N/A[n]
Model 43-98	Beta-Gamma	1	290	0.160[m]	1	284	N/A[n]
Model 43-111	Beta-Gamma	1	100	0.151[m]	1	266	N/A[n]
Model 43-94	Beta-Gamma	1	44	0.227[m]	1	248	N/A[n]

[a] Detector models listed are used with the Ludlum 2350-1 Data Logger

[b] Calibration sources are Tc-99 and Pu-239. The efficiency is determined by counting the source with the detector in a fixed position from the source (reproducible geometry). The ε_t value is based on ISO-7503-1, "Evaluation of Surface Contamination - Part 1: Beta Emitters and Alpha Emitters (first edition)," [Reference 5-12] and conditions noted for each detector.

[c] Static MDC is calculated per Equation 5-7 and is the same as Brodsky and Gallagher in Table 3.1 of NUREG-1507. For conditions where the background and sample count time differ, the formula from Strom and Stanbury in Table 3.1 of NUREG-1507 is used.

[d] Scan MDC, in dpm/100 cm^2, is calculated per Equation 5-8 assuming a scan rate of 5.08 cm/sec, which is equivalent to a count time of 1.73 seconds (0.028 minutes) using a detector width of 8.8 cm. The 43-37 detector assumes a scan rate of 12.7 cm/s and results in a count time of 1.05 seconds (0.018 minutes) for a detector width of 13.34 cm. The 44-116 detector's width is 7.5 cm and results in a count time of 1.48 seconds at 5.08 cm/s scan speed. The 43-116-1 detector's width is 2.54 cm and results in a count time of 0.5 seconds at a scan speed of 5.08 cm/s. The 43-51 detector's width is 3.81 cm and at a scan rate of 5.08 cm/s results in a count time of 0.75 seconds. Both the 44-9 and 44-40-2 have window fields of vision of 5.08 cm wide and result in a 1.0 second count time for a scan speed of 5.08 cm/s.

[e] The ε_t value for the alpha mode of the 43-68 and 43-90 detectors is the 2π value as defined in ISO-7503-1 times the Standard's ε_s value of 0.25 for Pu-239

[f] Using the approach described in Section 6.7.2.2 of MARRSIM and a scan alpha DCGL value of 60 dpm/100 cm^2 results in a 21% probability for a surveyor to audibly detect 1 cpm, then acquire a static count for approximately 15 seconds to confirm the presence of alpha activity. A low probability of detection at the typical DCGL level implies that alpha scanning may not be practical.

[g] The ε_t used for the beta mode of the 43-68 is determined from analysis of concrete samples at Rancho Seco. The evaluation and results are found in DTBD-05-010.

[h] The ε_t value for the 44-116, 44-9 and 44-40-2 beta detectors is the 2π value as defined in ISO-7503-1 times the Standard's ε_s value of 0.50 for Cs-137.

[i] The 43-116, 51 and 37 are gas flow proportional detectors (GFPD) of the same type and window thickness as the 43-68 detector. The ε_s value for these detectors was derived from the concrete study performed using 43-68 detectors.

[j] Scan MDC in pCi/g is calculated using the approach described in Section 6.7.2.1 of MARSSIM for a Cs-137 nuclide fraction of 0.95 and a Co-60 fraction of 0.05 with a determined detector sensitivity of 1000 and 430 cpm per mR/hr for each radionuclide respectively. The weighted MicroShield-determined conversion factor was 0.282 pCi/g per uR/hr.

[k] In situ spectroscopy HPGe uses the "count to MDA" function in order to achieve the required MDC.

[l] The efficiency varies for the pipe detectors depending on the pipe diameter used. The efficiency used for the table is the averaged efficiency value for the pipe diameters. The detectors and diameters are: model 44-159: 2-4 in. dia., model 1062000: 1-4 in. dia., model 44-157: 4-8 in. dia., model 44-162: 8-12 in. dia.

[m] The 43-94, 98 and 111 are GFPD pipe detectors and the efficiency used is the ε_t value. The source used is a metal source and the piping area of interest will be cleaned metal surfaces.

[n] Piping surveys may be conducted without scanning. Direct measurements will be acquired at intervals of 6 to 12 inches.

[o] Portable MCA multi-faceted instrument to be used to perform field spectrum analysis in varied data collection modes which may include in-situ object calibration software (ISOCS). Site procedures, instructions and as required, DTBD's will define the analysis parameters.

5.4.3.2 Calibration And Maintenance

Instruments and detectors are calibrated for the radiation types and energies of interest at the site. The calibration source for beta survey instruments is Cs-137 because the average beta energy (188 keV) approximates the beta energy of the radionuclides found on surfaces or in piping on site (average beta energy of 166 keV). The alpha calibration source when used is Pu-239 that has an appropriate alpha energy for plant-specific alpha emitting nuclides. Gamma scintillation detectors are typically calibrated using Cs-137.

Instrumentation used for final status survey will be calibrated and maintained in accordance with RP.311.II.03, Ludlum 2350-1 Datalogger Calibration procedure. Radioactive sources used for calibration are traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. When characterized HPGe detectors are used, suitable NIST-traceable sources are used for calibration, and the software is set up appropriately for the desired geometry. If vendor services are used, these will be obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality.

5.4.3.3 Response Checks

Instrumentation response checks are conducted to assure proper instrument response and operation. An acceptable response for field instrumentation is an instrument reading within ±20% of the established check source value. Laboratory instrumentation standards will be within ±3 sigma as documented on a control chart. Response checks are performed daily before instrument use and again at the end of use. Check sources contain the same type of radiation as that being measured in the field and are held in fixed geometry jigs for reproducibility. If an instrument fails a response check, it is labeled with a Rancho Seco "Radiac Repair Tag" and is removed from service until the problem is corrected in accordance with applicable procedures. Measurements made between the last acceptable check and the failed check are evaluated to determine if they should remain in the data set.

5.4.3.4 Minimum Detectable Concentration (MDC)

The MDC is determined for the instruments and techniques used for final status surveys (Table 5-12). The MDC is the concentration of radioactivity that an instrument can be expected to detect 95 percent of the time.

5.4.3.4.1 Static MDC For Structure Surfaces

For static (direct) surface measurements, with conventional detectors, such as those listed in Table 5-12, the MDC is calculated by Equation 5-7 as follows:

$$MDC_{static} = \frac{3 + 4.65\sqrt{B}}{(K)(t)}$$

Equation 5-7

where:

MDC_{static} = minimum detectable concentration for direct counting (dpm/100 cm^2),

B = number of background counts during the count interval t,

t = count interval (for paired observations of sample and blank, usually 1 minute), and

K = calibration constant (counts/min per dpm/100 cm^2).

The value of K includes correction factors for efficiency (ε_i and ε_s). The value of ε_s is dependent on the material type. Corrections for radionuclide absorption have been made.

5.4.3.4.2 Structural Surface Beta-Gamma Scan MDCs

Following the guidance of Sections 6.7 and 6.8 of NUREG-1507, MDCs for surface scans of structural surfaces for beta and gamma emitters will be computed by Equation 5-8 below. For determining scan MDCs, a rate of 95% of correct detections is required and a rate of 60% of false positives is determined to be acceptable: therefore, a sensitivity index value of 1.38 was selected from Table 6.1 of NUREG-1507 and Equation 5-7 becomes:

$$MDC_{structural\ surface\ scan}\ (dpm/100\ cm^2) = \frac{1.38\sqrt{B}}{\sqrt{p}\,\varepsilon_i\,\varepsilon_s\left(\frac{A}{100}\right)t}$$

Equation 5-8

where:

B = number of background counts during the count interval t,

p = surveyor efficiency,

ε_i = instrument efficiency for the emitted radiation (cpm per dpm),

ε_s = source efficiency (intensity) in emissions per disintegration,

A = sensitive area of the detector (cm^2), and

t = time interval of the observation while the probe passes over the source (minutes).

The numerator in Equation 5-8 represents the minimum detectable count rate that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) will be taken to be 0.5, as recommended by Section 6.7.1 of NUREG-1507. The factor of 100 corrects for probe areas that are not 100 cm^2. In the case of a scan measurement, the counting interval is the time the probe is actually over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. The source efficiency term (ε_s) in Equation 5-8 may be adjusted to account for effects such as self-absorption, as appropriate.

5.4.3.4.3 Total Efficiency (ε_t) and Source Efficiency (ε_s) for Concrete Contamination

The source term inventory on contaminated concrete appears to be primarily located within the first millimeter of the concrete surface. Various fixed point measurement alternatives for determining the source term were evaluated including gross beta measurements on the surfaces, volumetric concrete sampling and *in situ* gamma spectroscopy. Gross beta fixed point measurements were determined to be cost-effective and technically defensible under the assumption that the instrument efficiencies for concrete could be satisfactorily calculated using the methods recommended in NUREG-1507. Determination of the average beta energy and detector response for structure surveys is described in DTBD 05-010.

For scan surveys, gross beta measurements appear to be a practical method. Under certain conditions, *in situ* gamma spectroscopy may be a reasonable method for replacing beta scan surveys.

The methods for determining efficiency in NUREG-1507 were specifically developed to address situations when the source, in this case concrete, affects radiation emission rate due to self-attenuation, backscatter, thin coverings, etc. This method accounts for these source effects by separating the efficiency calculation into two components, i.e., instrument efficiency ε_i and source efficiency ε_s. The total efficiency ε_t, is the product of ε_i and ε_s as shown below.

$$\varepsilon_t = (\varepsilon_i)(\varepsilon_s)$$

Equation 5-9

The ε_i was determined by calibration to a NIST traceable, large area Cs-137 source. The ε_s value was determined empirically through measurements of concrete cores and volumetric samples collected from representative site locations. Samples were taken from each of the buildings. The sample nuclide activities were determined by gamma spectrometry, then the pCi/g result was multiplied by the mass of the core sample and converted to total gross beta dpm. Detector response in c/m was divided by the sample activity in dpm to determine the ε_t. Dividing ε_t by ε_i resulted in the empirically derived ε_s. The empirically derived ε_s value of 0.46 compares reasonably with the ISO standard default value of 0.5 for beta energies greater than 0.4 MeV, considering most of the concrete activity is Cs-137 with a beta energy greater than 0.4.

The method used to adjust the efficiency for the effect of surface coatings was taken from NUREG-1705 and is discussed in DTBD 05-010.

5.4.3.4.4 Structural Surface Alpha Scan MDCs

In cases where alpha scan surveys are required, MDCs must be quantified differently than those for beta-gamma surveys because the background count rate from a typical alpha survey instrument is nearly zero (1 to 3 counts per minute typically). Since the time that an area of alpha activity is under the probe varies and the background count rates of alpha survey instruments is so low, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates.

For alpha survey instrumentation with a background around one to three counts per minute, a single count will give a surveyor sufficient cause to stop and investigate further. Thus, the probability of detecting given levels of alpha emitting radionuclides can be calculated by use of Poisson summation statistics. Doing so (see MARRSIM Section 6.7.2.2 and Appendix J for details), one finds that the probability of detecting an area of alpha activity of 300 dpm/100 cm^2 at a scan rate of 3 cm per second (roughly 1 inch per second) is 90% if the probe dimension in the direction of the scan is 10 cm. If the probe dimension in the scan direction is halved to 5 cm, the detection probability is still 70%. Choosing appropriate values for surveyor efficiency, instrument and surface efficiencies will yield MDCs for alpha surveys for structure surfaces. If for some reason lower MDCs are desired, then scan speeds can be adjusted, within practical limits, via the methods of Section 6.7.2.2 and Appendix J of MARSSIM.

5.4.3.4.5 Open Land Area Gamma Scan MDCs

In addition to the MDCR and detector characteristics, the scan MDC (in pCi/g) for land areas is based on areal extent of the hot spot, depth of the hot spot, and the radionuclide (i.e., energy and yield of gamma emissions). If one assumes constant parameters for each of the above variables, with the exception of the specific radionuclide in question, the scan MDC may be reduced to a function of the radionuclide alone.

The evaluation of open land areas requires a detection methodology of sufficient sensitivity for the identification of small areas of potentially elevated activity. Scanning measurements are performed by passing a 2" x 2" NaI(TI) gamma scintillation detector in gross count rate mode across the land surface under investigation. The centerline of the detector is maintained at a source-to-detector distance of less than 10 cm and moved from side to side in a 1-meter wide pattern at a rate of 0.5 m/sec. This serpentine scan pattern is designed to cross each survey cell (one square meter) five times in approximately ten seconds with a maximum separation of less than 50 cm between any path. The audible signal is monitored for detectable increases in count rate. An observed count rate increase results in further investigation to verify findings and define the level and extent of residual radioactivity.

This method represents the Stage 1 and Stage 2 surface scanning process for land areas defined in NUREG-1507 and is the basis for calculation of the scanning detection sensitivity (scan MDC). The sensitivity of this counting system has been verified by empirical measurements (DTBD 05-012, "Eberline SPA-3 and Ludlum 44-10 Detector Sensitivity (MDC)," [Reference 5-13]). The sensitivity is only slightly affected by the relative amounts of Cs-137 and Co-60 in the soil giving typical scan MDC values in the range of 5 to 6 pCi/g (approximately 10 percent of the soil DCGL) for instrument backgrounds of 8,000 to 10,000 cpm. Alternate methods of sufficient sensitivity for the identification of small areas of elevated radioactivity may be used where appropriate.

An *a priori* determination of scanning sensitivity is performed to ensure that the measurement system is able to detect concentrations of radioactivity at levels below the regulatory release limit. Expressed in terms of Scan MDC, this sensitivity is the lowest concentration of radioactivity for a given background that the measurement system is able to detect at a specified performance level and surveyor efficiency. The scan MDC value (in pCi/g) for open land area surface scanning is developed in the following steps following the guidance of MARSSIM, Section 6.7.2.1.

With a desired performance level of 95% correct detections and 60% false positive rate, the sensitivity index has a value of 1.38 resulting in a minimum detectable count rate (*MDCR*) of:

$$MDCR = 1.38\sqrt{b_i} \times (60\ \text{sec}/1\ \text{min})$$

Equation 10

where:

b_i = background counts in the observation interval.

Introducing the human factor performance element of surveyor efficiency, the surveyor minimum detectable count rate becomes:

$$MDCR_{surveyor} = \frac{MDCR}{\sqrt{p}}$$

Equation 11

where:

$MDCR_{surveyor}$ = Minimum detectable surveyor count rate (cpm), and

p = Surveyor efficiency = 0.5.

A corresponding minimum detectable exposure rate can be determined for a specified detector and radionuclide by dividing the $MDCR_{surveyor}$ value by the detector manufacturer's count rate to exposure rate ratio (cpm per μR/h) to give a minimum detectable exposure rate in units of μR/h. The minimum detectable exposure rate is then used to determine the minimum detectable radionuclide concentration (i.e., the Scan MDC) by modeling a specified small area of elevated activity using MicroShield™ to yield a conversion factor of μR/h per pCi/g. The minimum detectable exposure rate is then divided by the MicroShield™ conversion factor to give a Scan MDC in units of pCi/g.

5.4.3.4.6 HPGe Spectrometer Analysis

The onsite chemistry laboratory maintains two gamma isotopic spectrometers that are calibrated to various sample geometries, including a one-liter marinelli geometry for soil analysis. These systems are calibrated using a NIST-traceable mixed gamma source. Both detectors are manufactured by Canberra and operate using the Genie PC platform from Canberra Industries. Laboratory counting systems have software controlled count times which are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil; this is calculated by Equation 5-12 as follows:

$$MDC\ (pCi/g) = \frac{3 + 4.65\sqrt{B}}{K * V * t}$$

Equation 5-12

where:

B = number of background counts during the count interval t,

K = proportionality constant that relates the detector response to the activity level in a sample for a given set of measurement conditions,

V = mass of sample (g), and

t = count time (minutes)

An HPGe detector has been obtained for *in situ* gamma spectroscopy of soils and structures. Its sensitivity is similar to that of the lab spectrometer and is documented in DTBD-06-003.

5.4.3.4.7 Pipe Survey Instrumentation

Remaining pipe will be surveyed to ensure residual remaining activity is less than the DCGL. Pipe survey instruments proposed for use with pipe having diameters between 0.75 and 18 inches have been shown to have efficiencies ranging from approximately 0.02 to 0.57 (Table 5-12). This equates to detection sensitivities of approximately 350 dpm/100 cm^2 to 5,200 dpm/100 cm^2. This level of sensitivity is adequate to detect residual activity below the embedded pipe DCGL of 100,000 dpm/100 cm^2.

5.5 Data Collection and Processing

This section describes data collection, review, validation and record keeping requirements for final status surveys.

5.5.1 Sample Handling and Record Keeping

Sample collection and handling requirements are provided for each sample from the point of collection through obtaining the final results to ensure the validity of the sample data. Sample tracking records are controlled and maintained and, upon completion of the data cycle, are transferred to Document Control, in accordance with applicable procedures.

Each survey unit has a document package associated with it that covers the design and field implementation of the survey requirements. Survey unit records are quality records.

5.5.2 Data Management

Survey data are collected from several sources during the data life cycle and are evaluated for validity throughout the survey process. QC replicate measurements are not used as final status survey data. (See Section 5.8.2.4.1 for design and use of QC measurements.) Measurements performed during turnover and investigation surveys can be used as final status survey data if they were performed according to the same requirements as the final status survey data. These requirements are:

- Survey data shall reflect the as-left survey unit condition; i.e., no further remediation required,
- The application of isolation measures to the survey unit to prevent recontamination and to maintain final configuration are in effect; and
- The data collection and design were in accordance with FSS methods and procedures, e.g., scan MDC, investigation levels, survey data point number and location, statistical tests, and EMC tests.

Measurement results stored as final status survey data constitute the final survey of record and are included in the data set for each survey unit used for determining compliance with the site release criteria. Measurements are recorded in units appropriate for comparison to the applicable DCGL. Numerical values, even negative numbers, are recorded. Measurement records include, at a minimum, the surveyor's name, the location of the measurement, the instrument used, measurement results, the date and time of the measurement, any surveyor comments, and records of applicable reviews.

5.5.3 Data Verification and Validation

The final status survey data are reviewed before data assessment to ensure that they are complete, fully documented and technically acceptable. The review criteria for data acceptability will include at a minimum, the following items:

a) The instrumentation MDC for fixed or volumetric measurements was below the $DCGL_W$ or if not, it was below the $DCGL_{EMC}$ for Class 1, below the $DCGL_W$ for Class 2 and below 0.5 $DCGL_W$ for Class 3 survey units,

b) The instrument calibration was current and traceable to NIST standards,

c) The field instruments were source checked with satisfactory results before and after use each day data were collected or data was evaluated if instruments did not pass a source check in accordance with Section 5.4.3.3,

d) The MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey,

e) The survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed,

f) "Special methods" for data collection were properly applied for the survey unit under review. These special methods are either described in this LTP section or will be the subject of an NRC notice of opportunity for review,

g) The sample was controlled from the point of sample collection to the point of obtaining results,

h) The data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility, and

i) The data have been properly recorded.

If the data review criteria are not met, the discrepancy will be evaluated and the decision to accept or reject the data will be documented in accordance with approved procedures. The Rancho Seco Corrective Action Program will be used to document and resolve discrepancies as applicable.

5.5.4 Graphical Data Review

Survey data may be graphed to identify patterns, relationships or possible anomalies which might not be so apparent using other methods of review. A posting plot or a frequency plot may be made. Other special graphical representations of the data will be made as the need dictates.

5.5.4.1 Posting Plots

Posting plots may be used to identify spatial patterns in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal patches of elevated radioactivity or local areas in which the DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Incongruities in the background data may be the result of residual, undetected activity, or they may just reflect background variability.

5.5.4.2 Frequency Plots

Frequency plots may be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity. Multiple peaks in the data may indicate the presence of isolated areas of residual radioactivity or background variability due to soil types or differing materials of construction. Variability may also indicate the need to more carefully match background reference areas to survey units or to subdivide the survey unit by material or soil type.

5.6 Data Assessment and Compliance

An assessment is performed on the final status survey data to ensure that they are adequate to support the determination to release the survey unit. Simple assessment methods such as comparing the survey data to the DCGL or comparing the mean value to the DCGL are first performed. The statistical tests are then applied to the final data set and conclusions are made as to whether the survey unit meets the site release criterion.

5.6.1 Data Assessment Including Statistical Analysis

The results of the survey measurements are evaluated to determine whether the survey unit meets the release criterion. In some cases, the determination can be made without performing complex, statistical analyses.

5.6.1.1 Interpretation of Sample Measurement Results

An assessment of the measurement results is used to quickly determine whether the survey unit passes or fails the release criterion or whether one of the statistical analyses must be performed. The evaluation matrices are presented in Tables 5-13 and 5-14.

Table 5-13

Interpretation of Sample Measurements When the WRS Test Is Used

Measurement Results	Conclusion
Difference between maximum survey unit concentration and minimum reference area concentration is less than $DCGL_W$	Survey unit meets release criterion
Difference of survey unit average concentration and reference average concentrations greater than $DCGL_W$	Survey unit fails
Difference between any survey unit concentration and any reference area concentration is greater than $DCGL_W$ and the difference of survey unit average concentration and reference area average concentration is less than $DCGL_W$.	Conduct WRS test and elevated measurements test

Table 5-14

Interpretation of Sample Measurements When the Sign Test Is Used

Measurement Results	Conclusion
All concentrations less than $DCGL_W$	Survey unit meets release criterion
Average concentration greater than $DCGL_W$	Survey unit fails
Any concentration greater than $DCGL_W$ and average concentration less than $DCGL_W$	Conduct Sign Test and elevated measurements test

When required, one of four statistical tests will be performed on the survey data:

1) WRS Test
2) Sign Test
3) WRS Test Unity Rule
4) Sign Test Unity Rule

In addition, survey data are evaluated against the EMC criteria as previously described in Section 5.3.6.3 and as required by NUREG-1757, Volume 2. The statistical test is based on the null hypothesis (H_o) that the residual radioactivity in the survey unit exceeds the DCGL. There must be sufficient survey data at or below the DCGL to reject the null hypothesis and conclude the survey unit meets the site release criterion for dose. Statistical analyses are performed using a specially designed software package or, if necessary, using hand calculations.

5.6.1.2 Wilcoxon Rank Sum Test

The WRS test, or WRS Unity Rule (NUREG-1505, Chapter 11), may be used when the radionuclide of concern is present in the background or measurements are used that are not radionuclide-specific. In addition, this test is valid only when "less than" measurement results do not exceed 40 percent of the data set.

The WRS test is applied as follows:

1) The background reference area measurements are adjusted by adding the $DCGL_W$ to each background reference area measurement, X_i; i.e., $Z_i = X_i + DCGL$.

2) The number of adjusted background reference area measurements, m, and the number of survey unit measurements, n, are summed to obtain N, ($N = m + n$).

3) The measurements are pooled and ranked in order of increasing size from 1 to N. If several measurements have the same value, they are assigned the average rank of that group of measurements.

4) The ranks of the adjusted background reference area measurements are summed to obtain W_r.

5) The value of W_r is compared with the critical value in Table I.4 of MARSSIM. If W_r is greater than the critical value, the survey unit meets the site release dose criterion. If W_r is less than or equal to the critical value, the survey unit fails to meet the criterion.

5.6.1.3 Sign Test

The Sign test and Sign test Unity Rule are one-sample statistical tests used for situations in which the radionuclide of concern is not present in background, or is present at acceptable low fractions compared to the $DCGL_W$. If present in background, the gross measurement is assumed to be entirely from plant activities. This option is used when it can be reasonably expected that including the background concentration will not affect the outcome of the Sign test. The advantage of using the Sign test is that a background reference area is not needed.

The Sign test is conducted as follows:

1) The survey unit measurements, X_i, i = 1, 2, 3, ...N; where N = the number of measurements, are listed.

2) X_i is subtracted from the $DCGL_W$ to obtain the difference $D_i = DCGL_W - X_i$, where i = 1, 2, 3,..., N.

3) Differences where the value is exactly zero are discarded and N is reduced by the number of such zero measurements.

4) The number of positive differences are counted. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the site release criterion.

5) The value of S+ is compared to the critical value given in Table I.3 of MARSSIM. The table contains critical values for given values of N and α. The value of α is set at 0.05 during survey design. If S+ is greater than the critical value given in the table, the survey unit meets the site release criterion. If S+ is less than or equal to the critical value, the survey unit fails to meet the release criterion.

5.6.2 Unity Rule

5.6.2.1 Multiple Radionuclide Evaluations

The Cs-137 to Co-60 (or other gamma nuclide) ratio will vary in the final survey soil samples, and this will be accounted for using a "unity rule" approach as described in NUREG-1505 Chapter 11. Unity Rule Equivalents will be calculated for each measurement result using the surrogate adjusted Cs-137 DCGL and the adjusted Co-60 DCGL, as shown in the following Equation 5-13.

$$\text{Unity Rule Equivalent} \le 1 = \frac{Cs-137}{DCGL_{Cs-137_s}} + \frac{Co-60}{DCGL_{Co-60}} + \ldots + \frac{R_N}{DCGL_N}$$

Equation 5-13

where:

Cs-137 and Co-60 are the gamma results,

$DCGL_{Cs\text{-}137_s}$ = the surrogate Cs-137$_s$ DCGL, as applicable,

$DCGL_{Co\text{-}60}$ = the Co-60 DCGL,

R_N = any other identified gamma emitting radionuclide, and

$DCGL_N$ = the DCGL for radionuclide N.

The unity rule equivalent results will be used to demonstrate compliance assuming the DCGL is equal to 1.0 using the criteria listed in the LTP, Tables 5-13 and 5-14. If the application of the WRS or Sign test is necessary, these tests will be applied using the unity rule equivalent results and assuming that the DCGL is equal to 1.0. An example of a WRS test using the unity rule is provided in NUREG-1505, Page 11-3, Section 11.4. (If the WRS test were used, or background subtraction were used in conjunction with the Sign test, background concentrations would also be converted to Unity Rule Equivalents prior to performing test.)

The Sign test will be used without background subtraction if background Cs-137 is not considered a significant fraction of the DCGL. Note that the surrogate Cs-137 DCGL will be used for both the statistical tests and comparisons with the criteria in LTP Tables 5-13 and 5-14.

The same general surrogate and unity rule methods described above for soil will be applied to other materials, such as activated concrete, where sample gamma spectroscopy is used for final survey as opposed to gross beta measurements.

5.6.2.2 Elevated Measurement Comparison Evaluations

During final surveys, areas of elevated activity (hot spots) may be detected and they must be evaluated both individually and in total to ensure compliance with the release criteria. The hot spots are each compared to the specific $DCGL_{EMC}$ value calculated for the size of the specific hot spot. If the individual hot spots pass, then they are combined and evaluated under the unity rule.

The average activity of each hot spot is determined as well as the average value for the survey unit. The survey unit average value is divided by the $DCGL_W$, the survey unit average value is subtracted from the hot spot average activity value and the result is divided by the hot spot $DCGL_{EMC}$. Each hot spot net average activity is evaluated against its $DCGL_{EMC}$. The fractions are summed and the result must be less than unity for the survey unit to pass. This is summarized in Equation 5-14 below.

$$\frac{\delta}{DCGL_W} + \frac{\tau_1 - \delta}{DCGL_{EMC_1}} + \frac{\tau_2 - \delta}{DCGL_{EMC_2}} + \bullet\bullet\bullet + \frac{\tau_n - \delta}{DCGL_{EMC_n}} < 1$$

Equation 5-14

where:

δ = the survey unit average activity,

$DCGL_W$ = the survey unit DCGL concentration,

τ_n = the average activity value of hot spot *n*, and

$DCGL_{EMC_n}$ = the $DCGL_{EMC}$ concentration of hot spot *n*.

5.6.3 Data Conclusions

The results of the statistical tests, including application of the EMC, allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release dose criterion. The data provide statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the release criterion. The decision to release the survey unit is made with sufficient confidence and without further analysis.

The second conclusion that can be made is that the survey unit fails to meet the release criterion. The data are not conclusive in showing that the residual radioactivity is less than the release criterion. The data are analyzed further to determine the reason for the failure.

Possible reasons are that:

- The average residual radioactivity exceeds the $DCGL_W$,
- The average residual radioactivity is less than the $DCGL_W$; however, the survey unit fails elevated measurement comparison,
- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release, or
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

The power of the statistical test is a function of the number of measurements made and the standard deviation in measurement data. The power is determined from $1-\beta$ where β is the value for Type II errors. A retrospective power analysis may be performed using the methods described in Appendices I.9 and I.10 of MARSSIM. If the power of the test is insufficient due to the number of measurements, additional samples may be collected as directed by procedure. A greater number of measurements increases the probability of passing if the survey unit actually meets the release criterion.

If failure was due to the presence of residual radioactivity in excess of the release criterion, the survey unit shall be remediated. Survey unit failure due to inadequate design or implementation shall require investigation and re-initiation of the FSS process.

5.6.4 Compliance

The FSS is designed to demonstrate licensed radioactive materials have been removed from Rancho Seco property to the extent that remaining residual radioactivity is below the radiological criteria for unrestricted release. The site-specific radiological criteria presented in this plan demonstrate compliance with the criteria of 10 CFR 20.1402. If the measurement results pass the requirements of Table 5-7 and the elevated areas evaluated per Section 5.3.6.3 pass the elevated measurement comparison, the survey unit is suitable for unrestricted release. If survey measurements do not meet the criteria specified in Table 5-7, an investigation will be performed. Investigations will include an evaluation of survey design, instrumentation use and calculations, as necessary. All investigations of this nature will be documented using the corrective action process as discussed in Section 5.8.2.

5.7 Reporting Format

Survey results and a brief operating history are documented in survey unit release records and in the FSS Report. Other reports may be generated as requested by the NRC.

5.7.1 Operating History

A brief operational history including relevant operational and decommissioning data is compiled. The purpose of the history information is to provide additional, substantive data which forms a portion of the basis for the survey unit classification, and hence, the level of intensity of the FSS. The history information includes the following items:

- Operating history which could affect radiological status,
- Summarized scoping and site characterization data, and
- Other relevant information, as deemed necessary.

5.7.2 Survey Unit Release Record

A separate release record is prepared for each survey unit. The survey unit release record is a stand-alone document containing the information necessary to demonstrate compliance with the site release criteria. This record includes:

- Description of the survey unit,
- Survey unit design information,
- Survey unit measurement locations and corresponding data,
- Survey unit investigations performed and their results, and
- Survey unit data assessment results.

When a survey unit release record is given final approval it becomes a quality record.

5.7.3 Final Status Survey Report

Survey results will be described in a written report to the NRC. The actual structures, land, or piping system included in each written report may vary depending on the status of ongoing decommissioning activities.

The final status survey report provides a summary of the survey results and the overall conclusions that demonstrate that the Rancho Seco facility and site meet the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results are included in the report. The level of detail is sufficient to clearly describe the final status survey program and to certify the results. The format of the final report will contain the following topics:

- Overview of the Results;
- Discussion of Changes to FSS;
- Final Status Survey Methodology;
    - Survey unit sample size,
    - Justification for sample size;
- Final Status Survey Results;
    - Number of measurements taken,
    - Survey maps,
    - Sample concentrations,
    - Statistical evaluations,
    - Judgmental and miscellaneous data sets;
- Anomalous Data;
- Conclusion for each survey unit; and
- Any Changes from initial assumptions on extent of residual activity.

5.7.4 Other Reports

Other reports relating to final status survey activities may be prepared and submitted as necessary.

5.8 Final Status Survey Quality Program

Quality is built in to each phase of the FSS Program and measures must be taken during the execution of the plan to determine whether the expected level of quality is being achieved. The FSS Program will ensure that the site will be surveyed, evaluated and determined to be acceptable for unrestricted release if the residual activity results in an annual Total Effective Dose Equivalent (TEDE) to the average member of the critical group of 25 mrem/year or less for all pathways. The following sections provide a description of applicable Rancho Seco quality programs and specific quality elements of the FSS Program.

5.8.1 Rancho Seco Quality Assurance Program

The Rancho Seco Quality Assurance Program (QAP) [Reference 5-14] is applied to systems, structures, components and activities important to the safe storage, control and maintenance of spent nuclear fuel and to the monitoring and control of radiological hazards. The Rancho Seco Quality Manual (RSQM) defines the responsibilities and requirements to ensure decommissioning and operation of the ISFSI comply with licenses and applicable regulations (10 CFR 50 and 10 CFR 72). The RSQM addresses organizational responsibilities, staff qualifications, procedure review and approval, design and modification controls, procurement, measurement and test equipment (M&TE) calibration and control, testing of installed equipment, document control, corrective action and other information pertinent to quality.

5.8.2 FSS Quality Assurance Project Plan (QAPP)

The objective of the FSS QAPP is to ensure the survey data collected are of the type and quality needed to demonstrate with sufficient confidence the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. The plan ensures the following items are accomplished:

1) The elements of the final status survey plan are implemented in accordance with the approved procedures,
2) Surveys are conducted by trained personnel using calibrated instrumentation,
3) The quality of the data collected is adequate,
4) All phases of package design and survey are properly reviewed, with QC and management oversight provided, and
5) Corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

The following sections describe the basic elements of the FSS QAPP.

5.8.2.1 Project Management and Organization

An FSS organization will be established for the Rancho Seco site in RSAP-1901. This organization will be responsible for planning and implementation of final status surveys. Since the FSS organization has not been fully implemented at the time of LTP development, specific job titles may vary over the period of project execution. However, the following descriptions refer to various functional areas of responsibility and do not necessarily correspond to specific job titles. It is also important to note qualified individuals may assume the responsibilities of more than one of the functional positions described below. The FSS organization consists of the following functional areas:

5.8.2.1.1 Dismantlement Superintendent - Radiological

The Dismantlement Superintendent – Radiological has overall responsibility for program direction, technical content, and ensuring the program complies with applicable NRC regulations and guidance. This supervisor is responsible for preparation and implementation of the FSS procedures. Additional responsibility areas may include resolution of issues or concerns raised by the NRC or other Stakeholders, as well as programmatic issues raised by Rancho Seco site management. The Dismantlement Superintendent – Radiological provides overall FSS project coordination, which may include, but is not limited to, interfaces with site personnel in areas of nuclear licensing, demolition and waste disposal.

5.8.2.1.2 Final Status Survey Technical Specialists

Responsibilities of FSS Technical Specialists may include technical support and development of FSS procedures, design of final status surveys, preparation of survey execution instructions, development of specific technical analysis documents supporting FSS activities, and review of survey packages and data collected in support of the FSS.

5.8.2.1.3 Work Planning Coordinators

Work Planning Coordinators develop detailed, job-specific work instructions using the site work order process. These individuals are tasked with ensuring the appropriate interface between various site functional groups is specified in work order documents. These individuals possess specific knowledge regarding Radiation Protection, FSS, and Industrial Safety requirements.

5.8.2.1.4 Final Status Survey Field Coordinators

Final Status Survey Field Coordinators are responsible for control and implementation of survey packages during field activities. Specific responsibilities are likely to include:

- Coordination of turnover surveys,
- Survey area preparation (e.g., gridding),
- Ensuring final status survey sampling is conducted in accordance with applicable procedures and work instructions,
- Maintaining access controls over completed FSS survey areas,
- Determining survey area accessibility requirements,
- Coordination and scheduling of FSS Technicians to support the decommissioning schedule, and
- Ensuring all necessary instrumentation and other equipment is available to support survey activities.

5.8.2.1.5 Final Status Survey Data Specialist

The FSS Data Specialist is responsible for maintaining the FSS data records in both electronic formats and hardcopy files, as applicable. This includes maintaining survey measurement data and supporting data files and generating reports of survey results. Responsibilities also include maintaining the integrity of the FSS database and implementing FSS Database QA requirements.

5.8.2.1.6 Final Status Survey Technician

Final Status Survey Technicians are responsible for performance of final status survey measurements and collection of final status survey samples in accordance with applicable site procedures and survey package instructions. An FSS Technician will be responsible for maintaining the pedigree of instrumentation used in the survey by implementing the procedural requirements for calibration, maintenance and daily checks. Final Status Survey Technicians will be trained and task-qualified for the performance of the final status activities assigned to them. Final Status Survey Technicians may also participate in survey area preparations.

5.8.2.2 Written Procedures

Sampling and survey tasks must be performed properly and consistently in order to assure the quality of final status survey results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for final status survey measurements. Those procedures have been cited in Section 5.9.1.

5.8.2.3 Training and Qualification

Personnel performing final status survey measurements will be trained and qualified. Training will include the following topics:

- Procedures governing the conduct of the FSS,
- Operation of field and laboratory instrumentation used in the FSS, and
- Collection of final status survey measurements and samples.

Qualification is obtained upon satisfactory demonstration of proficiency in implementation of procedural requirements. The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity required to be performed by that individual. Records of training and qualification will be maintained in accordance with approved training procedures

5.8.2.4 Measurement and Data Acquisitions

The FSS records have been designated as quality documents and will be governed by site quality programs and procedures. Generation, handling and storage of the original final status survey design and data packages will be controlled by site procedures. Each final status survey measurement will be identified by individual, date, instrument, location, type of measurement, and mode of operation.

5.8.2.4.1 Quality Control Surveys

Procedures establish built-in Quality Control checks in the survey process for both field and laboratory measurements, as described in LTP Section 5.8.2.2. For structures and systems, QC replicate scan measurements will consist of resurveys of a minimum of 5% of randomly selected class 1, 2, or 3 survey units typically performed by a different technician with results compared to the original survey result. The acceptance criterion shall be that the same conclusion as the original survey was reached based on the repeat scan. If the acceptance criterion is not met, an investigation will be conducted to determine the cause and corrective action.

Quality Control for direct surface contamination and/or exposure rate measurements will consist of repeat measurements of a minimum of 5% of the survey units using the same instrument type, taken by a different technician (except in cases where there is only one instrument or specialized training is required to operate the equipment) and the results compared to the original measurements using the same instrument type. The acceptance criterion for direct measurements is specified in approved procedures.

For soil, water and sediment samples, Quality Control will consist of participation in the laboratory Inter-comparison Program. However, as an additional quality measure, approximately 5% of such samples may be subjected to blind duplicate samples and/or third party analyses. The acceptance criterion for blank samples is that no plant-derived radionuclides are detected. The criterion for blind duplicates is that the two measurements are within the value specified by approved procedure. For third party analyses, the acceptance criterion is the same as those for blind duplicates. Some sample media, such as asphalt, will not be subjected to split or blind duplicate analyses due to the lack of homogeneity. These samples will simply be recounted to determine if the two counts are within 20% of each other, when necessary.

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be performed in accordance with approved procedures; and if necessary, the Corrective Action Process described in Section 5.8.3.3 will be implemented. The investigation will typically involve verification that the proper data sets were compared, the relevant instruments were operating properly and the survey/sample points were properly identified and located. Relevant personnel are interviewed, as appropriate, to determine if proper instructions and procedures were followed and proper measurement and handling techniques were used including chain of custody, where applicable. When deemed appropriate, additional measurements are taken. Following the investigation, a documented determination is made regarding the usability of the survey data and if the impact of the discrepancy adversely affects the decision on the radiological status of the survey unit.

5.8.2.4.2 Instrumentation Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as specified in Section 5.4.3.4 as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the Instrumentation Program procedures.

5.8.2.5 Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the final survey results is established by procedure. When custody is transferred outside of the organization, a chain of custody form will accompany the sample for tracking purposes. Secure storage will be provided for archived samples.

5.8.2.6 Control of Consumables

In order to ensure the quality of data obtained from FSS surveys and samples, new sample containers will be used for each sample taken. Tools used to collect samples will be cleaned to remove contamination prior to taking additional samples. Tools will be decontaminated after each sample collection and surveyed for contamination.

5.8.2.7 Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, will be procured from appropriate vendors in accordance with approved quality and procurement procedures.

5.8.2.8 Database Control

Software used for data reduction, storage or evaluation will be fully documented and certified by the vendor. The software will be tested prior to use by an appropriate test data set.

5.8.2.9 Data Management

Survey data control from the time of collection through evaluation is specified by procedure. Manual data entries will be secondarily verified.

5.8.3 Assessment and Oversight

5.8.3.1 Assessments

FSS self-assessments will be conducted in accordance with approved procedures. The findings will be tracked and trended in accordance with these procedures. In addition, QC will perform assessments of FSS activities in accordance with the Quality Assurance Program.

5.8.3.2 Independent Review of Survey Results

Randomly selected survey packages (approximately 5%) from survey units will be independently reviewed by the Quality Assurance personnel to ensure that the survey measurements have been taken and documented in accordance with approved procedures.

5.8.3.3 Corrective Action Process

The corrective action process, already established as part of the site's 10 CFR Part 50 Appendix B Quality Assurance Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. The process will be conducted in accordance with approved procedures which describe the methods used to initiate potential deviation from quality (PDQ) reports and resolve self assessment and corrective action issues related to FSS. The PDQ evaluation effort is commensurate with the classification of the PDQ and could include root cause determination, extent of condition reviews, and preventive and remedial actions.

5.8.3.4 Reports to Management

Reports of audits and trend data will be reported to management in accordance with approved procedures.

5.8.4 Data Validation and Verification

Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels will be made and measurements exceeding the investigation levels will be evaluated. Procedurally verified data will be subjected to the Sign test, the Unity Sign test, the WRS test, or WRS Unity test as appropriate. Technical evaluations or calculations used to support the development of DCGLs will be independently verified to ensure correctness of the method and the quality of data.

5.8.5 Confirmatory Measurements

The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the FSS and associated documentation demonstrate the site is suitable for release in accordance with the criteria for decommissioning in 10 CFR Part 20, subpart E. Confirmatory measurements may include collecting radiological measurements for the purpose of confirming and verifying compliance with NRC standards for unrestricted license termination. Timely and frequent communications with the NRC will ensure it is afforded sufficient opportunity for these confirmatory measurements prior to implementing any irreversible decommissioning actions.

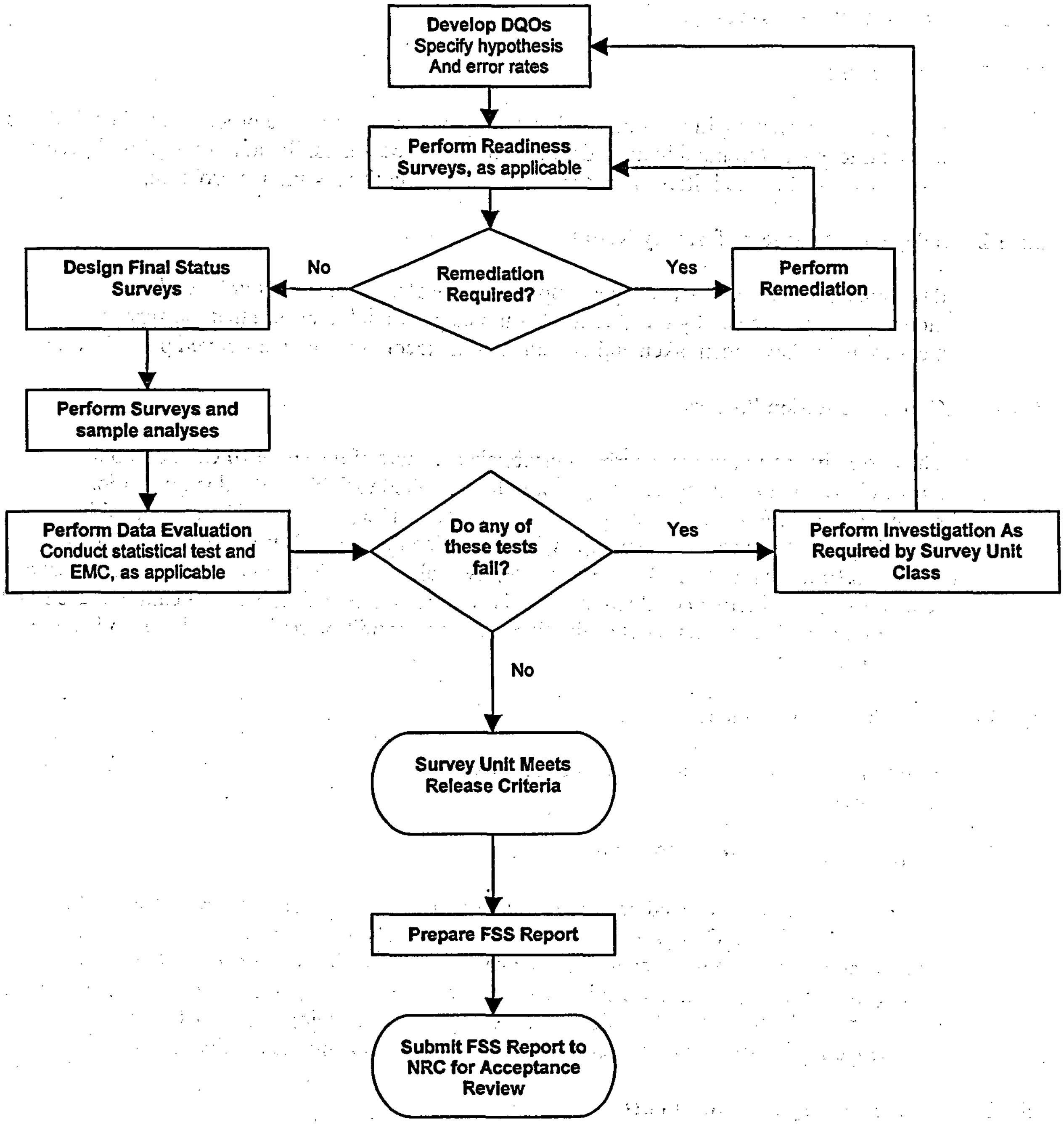
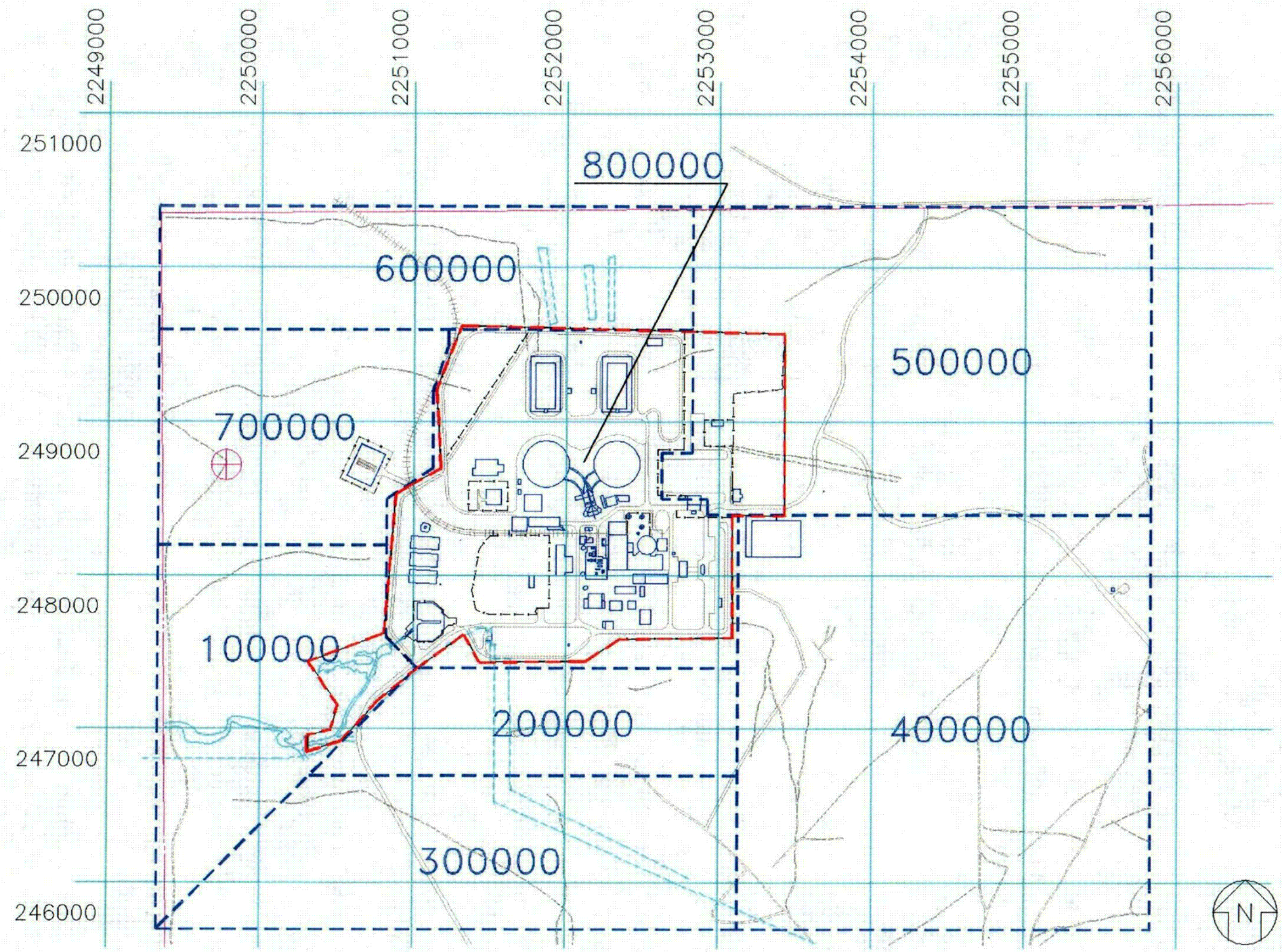


Figure 5-1

FSS Process Overview



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UTILITY DISTRICT

Figure 5-2
Area Designations

DWG: Figure 5-2

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5.9 References

5-1 U.S. Nuclear Regulatory Commission NUREG–1757, Vol. 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Final Report," September 2003

5-2 U.S. Nuclear Regulatory Commission NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," August 2000

5-3 U.S. Nuclear Regulatory Commission NUREG-1505, Revision 1, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," June 1998 draft

5-4 U.S. Nuclear Regulatory Commission NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," June 1998

5-5 U.S. Nuclear Regulatory Commission NUREG-1700, Revision 1, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," April 2003

5-6 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999

5-7 U.S. Nuclear Regulatory Commission NUREG/CR-5512, Volume 1, Final Report, "Residual Radioactive Contamination from Decommissioning," October 1992

5-8 Rancho Seco Decommissioning Technical Basis Document DTBD-06-002, Revision 0, "Use of a Survey Unit Size of 319 m^2 for Class One Structure Surveys at Rancho Seco Nuclear Generating Station"

5-9 Rancho Seco Decommissioning Technical Basis Document DTBD-06-001, Revision 0, "RSNGS Initial Classification of Survey Areas and Survey Design Sigma Values"

5-10 Rancho Seco Decommissioning Technical Basis Document DTBD-06-003, Revision 0, "Use of *In Situ* Gamma Spectroscopy for Final Status Surveys"

5-11 Rancho Seco Decommissioning Technical Basis Document DTBD-05-010, Revision 0, "Beta Detection Including Beta Energy and Source Efficiency"

5-12 International Organization for Standardization, ISO 7503-1, "Evaluation of Surface Contamination - Part 1: Beta Emitters and Alpha Emitters (first edition)," 1988

5-13 Rancho Seco Decommissioning Technical Basis Document DTBD 05-012, Revision 0, "Eberline SPA-3 and Ludlum 44-10 Detector Sensitivity (MDC)"

5-14 Sacramento Municipal Utility District, Rancho Seco Quality Manual

5.9.1 Applicable Site Procedures For FSS

DSIP-0020	Survey Unit Design
DSIP-0030	Survey Unit Remediation and ALARA Evaluation

DSIP-0050	Survey Unit Release Controls
DSIP-0060	Department Training and Qualification
DSIP-0100	Final Status Survey Design
DSIP-0110	Final Survey Performance
DSIP-0120	Decommissioning Survey Analysis (DQA)
DSIP-0130	Decommissioning Survey Background Reference Area Evaluations
DSIP-0200	DEG QC Program
DSIP-0210	QC Verification Survey Design
DSIP-0220	QC Verification Survey Performance
DSIP-0300	Decommissioning Sampling and Controls-Solids
DSIP-0500	Instrumentation
DSIP-0510	Ludlum 2350 Operation
DSIP-0520	Ludlum 2350 Download Procedure
QAIP-0601	Procedure Control
QAIP-1702	Records Management
RP.305	Radiation Protection Plan
RP.305.11	Radioactive Source Handling
RP.305.22	Departmental Training & Qualification
RP.311	Radiation Protection Instrument Control Program
RP.311.II.03	Ludlum 2350-1 Datalogger Calibration
RP.311.VI.01	RP Counting Statistics
RP.311.VI.02	Radiation Protection Instrument Checks
RP.311.VI.03	Counting Equipment Voltage Plateaus
RP.311.VII.02	Shepard Model 89 Cesium Calibrator
RP.311.VII.02	Verification/Certification of Gamma Calibrators
RSAP-0101	Nuclear Organization Responsibilities & Authorities
RSAP-0309	Vendor Documentation Requirements, Review, and Approval
RSAP-0409	Procurement Program for Defueled Plant
RSAP-0500	Review, Approval and Changes of Procedures
RSAP-0505	Document Control
RSAP-0506	Procedure Review
RSAP-0601	Nuclear Records Management
RSAP-1101	ALARA Manual
RSAP-1204	Training Programs
RSAP-1301	Corrective Action Program

RSAP-1305	Corrective Action Request
RSAP-1308	Potential Deviation from Quality
RSAP-1310	Deviation from Quality
RSAP-1501	Controlled Software Change Request
RSAP-1702	Quality Assurance for Radiological Monitoring Programs
RSAP-1900	Control of Decommissioning Projects
RSAP-1901	Decommissioning Survey Program