Commonwealth Edison Company

Dresden Nuclear Power Station, Unit 1

Radiological and Thermal Characteristics of the Dresden Unit 1 Fuel Storage Pool with Lower Than Normal Water Levels

Prepared By

SARGENT & LUNDY

Preliminary Report

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PRELIMINARY REPORT

SL-4904

Sargent & Lundy

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EXECUTIVE SUMMARY

On February 1, 1994, the NRC issued a Confirmatory Action Letter requiring Commonwealth Edison Company to take specific actions regarding the potential freezing and rupturing of piping in the Unit 1 containment at Dresden Nuclear Power Station. One requirement was to evaluate the potential consequences of a pipe break that results in a lowering of the water in the fuel storage pool, including an estimate of expected doses to occupational workers and members of the public. This report is a formal analysis of that requirement.

The first step in the analysis is characterization of the radioactive material (i.e., the spent reactor fuel) in the Fuel Building. There are 683 irradiated fuel assemblies in the Fuel Building, 534 of which are in channels, and one rod basket containing two complete fuel rods and one 16" section of a rod. Although 23 of the assemblies are currently in the fuel transfer pool, all assemblies were assumed to be in the storage pool. Since the storage pool is at a higher elevation than the transfer pool, this assumption produces a conservative radiological and heat up analysis that is bounding for any arrangement of the fuel assemblies in the transfer or storage pools. The fission product inventory used to calculate heating and radiological dose rates was calculated from the burnup and decay history of each fuel assembly.

The second step in the analysis is the determination of the temperature and evaporation rate of the storage pool water, and the expected temperatures of the cladding and at the fuel centerline. The average heat generation rate is 51.5 watts/assembly with a peak heat generation rate of 81.1 watts/assembly. The bulk temperature of the pool water remains below 120°F, and the rate at which the pool water level will drop at the maximum evaporation rate is less than 1/16 inch per hour. Under the worst case conditions (partially uncovered fuel), the peak temperature inside a fuel pin is less than 270°F, which is far below the temperature at which fuel cladding is expected to fail (570°C or 1058°F).

The final step in the analysis is the evaluation of the potential radiation exposures. The primary exposure pathway is the direct and scattered radiation from uncovered fuel assemblies. Inside the Fuel

Building dose rates will range from 5 rem/hr at the entrance to 800 rem/hr at the pool edge if the assemblies are completely uncovered. Inside the Unit 2/3 Control Room the dose rate is expected to be less than 1 mrem/hr, so uncovering the fuel will not affect the safe operation of Units 2 and 3. The fuel temperature following loss of water coverage is much lower than the vaporization temperature of volatile fission products, so no significant additional contamination is expected of existing defects.

The general public may be exposed to direct or scattered radiation from the fuel assemblies, or to activity in an effluent plume. The calculated dose rate due to scattered radiation at the closest point on the site boundary is less than 2 mrem/hr. Due to the extensive decay time since removal of the last of the fuel assemblies from the core, Kr-85 remains as the only source of potential airborne radioactivity. Although additional fuel cladding failure is not predicted as a result of the loss of water coverage, a previous analysis evaluated the consequences of releasing all of the Kr-85 by failing all the fuel rods. This previous analysis, which was submitted to the NRC by Commonwealth Edison on April 10, 1989, estimated a worst case exposure at the site boundary of 1.7 rads to the skin and 0.016 rads whole body in two hours. Therefore uncovering the fuel does not present a significant additional risk to the general public.

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Section 1
INTRODUCTION

1.1 PURPOSE

The Unit 1 Fuel Building at Dresden Nuclear Power Station is used to store spent fuel assemblies from the Unit 1 reactor. Under certain conditions, it is possible that the water in the pools containing this spent fuel may be accidentally drained to the point where the fuel assemblies are exposed. The purpose of this report is to summarize the analyses performed by Sargent & Lundy to assess certain radiological and thermal conditions that would result from uncovered fuel assemblies. The conditions assessed are:

Radiation Dose Rates The dose rates caused by the radioactive material in the fuel assemblies are calculated in the Fuel Building, the Unit 2/3 control room, and at the closest point on the site boundary.

Bulk Pool Temperature The temperature of the water in the pool containing the fuel assemblies and the resulting evaporation rate are calculated for normal and partially uncovered conditions.

Fuel Rod Temperature The temperature of the fuel rod cladding and at the center of the fuel are calculated for normal and uncovered conditions.

Note that this report does not address existing radiological conditions, such as surface contamination and activity in the pool water. Further this report does not address the radiological consequences of radioactivity released into liquid or airborne pathways. The potential airborne exposure from the fuel assemblies was previously analyzed by conservatively assuming all the gap activity of Kr-85 (30% of total inventory) in all the assemblies was released simultaneously to the environment [Reference 1]. The result was offsite exposures well within regulatory limits, so additional analysis is not required in this report.

1.2 BACKGROUND

Unit 1 of Dresden Nuclear Power Station operated from 1960 until October 31, 1978, when it was taken out of service for backfit and decontamination and never restarted. In July of 1986 the Unit 1 license was amended to a possess-but-not-operate status. Units 2 and 3 are located adjacent to Unit 1 and are licensed for full power operation. The layout of the Dresden Nuclear Power Station is shown in Figure 1.

On January 25, 1994 a large amount of water was discovered in the Unit 1 containment. This release of water resulted from a break in a service water pipe that had frozen during recent cold weather. Since the storage pools in the Unit 1 Fuel Building are connected to the Unit 1 containment by the fuel transfer tube and since the tube is metal and full of water, it was determined that the tube was susceptible to freezing and failure. This could result in a lowering of the water level in the fuel transfer pool and possibly in the fuel storage pool (if the gates are removed). The limiting break identified in containment would lower the water level in the storage pool far enough to uncover the top three feet of the active fuel region of the fuel assemblies. The NRC was notified on January 27, 1994, and on February 1, 1994 issued a confirmatory action letter [Reference 2] requiring, among other items, an estimate of the consequences of such a break, including an estimate of expected doses to occupational workers and members of the public. An initial estimate was required within 48 hours with a formal analysis within 30 days.

S&L was contacted by Commonwealth Edison personnel on February 2, 1994 with a request for assistance in responding to the NRC. Preliminary dose rates due to the uncovered fuel assemblies were provided on February 3, 1994 [Reference 3]. Subsequently, more detailed analyses of the dose rates have been performed that confirm the original estimates were conservative. Additionally, thermal hydraulic analyses requested by Commonwealth Edison have been performed to demonstrate the worst scenario has been analyzed.

Section 2
DESCRIPTION OF THE FUEL STORAGE POOL

2.1 FUEL BUILDING AND FUEL STORAGE POOL

The Fuel Building for Unit 1 is a rectangular building 92'-0" long, 57'-0" wide and 48'-0" high located immediately south of the Unit 1 containment and east of the Access Control Building. Except for the truck bay and entry way, the floor is a concrete slab on backfill 4'-3" above grade. The walls of the building are steel frame with siding, and the roof is made of steel decking with one inch of insulation.

The majority of the fuel assemblies are in the fuel storage pool, which is located in the North-East corner of the Fuel Building. The storage pool is 20'-0" wide and 29'-0" long and has a depth of 26'-5". The storage pool is connected to the transfer pool and tube by a four-foot wide gate along the north edge of the pool. The bottom of the transfer pool, which is 20'-0" wide and 25'-6" long, is 15'-7" lower than the storage pool. Under normal conditions, the water level is one foot below the Fuel Building floor, which provides approximately fifteen feet of water shielding above the active region of the fuel assemblies. Schematics of the Fuel Building and other structures in the vicinity are shown in Figures 2 and 3.

2.2 FUEL ASSEMBLIES IN THE FUEL STORAGE POOL

There are a total of 683 irradiated fuel assemblies in the Fuel Building, with 534 of the assemblies in channels. Currently 660 of the assemblies are in fuel racks in the storage pool and the remaining 23 are in baskets in the transfer pool. In this analysis, it is assumed that all assemblies are in the fuel storage pool so that this analysis bounds the situation where the remaining assemblies are moved into the storage pool. In addition to the fuel assemblies and fuel racks, there are two complete fuel rods and a sixteen inch length of fuel rod in a rod storage basket in the storage pool. These rods make a negligible contribution to the activity in the storage pool and are not specifically included in this report.

The assemblies in the Fuel Building are the last core off-load from End of Cycle 11 and the assemblies discharged at the end of the previous four cycles, so their last burn date ranges from

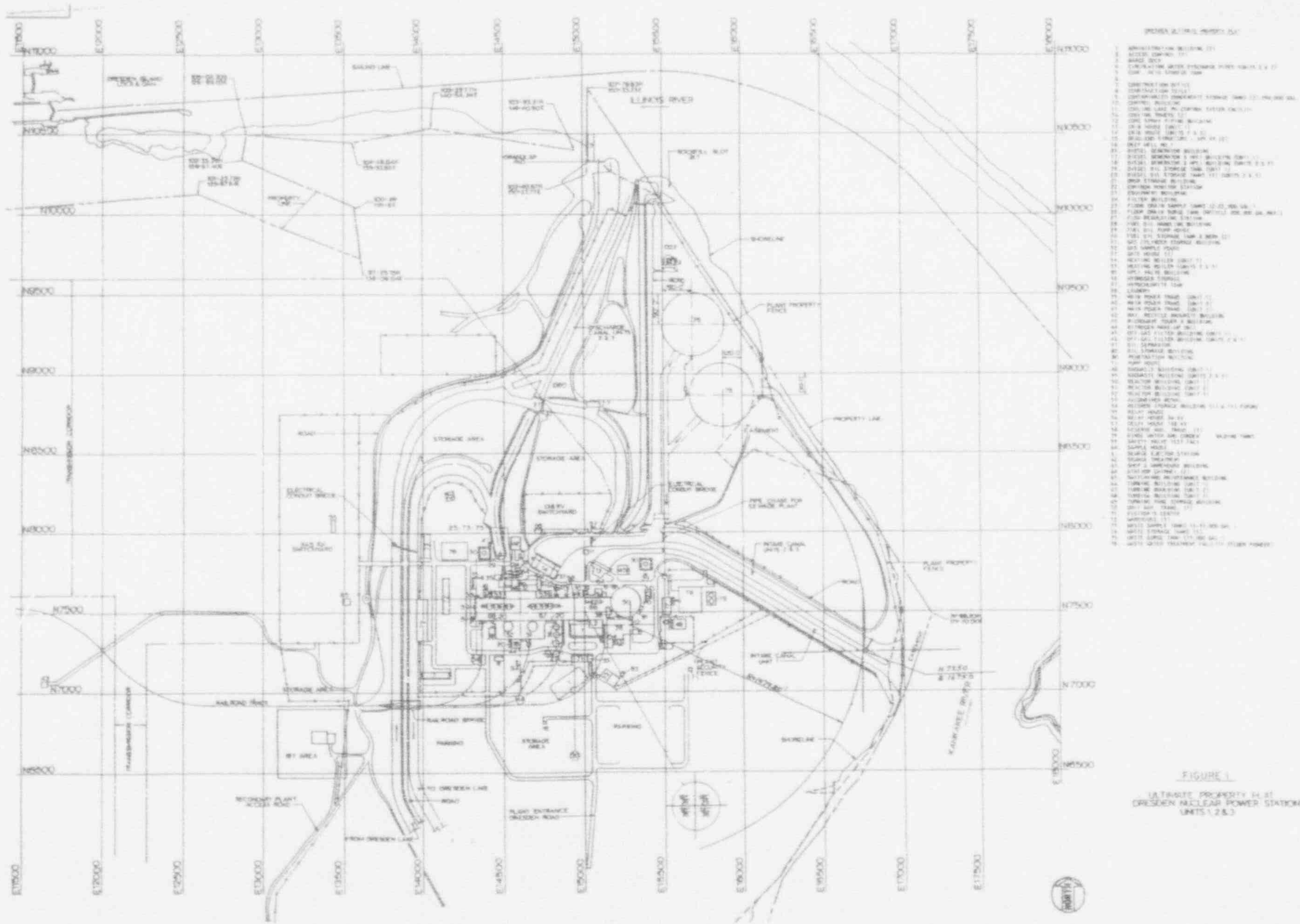


FIGURE 1

ULTIMATE PROPERTY PLAT
DRESDEN NUCLEAR POWER STATION
UNITS 1, 2 & 3

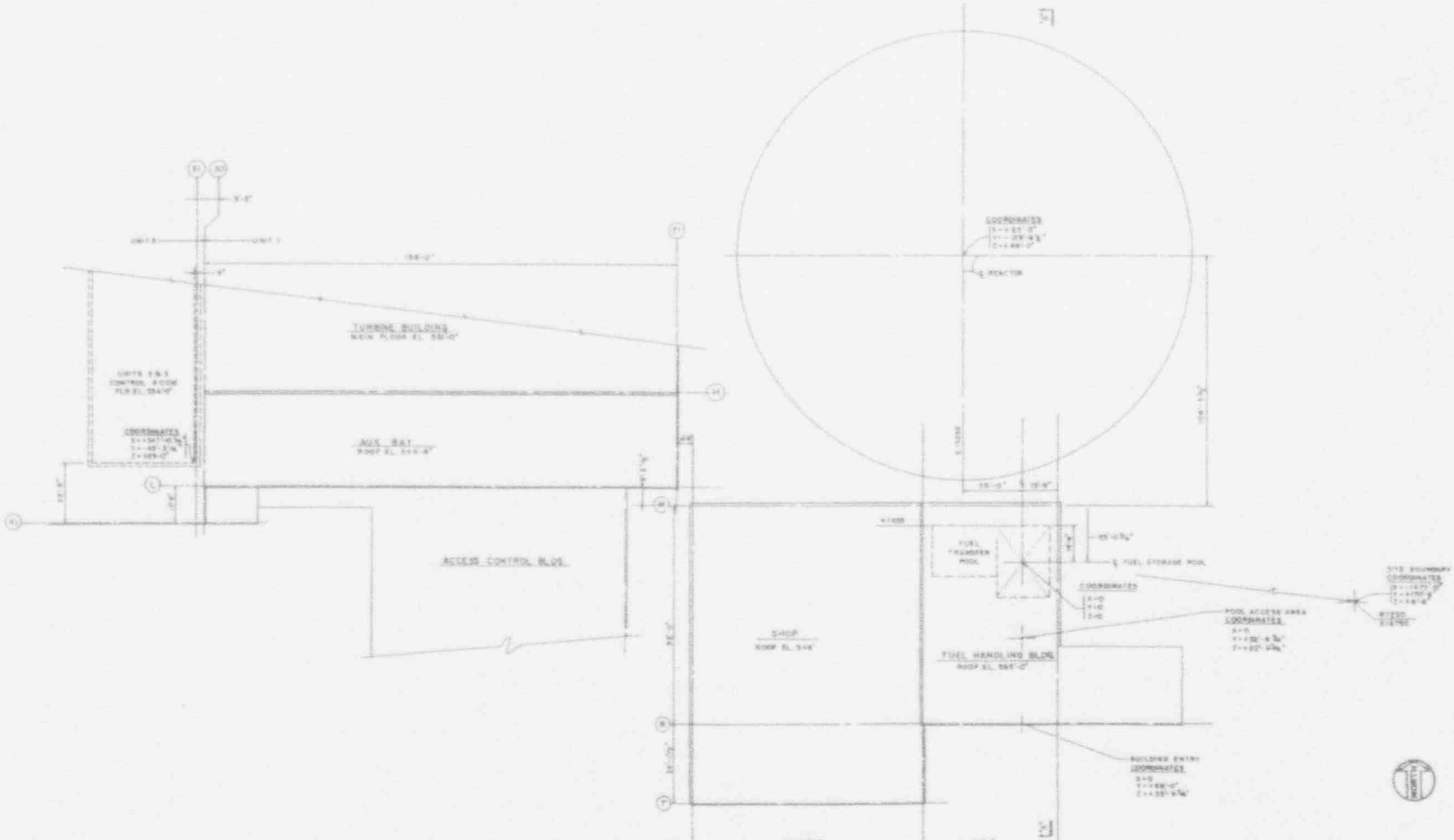


FIGURE 2

LAYOUT OF BUILDINGS IN VICINITY
OF FUEL HANDLING BUILDING
PLAN AT GRADE EL 517'-0"

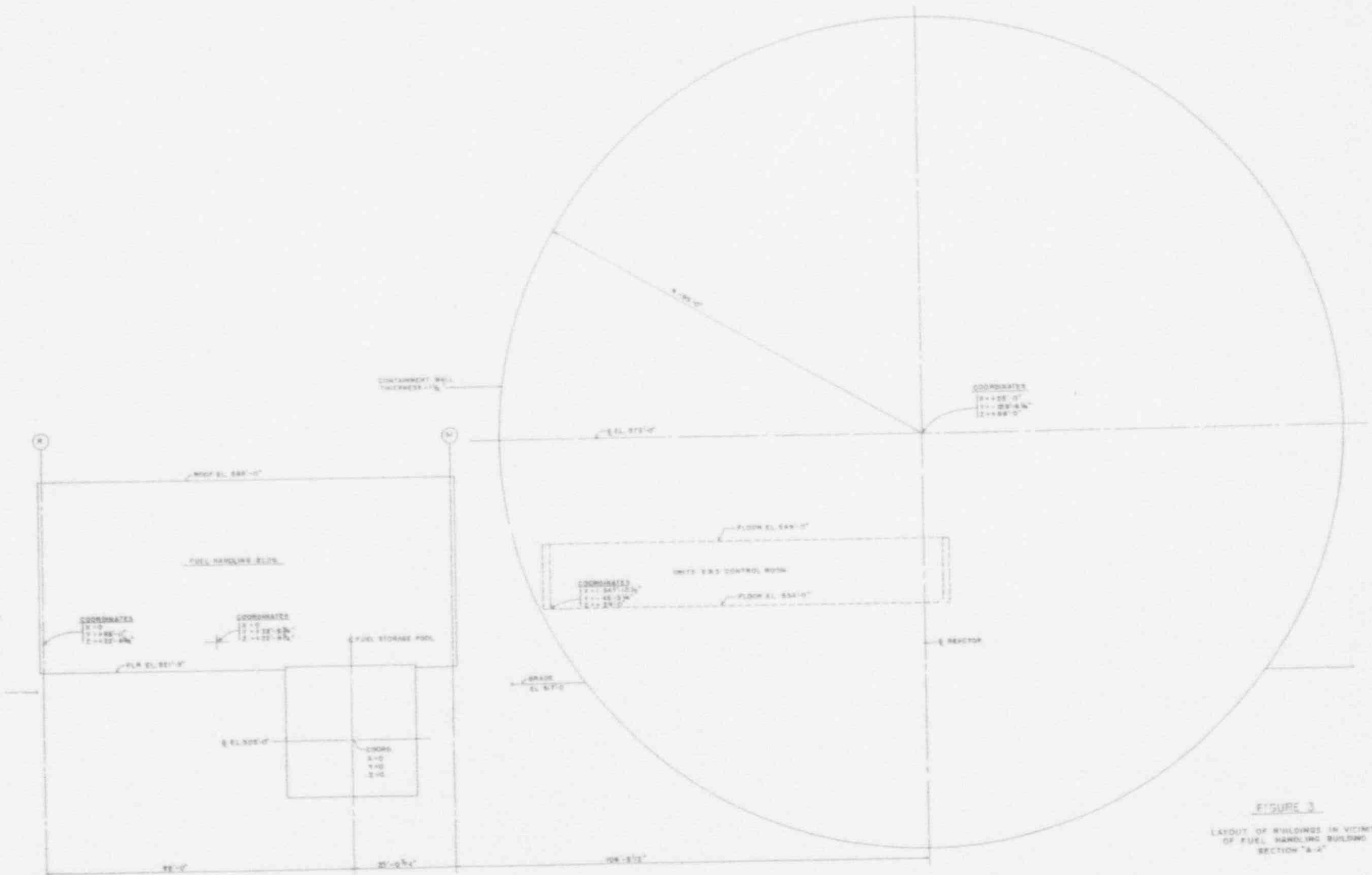
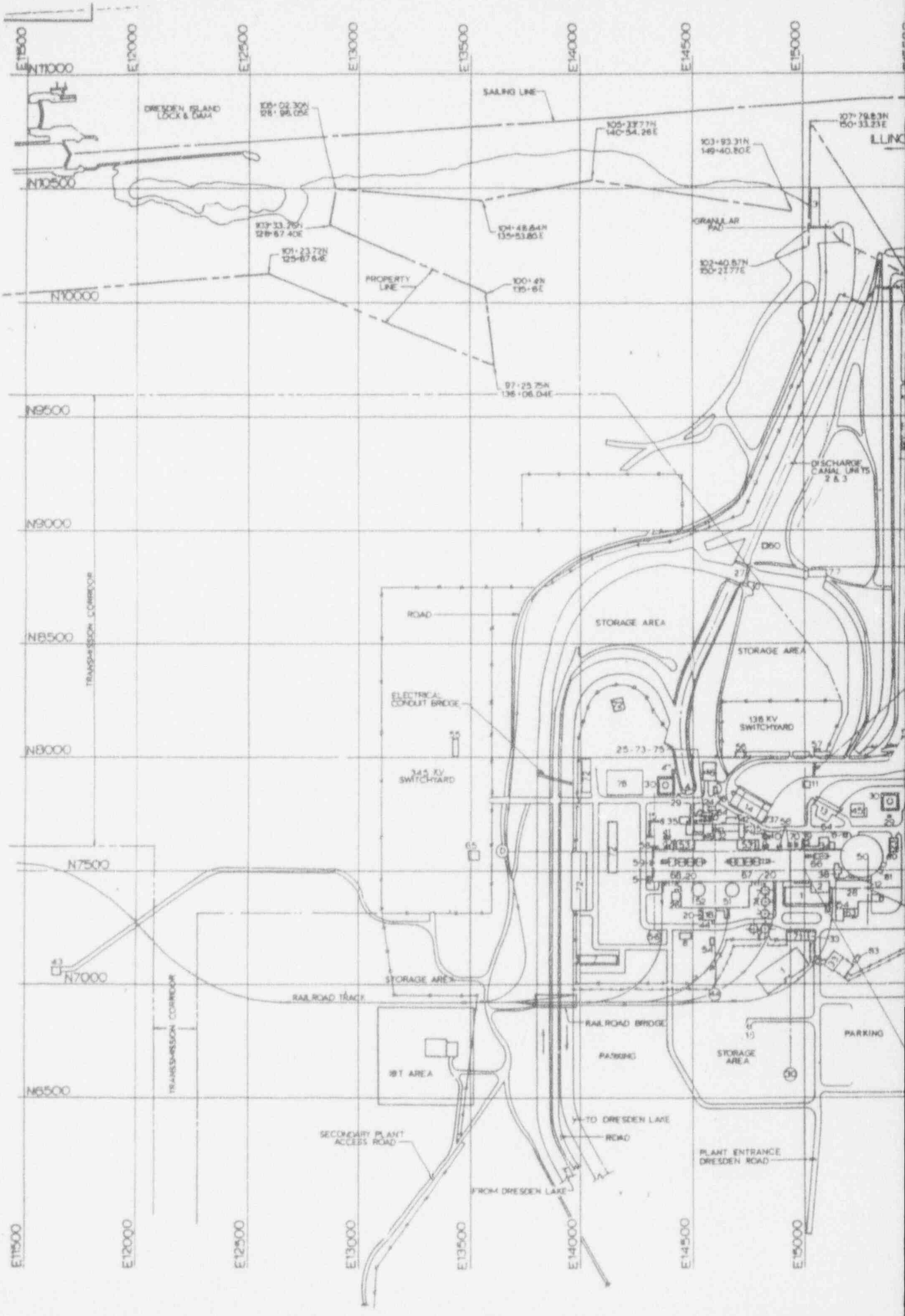


FIGURE 3

LAYOUT OF BUILDINGS IN VICINITY
OF FUEL HANDLING BUILDING
SECTION "A-A"



E11500
E12000
E12500
E13000
E13500
E14000
E14500
E15000
N11000
N10500
N10000
N9500
N9000
N8500
N8000
N7500
N7000
N6500
SAILING LINE
DRESDEN ISLAND LOCK & DAM
PROPERTY LINE
GRANULAR PAD
DISCHARGE CANAL UNITS 2 & 3
STORAGE AREA
STORAGE AREA
ROAD
ELECTRICAL CONDUIT BRIDGE
138 KV SWITCHYARD
345 KV SWITCHYARD
TRANSMISSION CORRIDOR
TRANSMISSION CORRIDOR
STORAGE AREA
RAILROAD TRACK
RAILROAD BRIDGE
PARKING
STORAGE AREA
PARKING
IBT AREA
SECONDARY PLANT ACCESS ROAD
TO DRESDEN LAKE
ROAD
FROM DRESDEN LAKE
PLANT ENTRANCE DRESDEN ROAD

DRESDEN ULTIMATE PROPERTY PLAT

1. ADMINISTRATION BUILDING (2)
2. ACCESS CONTROL (2)
3. BARGE DOCK
4. CIRCULATING WATER DISCHARGE PIPES (UNITS 2 & 3)
5. CONC. ACID STORAGE TANK
6.
7. CONSTRUCTION OFFICE
8. CONSTRUCTION TOILET
9. CONTAMINATED CONDENSATE STORAGE TANKS (2) 290,000 GAL.
10. CONTROL BUILDING
11. COOLING LAKE pH CONTROL SYSTEM FACILITY
76. COOLING TOWERS (2)
12. CORE SPRAY PIPING BUILDING
13. CRIB HOUSE (UNIT 1)
14. CRIB HOUSE (UNITS 2 & 3)
15. DEAD-END STRUCTURE - 345 KV (2)
16. DEEP WELL NO.1
83. DIESEL GENERATOR BUILDING
17. DIESEL GENERATOR & HPCI BUILDINGS (UNIT 1)
18. DIESEL GENERATOR & HPCI BUILDING (UNITS 2 & 3)
19. DIESEL OIL STORAGE TANK (UNIT 1)
20. DIESEL OIL STORAGE TANKS (3) (UNITS 2 & 3)
21. DRUM STORAGE BUILDING
22. ENVIRON MONITOR STATION
23. EQUIPMENT BUILDING
24. FILTER BUILDING
25. FLOOR DRAIN SAMPLE TANKS (2-22,000 GAL.)
26. FLOOR DRAIN SURGE TANK (RECYCLE 200,000 GAL.MAX.)
27. FLOW REGULATING STATION
28. FUEL OIL HANDLING BUILDING
29. FUEL OIL PUMP HOUSE
30. FUEL OIL STORAGE TANK & BERM (2)
31. GAS CYLINDER STORAGE BUILDING
32. GAS SAMPLE HOUSE
33. GATE HOUSE (3)
34. HEATING BOILER (UNIT 1)
35. HEATING BOILER-(UNITS 2 & 3)
81. HPCI VALVE BUILDING
36. HYDROGEN STORAGE
37. HYPOCHLORITE TANK
38. LAUNDRY
39. MAIN POWER TRANS. (UNIT 1)
40. MAIN POWER TRANS. (UNIT 2)
41. MAIN POWER TRANS. (UNIT 3)
42. MAX. RECYCLE RADWASTE BUILDING
43. MICROWAVE TOWER & BUILDING
44. NITROGEN MAKE-UP UNIT
45. OFF-GAS FILTER BUILDING (UNIT 1)
46. OFF-GAS FILTER BUILDING (UNITS 2 & 3)
47. OIL SEPARATOR
82. OIL STORAGE BUILDING
80. PENETRATION BUILDING
77. PUMP HOUSE
48. RADWASTE BUILDING (UNIT 1)
49. RADWASTE BUILDING (UNITS 2 & 3)
50. REACTOR BUILDING (UNIT 1)
51. REACTOR BUILDING (UNIT 2)
52. REACTOR BUILDING (UNIT 3)
53. RECOMBINER ROOMS
54. RECORDS STORAGE BUILDING (1) & (1) FUTURE
55. RELAY HOUSE
56. RELAY HOUSE 34 KV
57. RELAY HOUSE 138 KV
58. RESERVE AUX. TRANS. (3)
79. RINSE WATER AND CONDENSATE HOLDING TANKS
59. SAFETY VALVE TEST FACILITY
60. SAMPLE HOUSE
61. SEWAGE EJECTOR STATION
62. SEWAGE TREATMENT
63. SHOP & WAREHOUSE BUILDING
64. STATION CHIMNEY (2)
65. SWITCHYARD MAINTENANCE BUILDING
66. TURBINE BUILDING (UNIT 1)
67. TURBINE BUILDING (UNIT 2)
68. TURBINE BUILDING (UNIT 3)
69. TURNING VANE STORAGE BUILDING
70. UNIT AUX. TRANS. (3)
71. VISITOR'S CENTER
72. WAREHOUSE (3)
73. WASTE SAMPLE TANKS (3-33,000 GAL.)
74. WASTE STORAGE TANKS (3)
75. WASTE SURGE TANK (77,000 GAL.)
78. WASTE WATER TREATMENT FACILITY (FLUOR PIONEER)

FIGURE I

ULTIMATE PROPERTY PLAT
DRESDEN NUCLEAR POWER STATION
UNITS 1, 2 & 3

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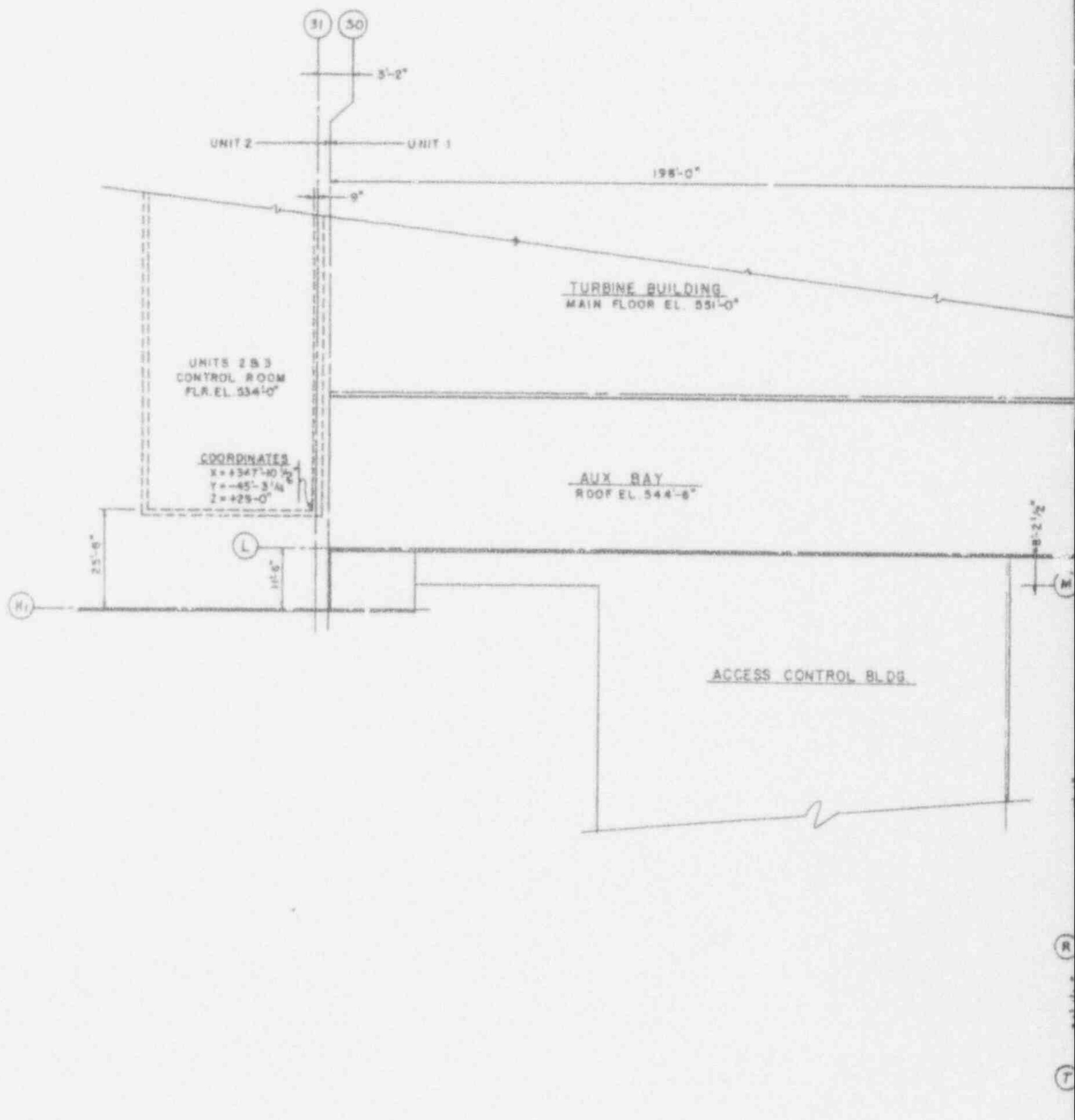

31
30
3'-2"
UNIT 2
UNIT 1
198'-0"
9"
TURBINE BUILDING
MAIN FLOOR EL. 551'-0"
UNITS 2 & 3
CONTROL ROOM
FLR. EL. 534'-0"
COORDINATES
X = +347'-10 1/2"
Y = -45'-3 1/4"
Z = +25-0"
AUX. BAY
ROOF EL. 544'-6"
8'-2 1/2"
L
25'-6"
11'-6"
H1
ACCESS CONTROL BLDG.
R
T

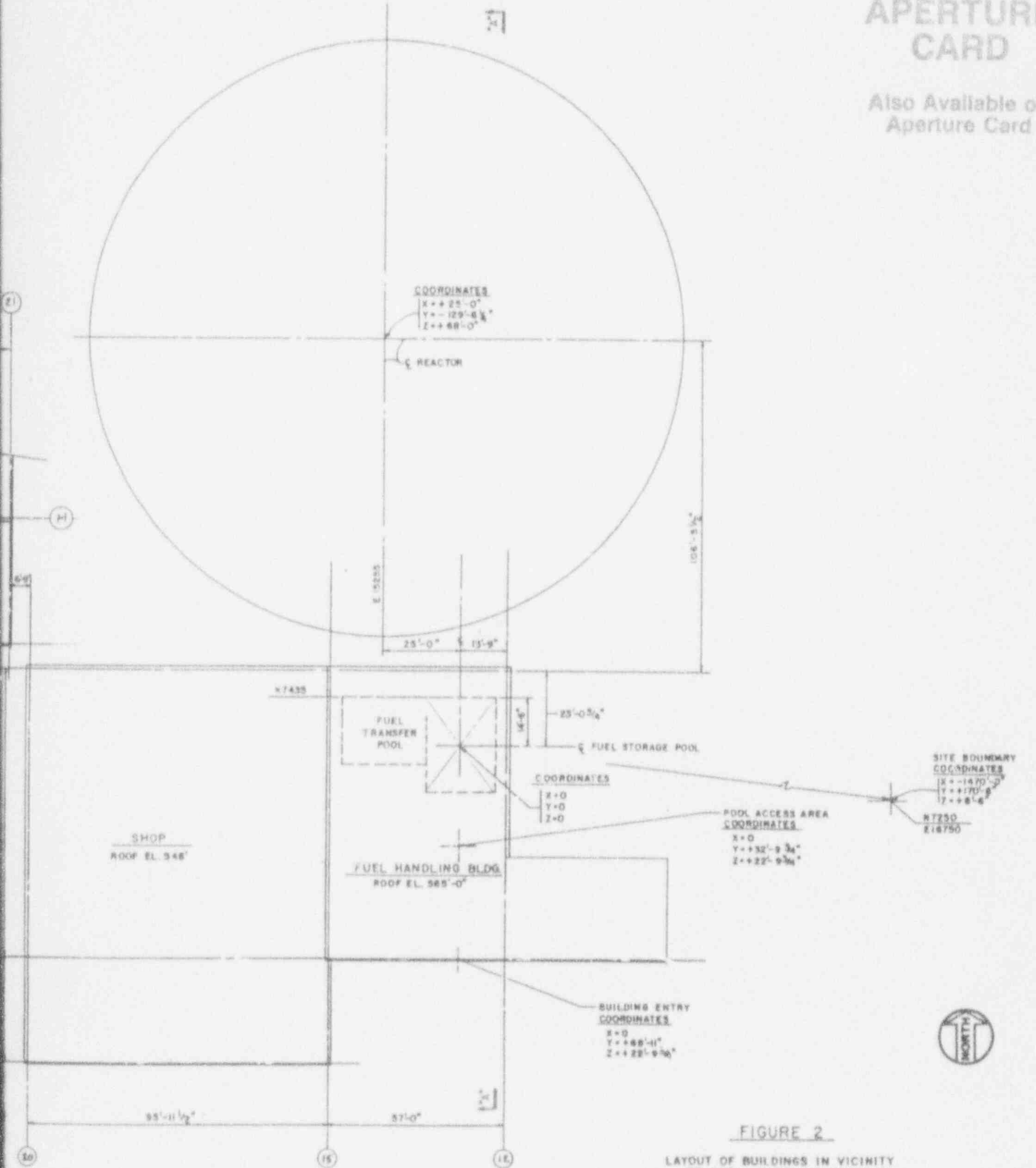

FIGURE 2

LAYOUT OF BUILDINGS IN VICINITY
OF FUEL HANDLING BUILDING
PLAN AT GRADE EL. 517'-0"

9403160186-02


CONTAINMENT WALL
THICKNESS = 1¼"
R
M
℄ EL. 573'-0"
ROOF EL. 565'-0"
FUEL HANDLING BLDG.
COORDINATES
X = 0
Y = +88'-11"
Z = +22'-9¾"
COORDINATES
X = 0
Y = +32'-9¾"
Z = +22'-9¾"
℄ FUEL STORAGE POOL
COORDINATES
X = +347'-10½"
Y = -45'-3⅛"
Z = +29'-0"
FLR. EL. 521'-3"
GRADE
EL. 517-0
℄ EL. 505'-0"
COORD.
X = 0
Y = 0
Z = 0
92'-0"
23'-0¾"
106'-5½"

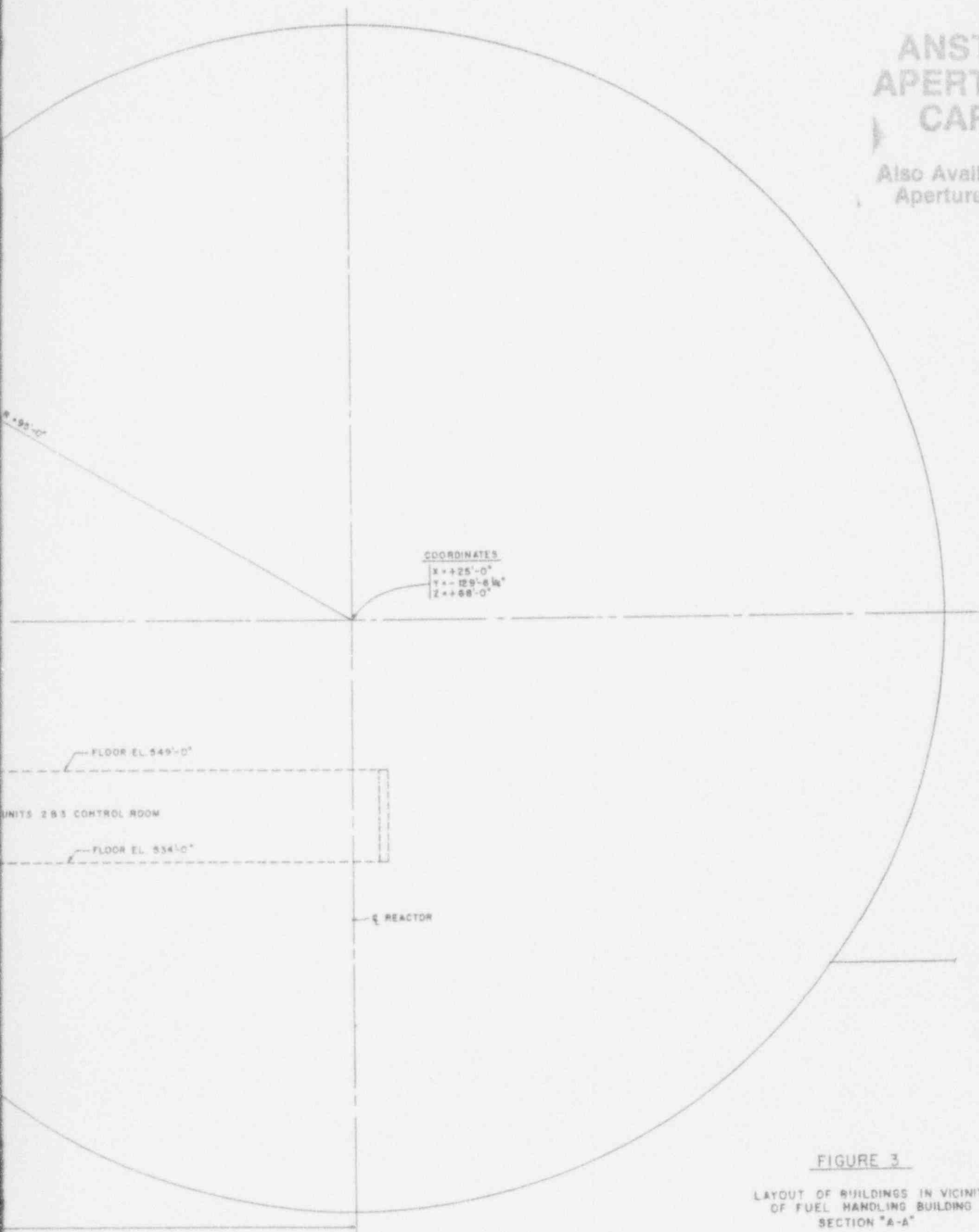

FIGURE 3

LAYOUT OF BUILDINGS IN VICINITY
OF FUEL HANDLING BUILDING
SECTION "A-A"

9403160186-03

October 1973 to October 1978. They range in burnup from 2,940 megawatt days per short ton of initial heavy metal (MWD/STIHM) for the newest fuel assemblies to a maximum of 25,527 MWD/STIHM, with an average burnup of 16,164 MWD/STIHM. They range in initial assembly average enrichment from 1.83% to 2.26% with an average of 2.22%.

2.3 CALCULATED THERMAL AND RADIOLOGICAL PROPERTIES OF THE FUEL ASSEMBLIES

The fuel assembly composition was determined by multiple runs of the computer code ORIGEN2. The fuel was modeled as uranium oxide. Less than 100 fuel rods contain mixed oxide fuel. Because of the small number of mixed oxide rods, modeling the fuel as uranium oxide will not have a significant effect on the results. The properties of interest are described below.

2.3.1 Radionuclide Inventories

While in the core, radionuclides are formed by fission, activation of structural materials, and multiple neutron absorptions in uranium that produce actinides and their daughters. Table 1 contains a listing of the twenty-two major nuclides in the fuel assemblies, which represent more than 99.9% of the activity, along with the total activity in the 683 fuel assemblies. The activity is dominated by the fission products Sr-90/Y-90 and Cs-137/Ba-137m, and the actinide Pu-241.

2.3.2 Gamma Radiation Source and Energy Spectrum

ORIGEN2 calculates an eighteen group energy spectrum for gamma rays produced by decay of the radionuclides in the fuel. The total photon production rate as a function of energy for all 683 assemblies is shown in Table 2. Note that the spectrum is dominated by the energy group with a mean energy of 0.575 MeV, which contains the characteristic decay energy (.67 MeV) of Cs-137/Ba-137m. The other major nuclides are primarily beta emitters and therefore contribute to the much lower energy groups.

2.3.3 Thermal Heat Generation Rate

ORIGEN2 also calculates the heat generated in the fuel due to radioactive decay. For the entire group of assemblies, the average heat generation rate is 51.5 watts per assembly. To identify the maximum heat generation rate, several high burnup assemblies were analyzed. The assembly with the largest heat generation rate was UN0024, which had an initial enrichment of 2.25% and was discharged in October 1978 with a cumulative burnup of 23,695 MWD/STIHM. The heat generation rate for this peak assembly is 81.1 watts per assembly.

Section 3
THERMAL BEHAVIOR OF THE FUEL STORAGE POOL

The initial concern following a loss of pool inventory event will be the time available to provide remedial action. This time will be determined by a number of factors: the post accident dose rate, the rate at which the fuel cladding heats up when exposed to air and the rate at which the pool heats up. This section summarizes the investigations into the factors that affect the thermal behavior of the pool. These include the heat up of the exposed fuel and the heat up of the pool which, includes an investigation of the rate of evaporation at the pool surface and the potential for boiling of the pool when the fuel is completely submerged at normal water level and when partially submerged at a lowered water level.

3.1 POOL WATER TEMPERATURE

The purpose of this investigation is to confirm that pool boiling will not occur following the postulated event that the pool level is reduced to El. 502'-0", which causes the top three feet of the active region of the fuel to be exposed in the fuel storage pool.

A thermal hydraulic model of the pool was constructed that performs an energy balance on the pool boundaries. The heat addition from the decay heat in the fuel is compared with the heat loss to the pool boundaries through convection, conduction and evaporation. The excess energy causes the pool temperature to rise.

The rate of heat addition from the fuel is 51.5 watts per assembly, as discussed in Section 2. The evaporation at the surface is determined using the "Carrier Equation" from Reference 4. Convection at the pool surface is determined from Reference 5. The remaining heat loss is by convection and conduction through the pool boundaries. The model is benchmarked for the single data point available, a steady-state pool temperature of 85°F with an air temperature of 10°F inside the building and -5°F outside the building. With this model and a design basis summer air temperature of 95°F, the predicted pool temperature is 119.5°F and the inside air temperature is 106°F.

The model is then used to estimate the effects of the rapid reduction of the pool water level to El. 502'-0". In this simulation, the pool temperature is initialized at the summer condition, the pool wall surface area is reduced to be consistent with the pool level and the heat added from the fuel is reduced so that only the submerged fuel is communicating with the pool.

The results of the simulation are:

- The pool temperature rises to 120°F at 500 hours,
- The predicted rate of evaporation at the pool surface is 70 pounds per hour.
- The rate at which the pool level is falling due to evaporation is one foot in 500 hours.

3.2 EVAPORATION RATE

A significant heat and mass transfer mechanism, evaporation, operates at the pool surface. This phenomenon provides a substantial portion of the cooling required by the pool to maintain a constant temperature for the pool water. Unfortunately, the penalty for this benefit is the reduction in the pool water inventory that results from the evaporation. As a means of assessing the significance of the cooling and mass transfer that takes place at the pool surface, a parametric study was performed to define the limits of the phenomena that take place at the pool surface.

The "Carrier Equation", Reference 4, was used to evaluate the heat and mass transport through the pool's surface. Ambient air conditions of 35°F, 50°F and 100°F were considered in an attempt to bound the range of temperatures within the fuel storage building. Commonwealth Edison personnel indicated that the only data relating to pool temperature was a recent measurement performed when the air temperature was approximately 10°F. The pool temperature was measured at 85°F. For sake of conservatism, this study investigated the effect of pool temperatures of 85°F, 100°F and 120°F on the evaporation rate. A significant contributor to the rate of evaporation is the velocity of air moving over the pool. The Fuel Building has a high volume air moving device located above the pool, which was assumed to be operating to maximize evaporation. A velocity of 50 feet per minute was determined to be adequate to account for the effects of air movement in the Fuel Building. The remaining parameter in the study was the relative humidity of the air. This was varied from 19% to 100% as a means of demonstrating the effect on the evaporation rate.

The maximum evaporation rate occurs at the maximum pool temperature (120°F), minimum air temperature (35°F) and maximum air velocity (50 fpm). For these conditions, rate of evaporation is approximately 400 pounds per hour. The corresponding rate of pool level reduction is approximately 1/16-inch per hour. The expected rates of evaporation in the winter and summer are 124.3 and 33.7 pounds per hour, respectively. This rate of evaporation corresponds to a rate of pool level decrease of less than 1 inch per day.

From this study, we conclude that the rate of loss of pool inventory due to evaporation is small and therefore, rapid responses after an accident are not required to provide additional pool inventory to offset the effects of evaporation at the pool's surface.

3.3 CALCULATED FUEL ASSEMBLY TEMPERATURES

3.3.1 Current Fuel Temperatures

The current fuel cladding and fuel centerline temperature were calculated using the S&L computer program FPT/L. The program evaluates the localized temperatures in the spent fuel pool based on natural convection induced flow through the fuel cells. Cross flow is ignored. The analysis was performed based on a bulk pool temperature of 85°F and resulted in a fuel cladding and fuel centerline temperature of 86.8°F and 86.9°F, respectively.

3.3.2 Fuel Temperature for Partially Covered Rod

A hand calculation was performed in order to compute the fuel temperatures for the condition with the top 3-foot section of the fuel rod exposed to air. The analysis determined the thermally induced air flow for an assumed ambient air temperature of 120°F. The incoming air temperature is assumed to be represented by the average of the ambient temperature and the temperature of the air exiting the flow channel between the fuel rods. The calculated fuel cladding and fuel centerline temperature for this case are 265.6°F and 265.7°F, respectively. The results are applicable for ambient air conditions ranging from 85°F to 120°F.

3.3.3 Fuel Temperature for Completely Uncovered Rod

The fuel cladding and fuel centerline temperatures were calculated using the S&L computer program FPT/L for the condition where the fuel pool is assumed to be completely drained. The program calculates the localized temperatures within the channels surrounding the fuel cells. Cross flow is ignored. The fuel is, therefore, only cooled by thermally induced air flow entering at the bottom plenum and exiting at the top of the fuel cell. The analysis was performed based on a bulk air temperature of 120°F entering the channel, and resulted in a fuel cladding and fuel centerline temperature of 225.0°F and 225.1°F, respectively.

3.3.4 Summary

The fuel centerline temperature is higher when the rod is partially uncovered and cooled by air rather than when the entire rod is uncovered and cooled by air. In the partially uncovered case, the air flow rate is smaller; and the incoming air temperature is higher, since the ambient air is assumed to mix with the hot air exiting at the top of the assembly. When the upper 3 feet of the rod is exposed to air, approximately 1/3 of the heat generated in the entire rod is removed by air, and the other 2/3 of the heat generated in the lower portion of the rod is removed by water. The exit air temperature depends on the incoming air temperature and the air temperature rise. The air temperature rise is proportional to the heat flow to the air and inversely proportional to the air flow rate. Therefore, when the partially uncovered rod is cooled by air, the maximum air temperature which occurs at the flow channel exit (264°F vs. 207°F) is higher due to higher temperature of the incoming air (192°F vs. 120°F). This occurs even though the air temperature rise is higher in the case where the entire rod is uncovered. The air temperature rise is 72°F vs. 87°F while the difference in air inlet temperature between the two cases is 72°F.

The maximum fuel centerline temperature and the maximum cladding temperature are calculated based on the local peak heat rate, which is identical in both cases, and the maximum air temperature. Therefore, these maximum temperatures are also higher when the partially uncovered rod is cooled by air.

Section 4
RADIOLOGICAL ENVIRONMENT

4.1 EXPOSURE PATHWAYS

4.1.1 Direct and Scattered Radiation

The radioactive material contained in the fuel assemblies is a fixed radiation source that is normally shielded by about fifteen feet of water in the pool to the extent that the dose rate is far below background in accessible areas of the Fuel Building and outside the Fuel Building. When the water is lowered, the shielding is removed and significant radiation field is created in and around the Fuel Building. The radiation field has two components. The first component is direct radiation, which is the radiation that travels in a direct line from the source to the dose point. Due to the depth of the fuel storage pool, the only areas affected by direct radiation are immediately above and very close to the edge of the fuel storage pool. The other component of the radiation field is scattered radiation, which is that radiation that undergoes one or more scatters before reaching the dose point. The primary volume in which scattering will take place is the large cloud of air above the fuel building, which means scattered radiation may affect any area of the plant site.

4.1.2 Effluent Pathways

The fuel assemblies contain a significant amount of the radioactive noble gas nuclide Kr-85. The effect of releasing this activity was previously analyzed [Reference 1] by assuming all the fuel pins in all the assemblies in the Fuel Building fail. Since the Kr-85 activity used in that study (1.4×10^5 Ci) is larger than shown in Table 1, Reference 1 is considered the controlling analysis for the effluent pathway.

4.1.3 Contamination

A number of the fuel rods in the fuel storage pool have defects that allow fission products to enter the fuel pool water. Uncovering the fuel assemblies has the potential for releasing the fission products directly to the air where they may plate out on and contaminate building surfaces. This will occur if a fission product that is normally not gaseous is heated enough to evaporate and then condense on cooler surfaces. However, with the exception of the noble gas radionuclide Kr-85, the temperatures resulting from this accident are not high enough to cause this to occur. Cs-137, the dominant fission product in the fuel, is liquid at 85° F, the normal temperature of the fuel in the storage pool. However, it does not evaporate until it reaches more than 500° F, much higher than the worst case calculated here. In addition, since there is considerable stable iodine in the fuel, it is anticipated that the majority of the Cesium will combine with Iodine to form CsI, which is very soluble but does not melt until it reaches 1000° F. Therefore it is not anticipated that uncovering the fuel will result in increased contamination from activity in the fuel assemblies.

4.2 CALCULATED RADIATION DOSE RATES

4.2.1 Direct Radiation

The dose rates due to direct radiation were calculated using the computer code ISOSHLD-PC, which utilizes the point kernel transport method with infinite media buildup factors for fixed source-shield-dose point configurations. The source was modeled as a homogeneous rectangular solid with various depths of water, starting with an empty pool. The total activity and mass for 683 fuel assemblies and 534 channels were placed in the source. This accounts for the worst case arrangement of the fuel assemblies in the storage and transfer pools. If the channels are removed from the pool, there will be a reduction in mass of the source of less than 10%, which will reduce the self shielding of the source and slightly increase the dose rate. This increase is well within the conservatism of the ISOSHLD-PC infinite media buildup model, which tends to overestimate dose rates by 20-30%.

Dose rates were calculated at the top of the fuel assemblies or surface of the pool of water, and at floor elevation along the edge of the pool. The results are shown in Table 3. At the edge of the pool the dose rate with the pool empty is 800 rem/hr, which drops to 500 rem/hr when the pool is filled to the top of the active region of the fuel. The dose rate then drops rapidly with water level, so that with six feet over the active region of the fuel the dose rate is less than 1 mrem/hr.

4.2.2 Scattered Radiation

The dose rates caused by scattered radiation at specific locations on the site were calculated using the Monte Carlo code MCNP-4A. In addition to scattering in the air, the roof of the Fuel Building and the spherical shell of the Unit 1 containment were modeled as potential scattering surfaces. The source was modeled as a rectangular surface source with an energy dependent cosine flux distribution calculated using the discrete ordinates computer code ANISN. Four specific locations were analyzed (see Figures 2 and 3), which had the following dose rates for the pool with no water:

Inside the Fuel Building

Two locations inside the Fuel Building were analyzed: the pool access area, which is near the pool but out of direct line of sight with the fuel, and the building entrance. The dose rates were 12 rem/hr and 3 rem/hr at those locations, indicating that the entire Fuel Building would be a very high radiation area if the fuel assemblies are uncovered.

Unit 2/3 Control Room

The calculated dose rate at the closest point in the Unit 2/3 Control Room, without considering any intervening shield walls, is less than 200 mrem/hr. The control room has between 1'-6" and 3'-0" of concrete shielding in the walls and slabs that surround it. This concrete provides a protection factor that ranges from $5x10^{-3}$ to $2x10^{-5}$ for gamma radiation with an energy of 0.5 MeV, which is the dominant energy of the scattered radiation. This reduces the dose rate inside the control room to 1 mrem/hr, regardless of the direction of the scattered radiation.

Site Boundary

The closest point on the site boundary is a berm on the discharge canal about 451 meters from the center of the fuel storage pool. The calculated dose rate at this point is less than 2 mrem/hr.

Section 5

REFERENCES

1. "Determination of the Potential Radiological Consequences from a Fuel Handling Accident at the Dresden Nuclear Power Station Unit 1," TLG Engineering, Inc., Document C04-22-002, March, 1989
2. Letter from W.L Axelson, USNRC to M. J. Wallace, Commonwealth Edison Company, Subject: CONFIRMATORY ACTION LETTER (CAL) RIII-94-001 FOR DRESDEN 1, dated February 1, 1994
3. "Offsite and Fuel Bldg Dose Rates from Uncovered Spent Fuel," Sargent & Lundy Calculation ATD-0361, Rev. 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-009
4. ASHRAE 1987 Handbook of Fundamentals, Chapter 20
5. Brown, A.I., and Marco, S.M., Introduction to Heat Transfer, 2nd Ed., McGraw-Hill 1951

Table 1. Dresden Unit 1 Fuel Pool Radioactive Inventory

Isotope	Activity (Ci)
H-3	1.374E+04
Fe-55	1.734E+03
Co-60	2.161E+04
Ni-63	6.238E+03
Kr-85	1.172E+05
Sr-90	1.811E+06
Y-90	1.812E+06
Sb-125	1.187E+04
Te-125m	2.894E+03
Cs-134	1.838E+04
Cs-137	2.650E+06
Ba-137m	2.507E+06
Pm-147	8.550E+04
Sm-151	1.809E+04
Eu-154	7.008E+04
Eu-155	2.006E+04
Pu-238	6.836E+04
Pu-239	1.815E+04
Pu-240	2.068E+04
Pu-241	3.013E+06
Am-241	1.285E+05
Cm-244	2.633E+04
Total	1.244E+07

Table 2. Dresden Unit 1 Fuel Pool Photon Source Strength

Energy Group	Mean Energy (MeV)	Photons/sec
1	0.010	5.881E+16
2	0.025	1.206E+16
3	0.038	1.443E+16
4	0.058	1.261E+16
5	0.085	6.568E+15
6	0.125	5.222E+15
7	0.225	5.466E+15
8	0.375	2.405E+15
9	0.575	9.800E+16
10	0.850	2.192E+15
11	1.250	3.126E+15
12	1.750	5.009E+13
13	2.250	3.602E+10
14	2.750	1.193E+10
15	3.500	7.287E+08
16	5.000	1.669E+08
17	7.000	1.923E+07
18	9.500	2.208E+06
Total		2.209E+17

Table 3. Pool Edge and Water Surface Dose Rates

Water Elevation feet	Water Depth to top of active zone feet	Surface Center Dose Rate** mrem/hour	Pool Edge Dose Rate elev. 521'-3" mrem/hour
<496	<-9	5.875E+6*	7.974E+5
505	0	3.885E+6*	5.290E+5
506	1	1.039E+5	3.529E+4
507	2	6.527E+3	2.659E+3
508	3	5.738E+2	2.524E+2
509	4	6.307E+1	2.890E+1
510	5	7.943E+0	3.737E+0
511	6	1.084E+0	5.184E-1

* This dose point is 1.1 feet above the active zone.
** These dose points are 0.1 feet above water surface, except elev. <496 and 505.

ATTACHMENTS

1. "Unit 1 Fuel Pool Radiological Source Term and Heat Generation Rate," Sargent & Lundy Calculation ATD-0364, Revision 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-09 (56 pages)
2. "Direct Dose Rate at the Edge of the Fuel Storage Pool," Sargent & Lundy Calculation ATD-0365, Revision 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-09 (23 pages)
3. "Scattered Dose Rates from the Fuel Storage Pool," Sargent & Lundy Calculation ATD-0366, Revision 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-09 (33 pages)
4. "Fuel Pool Temperature and Evaporation Rate," Sargent & Lundy Calculation ATD-0367, Revision 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-09 (47 pages)
5. "Cladding and Fuel Centerline Temperature of a Partially Uncovered Spent Fuel Rod," Sargent & Lundy Calculation ATD-0370, Revision 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-09 (16 pages)
6. "Cladding and Fuel Centerline Temperature of a Water Cooled and Air Cooled Spent Fuel Rod," Sargent & Lundy Calculation ATD-0371, Revision 0, Dresden Nuclear Power Station, Unit 1, Proj. No. 09389-09 (22 pages)

ATTACHMENT B

CAL Item 3: Contamination Estimates.

D-1 Fuel Service Building
Revised Surface & Airborne Contamination Calculations

Scope:

The NRC has requested that Dresden Station calculate the estimated surface contamination levels in the Fuel Service Building if the fuel pool were to lose water level down to the 502' elevation. An estimate of the airborne contamination levels due to the surface contamination is included.

As part of a Unit-1 fuel pool clean-up campaign started late in 1993, CECo contracted Scientech, Inc. to conduct sampling and characterization of the materials in the fuel pool in accordance with 10 CFR 61. This report, dated October 13, 1993, was used to evaluate the contamination levels that might be present. The report is available for review.

Scientech vacuumed two fuel pool sludge samples as part of their sampling regimen. The total surface area over which these samples were taken is unknown, therefore these samples were not used for this evaluation. Scientech obtained surface samples on several components in the storage pool, including a fuel element flow channel. These samples should be representative of the activity deposited on other fuel elements, and the flow channel results were used for this report. The surface area of the samples ranged from 50 to 100 cm^2. To be conservative, it was assumed that each sample was taken over a 50 cm^2 area.

A report entitled "Resuspension Factors and Probabilities of Intake of Material in Process" by Alan Brodsky (*Health Physics Vol. 39*) was reviewed in an effort to calculate the potential airborne radionuclide concentration(s) in the Fuel Service Building should the fuel become uncovered. Resuspension factors may vary from 1.0 E -13 m^{-1} to 1.0 E -03 m^{-1}, and are dependant on many factors. Under conditions of light work and moderate area ventilation, a resuspension factor of 1.0 E -06 m^{-1} is reasonable and conservative. After conversion to more useful units, this corresponds to a resuspension factor of 1.0 E -08 cm^{-1}.

For the purposes of this estimation, we will assume a resuspension factor that is ten times more limiting. A factor of 1.0 E -07 cm^{-1} will be used.

Assumptions:

1. Each surface contamination sample was taken over a 50 cm^2 area.
2. The flow channel surface activity is representative of the activity that is deposited on the other fuel assemblies and the fuel racks.
3. Fission product decay gases and spread of radioactivity due to a postulated release of gases from exposed fuel assemblies were not considered in this estimation.
4. Heat up of the exposed fuel and remaining fuel pool water is negligible.
5. Building ventilation was not running, and would remain off.
6. A resuspension factor of 1.0 E -07 cm^{-1} is appropriate.

Surface Contamination:

Smears on the fuel pool water line averaged:
1.3 million dpm/100cm^2 Beta/Gamma radioactivity
780 dpm/100cm^2 Alpha radioactivity

A back-calculation of the fuel surface contamination levels indicates the potential for:
5.9 million dpm/100cm^2 Beta/Gamma radioactivity
1,000 dpm/100cm^2 Alpha radioactivity
(on exposed fuel surfaces)

Surface contamination plate-out due to airborne radioactivity is generally quite small in magnitude, and is usually seen only in extremely dusty conditions. These conditions are not expected in this scenario. Surface contamination (due to airborne radioactivity plate-out only) should be in the area of:
1K dpm/100cm^2 Beta/Gamma radioactivity
< 1 dpm/100cm^2 Alpha radioactivity

With the above assumptions, it is expected that there will be little additional spread of surface contamination outside of the recessed fuel pool areas. Attempts to correct problems and to re-fill the pool(s) could cause cross contamination, in some areas, of up to the above listed levels for the fuel pool water-line. Aggressively spraying water directly onto exposed fuel and fuel rack surfaces could result in the spread of significant levels of additional contamination.

Airborne Radioactivity:

Radionuclide concentrations are given in Table 7.6 of the Scientech report. The attached spreadsheet was used to estimate airborne radioactivity based upon the previously mentioned resuspension factor. DAC values were also calculated, and are provided.

To summarize, airborne radioactivity could be present at levels of:

Beta/Gamma	2.67 E -09 uCi/cc gross activity *or*	a total of 0.29 DACs
Alpha	4.70 E -13 uCi/ml gross activity *or*	a total of 0.15 DACs

It is important to note that these figures are for long term steady state conditions, and are for general areas of the Fuel Service Building. Aggressive work on highly contaminated surfaces could cause a localized high airborne radioactivity condition.

Prepared by: Harry Anagnostopoulos

Harry Anagnostopoulos
ALARA Planner
Dresden Station
02/16/94

D-1 Fuel Building Postulated Airborne Concentrations

2/16/94 9:17 AM

Radionuclide Beta / Gamma	Sample Activity (uCi / 50 cm2)	Calculated Activity (uCi / cm2)	Calculated Airborne (uCi / cc)	DAC value (uCi / cc)	Number of DACs
Co-60	1.22E+00	2.44E-02	2.44E-09	1.00E-08	0.244
Sr-89	1.74E-04	3.48E-06	3.48E-13	6.00E-08	0.0000058
Sr-90	3.51E-03	7.02E-05	7.02E-12	2.00E-09	0.00351
Cs-134	2.97E-04	5.94E-06	5.94E-13	4.00E-08	0.00001485
Cs-137	1.08E-01	2.16E-03	2.16E-10	6.00E-08	0.0036
Pu-241	1.80E-03	3.60E-05	3.60E-12	1.00E-10	0.036
Gross Totals		2.67E-02	2.67E-09		0.29

Radionuclide Alpha	Sample Activity (uCi / 50 cm2)	Calculated Activity (uCi / cm2)	Calculated Airborne (uCi / cc)	DAC value (uCi / cc)	Number of DACs
Pu-238	5.63E-05	1.13E-06	1.13E-13	3.00E-12	0.04
Pu-239	5.78E-05	1.16E-06	1.16E-13	3.00E-12	0.04
Am-241	1.08E-04	2.16E-06	2.16E-13	3.00E-12	0.07
Cm-244	1.31E-05	2.62E-07	2.62E-14	5.00E-12	0.01
Gross Totals		4.70E-06	4.70E-13		0.15

Calculated airborne radioactivity is based upon a resuspension factor of 1.0 E -07 cm-1

Based upon postulated loss of fuel pool water level

NUCLIDE CHARACTERIZATION OF FUEL POOL STORED COMPONENTS AT COMMONWEALTH EDISON DRESDEN NUCLEAR STATION UNIT #1

DRAFT REPORT
011R-93-008

Submitted to

Commonwealth Edison
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

October 13, 1993

CORPORATE HEADQUARTERS:
1690 INTERNATIONAL WAY
IDAHO FALLS, IDAHO 83402

Table 7.6

Unit #1, Flow Channel Sample Activity Concentrations				
NUCLIDE CONCENTRATION	BULK SAMPLE #23417		SURFACE SAMPLE #23418	
	ACTIVITY	UNCERTAINTY	ACTIVITY	UNCERTAINTY
C-14	<3.0E-06			
Fe-55	5.15E-02	10 %		
Ni-59	1.07E-01	16 %		
Co-60	1.06E+00	10 %	1.22E+00	10 %
Ni-63	6.73E+00	10 %		
Sr-89			1.74E-04	43 %
Sr-90			3.51E-03	10 %
Nb-94	<1.2E-05			
Tc-99	<8.7E-06			
Sb-125	9.83E-03	16 %		
Cs-134	1.87E-04	38 %	2.97E-04	21 %
Cs-137	4.98E-02	10 %	1.08E-01	10 %
Pu-238			5.63E-05	10 %
Pu-239			5.78E-05	10 %
Am-241			1.08E-04	10 %
Pu-241			1.80E-03	35 %
Cm-242			<4.2E-07	
Cm-244			1.31E-05	10 %
Gross Alpha	6.88E-04	14 %	2.39E-04	27 %

ATTACHMENT C

CAL Item 5: Fuel Assembly Audit.

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[illegible], NMC 2-17-94

Row	R	Q	P	O	N	M	L	K	J	I	H	G	F	E
1	UN 440	UN [illegible]	A-011	A-037	UN 295	UN 316	UN 332	UN 203	UN 021	UN 177	DU 34	DU 22	UN 380	UN 305
2	UN 406	UN 259	A-022	A-005	UN 202	UN 415	UN 397	UN 309	UN 008	UN 435	DU 18	DU 102	UN 360	UN 199
3	UN 148	UN 192	A-001	A-003	UN 409	UN 419	UN 325	UN 283	UN 451	UN 423	DU 87	DU 47	UN 350	UN 210
4	UN 349	UN 289	A-056	A-031	UN 417	UN 307	UN 329	UN 260	UN 172	UN 311	DU 74	DU 99	UN 450	UN 252
5	UN 279	UN 204	A-012	A-039	UN 080	UN 178	UN 385	UN 219	UN 454	UN 438	DU 105	DU 23	UN 447	UN 257
6	UN 352	UN 297	A-065	A-004	UN 188	UN 017	UN 333	UN 225	UN 388	UN 238	DU 49	DU 68	UN 326	UN 271
7	UN 323	UN 218	A-018	A-040	UN 130	UN 012	UN 378	UN 194	UN 328	UN 247	DU 78	DU 70	UN 184	UN 195
8	UN 449	UN 284	A-066	A-023	UN 412	UN 411	UN 363	UN 245	UN 321	UN 286	DU 17	DU 69	UN 330	UN 266
9	UN 371	UN 212	A-014	A-052	UN 426	UN 459	UN 310	UN 304	UN 322	UN 275	UN 013	DU 10	UN 354	UN 228
10	UN 369	UN 237	A-019	A-010	UN 296	UN 314	UN 398	UN 402	UN 339	UN 231	UN 067	DU 38	UN 361	UN 232
11	UN 377	UN 193	A-016	A-055	UN 288	UN 382	UN 433	UN 208	UN 376	UN 306	UN 125	UN 039	UN 033	UN 170
12	UN 023	UN 207	A-026	A-050	UN 287	UN 346	UN 018	UN 291	UN 386	UN 270	UN 030	UN 271	UN 136	UN 163
13	UN 126	UN 062	A-030	A-021	UN 286	UN 381	UN 248	UN 201	UN 367	UN 276	UN 429	UN 424	UN 455	UN 442
14	UN 055	UN 161	A-062	A-013	UN 253	UN 340	UN 206	UN 213	UN 358	UN 273	UN 357	UN 296	UN 437	UN 436
15	UN 091	UN 070	A-029	A-015	UN 235	UN 327	UN 149	UN 112	UN 339	UN 200	UN 395	UN 267	UN 432	UN 410
16	UN 401	UN 414	A-007	A-060	UN 087	UN 421	UN 430	UN 407	UN 373	UN 223	UN 359	UN 196	UN 461	UN 393
17	UN 418	UN 047	A-039	A-032	UN 171	UN 335	UN 175	UN 249	UN 324	UN 224	G-81	G-7	UN 251	UN 009
18	UN 298	UN 041	A-028	A-002	UN 285	UN 374	UN 268	UN 185	UN 338	UN 216	G-26	G-71	UN 029	UN 014
19	UN 060	UN 366	A-035	A-046	UN 279	UN 336	UN 317	UN 227	UN 368	UN 230	G-13	G-88	UN 444	UN 392
20	UN 300	UN 347	A-043	A-038	UN 260	UN 174	UN 381	UN 215	UN 320	UN 278	G-75	G-47	UN 443	UN 431
21	UN 262	UN 355	A-006	A-047	UN 455	UN 428	UN 353	UN 233	UN 383	UN 211	G110	G-55	UN 391	UN 445
22	UN 272	UN 365	A-064	A-025	UN 405	UN 400	UN 357	UN 313	UN 434	UN 456	G-43	G-62	UN 390	UN 441
23	UN 240	UN 318	A-008	A-009	UN 254	UN 186	UN 384	UN 312	UN 214	UN 141	G-38	E91	UN 146	UN 118
24	UN 197	UN 457	A-033	A-045	UN 145	UN 282	UN 379	UN 303	UN 043	UN 024	G-8	G-21	UN 057	UN ~~008~~ 053
25	UN 230	UN 345	A-017	A-024	UN 179	UN 169	UN 375	UN 221	G-35	UN 121	DU 29	DU 64	UN 051	UN 364
26	UN 222	UN 343	A-043	A-027	UN 269	UN 263	UN 331	UN 255	UN 088	UN 102	DU 71	DU 15	UN 162	UN 264
27	UN 244		A-051	A-042	UN 083	UN 182	UN 372	UN 217	DU 32	UN 296	DU 45	DU 57	UN 063	UN 394
28			A-061	A-048	UN 122	UN 453	UN 315	UN 220	UN 128	UN 114	DU 77	DU 54	UN 446	UN 448
29			A-053	A-049	UN 403	UN 452	UN 431	UN 427	DU 85	UN 105	DU 73	DU 62	UN 166	UN 006
30			A-058	A-020	UN 293	UN 301	UN 370	UN 280	UN 103	UN 045	DU 19	DU 6	UN 138	UN 116
31			A-057	A-044	UN 356	UN 281	UN 068	UN 198	UN 143	DU 11	DU 33	DU 2	UN 154	UN 044
32			A-036	A-041	UN 344	UN 229	UN 243	UN 209	UN 106	UN 057	DU 76	DU 44	UN 164	UN 159
33	UN 413	UN 422	A-074	A-059	UN 101	UN 108	G-65	G-57	UN 030	DU 1	DU 95	DU 7	UN 151	UN 181
34	UN 460	UN 425	UN 070	UN 160	UN 191	UN 119	G-66	DU 55	UN 089	UN 104	DU 39	DU 37	DU 16	UN 261
35	UN 183	G-22	UN 290	UN 081	UN 150	UN 117	DU 35	DU 21	UN 048	DU 74	DU 94	DU 75	UN 049	DU 12
36	G-100	G-41	UN 176	UN 090	UN 111	DU 56	UN 005A	UN 035	DU 28	UN 155	DU 8	DU 30	G-78	G-82
37	G-23	G-6	E156	DU 88	UN 031	DU 26	DU 36	E153	DU 31	UN 027	DU 84	DU 3	DU 17	G-84
38	G-70	G-92	UN 054	UN 022	UN 107	UN 050	E152	DU 97	UN 005	UN 115	DU 83	DU 53	UN [illegible]	E144
39	G-67	G-3	UN 258	UN 020	UN 165	UN 362	DU 91	G-19	UN 046	DU 41	DU 60	DU 90	E55	DU 40
40	G-32	G-12	XE50	E59	UN 075	UN 025	UN 069	E86	DU 85	UN 034	DU 4	DU 48	G-86	DU 104
41	G-4	G-18	E118	E94	UN 139	E109	UN 274	UN 205	UN 006	UN 086	DU 43	DU 30	G-7	E159
42	G-74	XE56	UN 142	UN 153	E160	UN 250	UN 120	UN 100	UN 127	DU 25	DU 19	DU 101	UN 377	E168
43	E98	G-99	UN 147	UN 098	XE109	E114	UN 256	DU 13	UN 133	UN 059	UN 037	UN 042	G-63	G-29
44	E110	E126	UN 129	UN 190	F4	E155	UN 234	UN 302	UN 056	DU 9	UN 137	UN 167	E102	UN 082
45	E112	G-94	UN 109	UN 113	UN 019	E25	UN 085	UN 015	G-40	UN 277	E95	UN 308	E77	DU 72
46	UN 44	UN 004	UN 131	UN 099	E52	UN 094	UN 061	UN 092	G-58	G-93	UN 246	UN 046	DU 42	E72
47	UN 026	UN 157	UN 123	UN 132	E49	E131	UN 052	UN 003	G-41	G-37	UN 299	UN 074	G-97	E75
48	UN 071	rod basket	UN 192	UN 152	UN 012	UN 079	XE2	UN 135	G-53	G-9	UN 007	UN 078	G-76	E150

Row	D	C	B	A
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There is one rod basket in cell Q-48. There are 23 fuel bundles in the 2 transfer baskets and 660 fuel bundles in the main of the rack fuel storage pool, bringing the total number of bundles to 683. ECK 2-17-94

Of the 683 fuel bundles, 394 are channeled and 289 are not channeled. ECK 2-17-94

□ = CHANNEL

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GJD 7-11-68

Mike Helgeson 2-17-94

T. A. S[illegible] 2-17-94

SPENT FUEL STORAGE INVENTORY

N

DATE 2-17-94

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	R	Q	P	O	N	M	L	K	J	I	H	G	F	E
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3	UN 348	UN 242	A-001	A-003	UN 409	UN 419	UN 325	UN 283	UN 451	UN 423	DU 87	DU 47	UN 350	UN 210
4	UN 349	UN 289	A-056	A-031	UN 417	UN 307	UN 329	UN 265	UN 172	UN 311	DU 74	DU 99	UN 450	UN 252
5	UN 279	UN 204	A-012	A-039	UN 080	UN 178	UN 385	UN 219	UN 454	UN 438	DU 105	DU 23	UN 447	UN 257
6	UN 352	UN 297	A-065	A-004	UN 188	UN 017	UN 333	UN 225	UN 388	UN 238	DU 49	DU 68	UN 326	UN 234
7	UN 323	UN 218	A-018	A-040	UN 130	UN 012	UN 378	UN 194	UN 328	UN 247	DU 78	DU 70	UN 184	UN 195
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9	UN 371	UN 212	A-014	A-052	UN 426	UN 459	UN 310	UN 304	UN 322	UN 275	UN 013	DU 10	UN 354	UN 228
10	UN 369	UN 237	A-019	A-010	UN 296	UN 314	UN 398	UN 402	UN 319	UN 231	UN 067	DU 38	UN 361	UN 232
11	UN 377	UN 193	A-016	A-055	UN 288	UN 382	UN 433	UN 208	UN 376	UN 306	UN 125	UN 039	UN 033	UN 170
12	UN 023	UN 207	A-026	A-050	UN 287	UN 346	UN 018	UN 291	UN 386	UN 270	UN 030	UN 271	UN 136	UN 163
13	UN 126	UN 062	A-030	A-021	UN 286	UN 381	UN 248	UN 201	UN 367	UN 276	UN 429	UN 424	UN 455	UN 442
14	UN 055	UN 161	A-062	A-013	UN 253	UN 340	UN 206	UN 213	UN 358	UN 273	UN 357	UN 396	UN 437	UN 436
15	UN 091	UN 076	A-029	A-015	UN 235	UN 327	UN 149	UN 112	UN 339	UN 200	UN 395	UN 267	UN 432	UN 410
16	UN 401	UN 414	A-007	A-060	UN 087	UN 421	UN 430	UN 407	UN 373	UN 223	UN 359	UN 196	UN 461	UN 393
17	UN 418	UN 047	A-034	A-032	UN 171	UN 335	UN 175	UN 249	UN 324	UN 224	G-81	G-7	UN 251	UN 009
18	UN 298	UN 041	A-028	A-002	UN 285	UN 374	UN 268	UN 18[illegible]	UN 335	UN 216	G-26	G-71	UN 029	UN 014
19	UN 060	UN 366	A-035	A-046	UN 274	UN 336	UN 317	UN 227	UN 368	UN 230	G-13	G-88	UN 444	UN 392
20	UN 300	UN 347	A-063	A-038	UN 260	UN 174	UN 389	UN 215	UN 320	UN 278	G-75	G-47	UN 443	UN 439
21	UN 262	UN 355	A-006	A-047	UN 458	UN 428	UN 353	UN 233	UN 383	UN 211	G110	G-55	UN 341	UN 445
22	UN 272	UN 365	A-064	A-025	UN 405	UN 400	UN 387	UN 313	UN 434	UN 456	G-43	G-62	UN 390	UN 441
23	UN 240	UN 318	A-008	A-009	UN 254	UN 186	UN 384	UN 312	UN 214	UN 141	G-38	E91	UN 146	UN 118
24	UN [illegible]	UN [illegible]	[illegible]	A-[illegible]	[illegible]	[illegible]	UN [illegible]	UN [illegible]	UN [illegible]	UN [illegible]	[illegible]		UN [illegible]	UN [illegible]

[illegible] 2-17-94

	D	C	B	A
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				

There is one rod basket in cell Q-48. There are 23 fuel bundles in the 2 transfer baskets and 660 fuel bundles in the racks of the spent fuel storage pool, bringing the total number of bundles

...les in the 2 transfer baskets and bringing the total number of bundles to 683. LEZ 2-17-93

Of the 683 fuel bundles, 534 are ... 149 are not channeled. LEZ 2-...

2 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

Row														
17	UN 418	UN 047	A-034	A-032	UN 171	UN 335	UN 175	UN 249	UN 324	UN 224	G-81	G-7	UN 251	UN 009
18	UN 298	UN 041	A-028	A-002	UN 285	UN 374	UN 268	UN 185	UN 338	UN 216	G-26	G-71	UN 029	UN 014
19	UN 060	UN 366	A-035	A-046	UN 274	UN 336	UN 317	UN 227	UN 368	UN 230	G-13	G-88	UN 444	UN 392
20	UN 300	UN 347	A-063	A-038	UN 260	UN 174	UN 381	UN 215	UN 320	UN 278	G-75	G-47	UN 443	UN 431
21	UN 262	UN 355	A-006	A-047	UN 455	UN 428	UN 353	UN 233	UN 383	UN 211	G110	G-55	UN 341	UN 445
22	UN 272	UN 365	A-064	A-025	UN 405	UN 400	UN 387	UN 313	UN 434	UN 456	G-43	G-62	UN 390	UN 441
23	UN 240	UN 313	A-008	A-009	UN 254	UN 186	UN 384	UN 312	UN 214	UN 141	G-38	E91	UN 146	UN 118
24	UN 197	UN 457	A-033	A-045	UN 145	UN 282	UN 379	UN 303	UN 043	UN 024	G-8	G-21	UN 057	UN ~~055~~ 053 LEZ 2-17-93
25	UN 236	UN 345	A-017	A-024	UN 179	UN 169	UN 375	UN 221	G-35	UN 121	DU 29	DU 64	UN 051	UN 364
26	UN 222	UN 343	A-043	A-027	UN 269	UN 263	UN 331	UN 255	UN 088	UN 102	DU 71	DU 15	UN 162	UN 264
27	UN 244		A-051	A-042	UN 083	UN 182	UN 372	UN 217	DU 32	UN 096	DU 45	DU 57	UN 063	UN 394
28			A-061	A-048	UN 122	UN 453	UN 315	UN 220	UN 128	UN 114	DU 77	DU 54	UN 446	UN 448
29			A-053	A-049	UN 403	UN 452	UN 431	UN 427	DU 85	UN 105	DU 73	DU 62	UN 166	UN 001A
30			A-058	A-020	UN 243	UN 301	UN 370	UN 280	UN 103	UN 045	DU 19	DU6	UN 138	UN 116
31			A-057	A-044	UN 356	UN 281	UN 068	UN 198	UN 143	DU 11	DU 33	DU2	UN 154	UN 044
32			A-036	A-041	UN 344	UN 229	UN 243	UN 209	UN 106	UN 097	DU 76	DU 44	UN 164	UN 159
33	UN 413	UN 422	A-054	A-059	UN 101	UN 108	G-65	G-57	UN 036	DU1	DU 95	DU 7	UN 151	UN 181
34	UN 460	UN 425	UN 070	UN 160	UN 191	UN 119	G-66	DU 55	UN 089	UN 104	DU 39	DU 37	DU 16	UN 261
35	UN 183	G-22	UN 290	UN 081	UN 150	UN 117	DU 35	DU 21	UN 048	DU 79	DU 94	DU 75	UN 049	DU 12
36	G-16	G-41	UN 176	UN 090	UN 111	DU 56	UN 005A	UN 035	DU 28	UN 155	DU8	DU 30	G-78	G-82
37	G-23	G-6	E156	DU 88	UN 031	DU 26	DU 86	E153	DU 31	UN 027	DU 84	DU3	DU 27	G-84
38	G-70	G-92	UN 054	UN 002	UN 107	UN 050	E152	DU 97	UN 095	UN 115	DU 83	DU 53	UN 028	E144
39	G-67	G-2	UN 058	UN 020	UN 165	UN 362	DU 91	G-69	UN 046	DU 41	DU 60	DU 90	E85	DU 40
40	G-32	G-12	XE50	E59	UN 075	UN 025	UN 069	E66	DU 65	UN 034	DU4	DU 48	G-36	DU 104
41	G-4	G-18	E118	E94	UN 139	E109	UN 294	UN 205	UN 006	UN 086	DU 43	DU 30	G-5	E159
42	G-74	XE56	UN 142	UN 153	E160	UN 250	UN 120	UN 100	UN 127	DU 25	DU [illegible]	DU 101	UN 077	E168

□ = CHANNEL

□ = CHANNEL

25	UN 236	UN 345	A-017	A-024	UN 179	UN 169	UN 375	UN 221	G-35	UN 121	DU 29	DU 64	UN 051	UN 364
26	UN 222	UN 343	A-043	A-027	UN 264	UN 263	UN 331	UN 255	UN 088	UN 102	DU 71	DU 15	UN 162	UN 264
27	UN 244		A-051	A-042	UN 083	UN 182	UN 372	UN 217	DU 32	UN 096	DU 45	DU 57	UN 063	UN 394
28			A-061	A-048	UN 122	UN 453	UN 315	UN 220	UN 128	UN 114	DU 77	DU 54	UN 446	UN 448
29			A-053	A-049	UN 403	UN 452	UN 431	UN 427	DU 85	UN 105	DU 73	DU 62	UN 166	UN 001A
30			A-058	A-020	UN 293	UN 301	UN 370	UN 280	UN 103	UN 045	DU 19	DU6	UN 138	UN 116
31			A-057	A-044	UN 356	UN 281	UN 068	UN 198	UN 143	DU 11	DU 33	DU2	UN 154	UN 044
32			A-036	A-041	UN 344	UN 229	UN 243	UN 209	UN 106	UN 097	DU 76	DU 44	UN 164	UN 159
33	UN 413	UN 422	A-054	A-059	UN 101	UN 108	G-65	G-57	UN 036	DU1	DU 95	DU 7	UN 151	UN 181
34	UN 460	UN 425	UN 070	UN 160	UN 191	UN 119	G-66	DU 55	UN 089	UN 104	DU 39	DU 37	DU 16	UN 261
35	UN 183	G-22	UN 290	UN 081	UN 150	UN 117	DU 35	DU 21	UN 048	DU 79	DU 94	DU 75	UN 049	DU 12
36	G-16	G-41	UN 176	UN 090	UN 111	DU 56	UN 005A	UN 035	DU 28	UN 155	DU8	DU 30	G-78	G-82
37	G-23	G-6	E156	DU 88	UN 031	DU 26	DU 86	E153	DU 31	UN 027	DU 84	DU3	DU 27	G-84
38	G-70	G-92	UN 054	UN 002	UN 107	UN 050	E152	DU 97	UN 095	UN 115	DU 83	DU 53	UN 028	E144
39	G-67	G-2	UN 058	UN 020	UN 165	UN 362	DU 91	G-69	UN 046	DU 41	DU 60	DU 90	E85	DU 40
40	G-32	G-12	XE50	E59	UN 075	UN 025	UN 069	E66	DU 65	UN 034	DU4	DU 48	G-86	DU 104
41	G-4	G-18	E118	E94	UN 139	E109	UN 294	UN 205	UN 006	UN 086	DU 43	DU 30	G-5	E159
42	G-74	XE56	UN 142	UN 153	E160	UN 250	UN 120	UN 100	UN 127	DU 25	DU 58	DU 101	UN 077	E168
43	E98	G-50	UN 147	UN 098	XE104	E114	UN 256	DU 13	UN 133	UN 059	UN 037	UN 042	G-63	G-29
44	E110	E120	UN 129	UN 190	F4	E155	UN 234	UN 302	UN 056	DU 9	UN 137	UN 167	E102	UN 082
45	E112	G-54	UN 109	UN 113	UN 019	E25	UN 085	UN 015	G-90	UN 277	E95	UN 308	E73	DU 72
46	UN 144	UN 004	UN 131	UN 099	E52	UN 094	UN 061	UN 092	G-58	G-93	UN 246	UN 040	DU 42	E72
47	UN 026	UN 157	UN 123	UN 132	E49	E131	UN 052	UN 003	G-91	G-37	UN 299	UN 074	G-77	E75
48	UN 071	rod basket	UN 192	UN 152	UN 072	UN 074	XE22	UN 135	G-53	G-9	UN 007	UN 078	G-76	E150

18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36

= 683. [illegible] 2-17-93

Of the 683 fuel bundles, 534 are channeled and 149 are not channeled. [illegible] 2-17-94

GJD 7-11-66

MAP OF THE DRESDEN STATION UNIT 1

SPENT FUEL STORAGE POOL (SFSP) TRANSFER BASKETS

North ↑

Northern Transfer Basket;

	D	C	B	A
1	UN 408	UN 292	EMPTY	EMPTY
2	UN 420	UN 124	UN 341	EMPTY
3	UN 416	UN 110	UN 241	EMPTY
4	UN 404	UN 011	UN 342	EMPTY

□ = CHANNEL

Southern Transfer Basket;

	D	C	B	A
1	UN 134	UN 084	UN 140	EMPTY
2	UN 148	EMPTY	UN 189	EMPTY
3	UN 180	UN 173	UN 158	EMPTY
4	UN 187	UN 168	UN 156	UN 351

Performed By: [signature] N/A 2-17-94

Performed On: 2-17-94

Approved By: [signature] 2-17-94 (NMC) [signature] 2-17-94 (Auditor)

Technical Services Superintendent: [signature] Date: 2-17-94