DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS OUTSIDE CONTAINMENT AT R. E. GINNA NUCLEAR POWER PLANT

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1.0 INTRODUCTION

This report has been prepared in response to the September 13, 1979 letter from Darrel G. Eisenhut of the NRC to all operating nuclear power plants. This letter presented an implementation schedule for the recommendations presented in "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578). The requirements defined in the September 13th letter were subsequently clarified in a letter from Harold R. Denton to all operating nuclear power plants dated October 30, 1979.

Item 2.1.6.b of NUREG-0578 requires a design review to determine whether post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of, or recovery from an accident; or unduly degrade the proper operation of safety equipment. Corrective actions for problems identified as a result of the review are also to be determined. This report presents the results of such a review for the R. E. Ginna Nuclear Power Plant.

The review was based on the following guidelines:

a) The post-accident dose rate in areas requiring continuous occupancy should not exceed 15 mr/hr.

b) The post-accident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 rem whole body or its equivalent.

c) The integrated dose to safety equipment as a result of the accident should be less than the dose for which the equipment has been qualified to ensure that the capability of the equipment to perform its safety function has not been degraded.

d) The minimum radioactive source term used in the evaluation should be equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.4.

2.0 SUMMARY

This report provides a review of the R. E. Ginna Nuclear Power Plant to determine whether post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of or recovery from an accident or unduly degrade the proper operation of safety equipment. This review was accomplished using the recommendations and guidelines of NUREG-0578 and subsequent clarifications as provided by the NRC.

This evaluation identified the most critical areas requiring personnel access following the onset of extreme accident conditions following a postulated major release of radioactivity into the Containment Building. Consideration was given to areas where predetermined post-accident functions would be performed (Nuclear Sample Room, Chemistry Laboratory, Count Room, Control Room and air sample penetrations) and to areas where personnel could be called upon to execute certain accident-mitigating or short-term recovery tasks (hydrogen recombiner panel, radwaste panel and building ventilation filters). Potential radiation exposures were determined for each task and included additional exposure due to accessing the areas considered.

The review has identified certain areas which may require modifications to existing procedures, shielding, and/or relocation of equipment. These are discussed in detail in Sections 4.0 and 5.0 and additionally in the RG&E response to Item 2.1.8.a.

The qualification of safety-related equipment has also been reviewed under the assumed accident condition guidance provided in NUREG-0578. The results of that review are presented in Section 5.2.

We have concluded that the actions which may be taken under the conservative post-accident conditions assumed in this analysis, can be accomplished without causing undue radiation exposures to personnel. Where procedural changes and equipment modifications are needed to conform with the dose criteria provided in NUREG-0578, we have identified those changes which are considered most practicable. It is recognized that in addition to such changes, administrative controls would be exercised to further reduce doses to individuals to the lowest practical extent.

We have not identified any safety-related component which would be likely subject to performance degradation, even under the highly conservative accident conditions assumed.

3.0 SOURCE TERMS AND CALCULATIONAL METHODOLOGY

3.1 SOURCE TERMS

The activity assumed for liquid source term calculation is based on 100% of the noble gas inventory, 50% of the halogen core inventory and 1% of all other nuclides in the core inventory. The activity assumed for gaseous source term calculation is based on 100% of the noble gas core inventory and 25% of the halogen core inventory. This approach is, it should be noted, quite conservative. For example, it is impossible to have the entire noble gas inventory of the core simultaneously in both the liquid and the gas sources.

Two liquid source terms were used in the evaluation. For systems which contain post accident recirculation fluid, the source term was based on diluting the liquid inventory discussed in the previous paragraph with 303,800 gallons of fluid filling the containment sump from the refueling water storage tank, both accumulators and a boric acid tank. These systems include the following:

Residual Heat Removal
Containment Spray Recirculation
High Pressure Injection
Nuclear Sampling (RHR Process Fluid)

For systems which can contain fluid from the reactor coolant system but do not take suction on the containment sump, the source term was based on diluting the liquid inventory with the 46,600 gallons in the reactor coolant system. This source was used for the nuclear sampling system (reactor coolant fluid) and the portion of the chemical and volume control system associated with reactor coolant degassing.

Gaseous source terms were determined for containment and for the waste gas system. The containment airborne source term was based on diluting the gaseous inventory discussed previously with the air contained in the containment free volume (970,000 cubic feet). The waste gas system source term was determined for a reactor coolant degassing operation by calculating the quantity of activity entering the volume control tank (VCT) via the normal letdown path and assuming that a quantity equal to the stripping fraction (based upon a stripping efficiency of 1.0) enters the vapor space and is immediately purged to the vent header system (Ref. 1). The stripping fraction is defined as:

$$SF = \frac{CR - CL}{CR}$$

where: SF = stripping fraction

CR = concentration entering VCT in liquid

CL = concentration leaving VCT in liquid

Table 3-1 provides the shielding source terms for liquid and gaseous radioactivity which were calculated using the assumptions above.

3.2 METHODOLOGY

3.2.1 Calculation of Dose Rates

Dose rates for the areas of interest in this review were calculated by determining the potential contributing sources at a representative location and using the appropriate source term from Table 3-1 adjusted for decay as required. The dose rate at the representative location was used as the general area dose rate for the area and chosen to envelope all critical locations of interest. The SDC Code (Ref. 2)

was used in performing the dose rate calculations. Energy groups required as input to the code were determined using the gamma ray energy and intensity data (Refs. 3 and 4) for the nuclides in Table 3-1.

3.2.2 Calculation of Doses to Personnel During Post Accident Access to Vital Areas

Personnel doses received in performing a specific task in a given vital area were calculated as the sum of the doses received during travel to and from the vital area and the dose received while performing the given operation in the vital area.

The doses received during travel were determined by calculating dose rates at selected locations (or at a single location if the dose rate along the travel route is relatively uniform) along the travel route using the methodology discussed in Section 3.2.1 and multiplying the dose rates by the appropriate travel times.

Doses received while performing a given operation were determined by multiplying the dose rate for the given area by the time to perform the operation. Dose rates for the given vital area were calculated using the methodology discussed in Section 3.2.1. Travel times were based upon an assumed access rate of 200 feet per minute. Time estimates for performing indicated tasks were based upon experience gained during a decade of plant operation at Ginna Station.

3.2.3 Calculation of Integrated Doses to Safety Equipment

The integrated dose to each piece of safety-related equipment was determined by integrating the appropriate dose rate over the time period that it is required to be available to perform its safety

function. Dose rates were calculated using the methodology discussed in Section 3.2.1.

TABLE 3-[illegible]

R. E. GINNA
Shielding Source Terms (T=0)

Isotope	Liquid(1) Source Activity (Ci)	Gaseous(2) Source Activity (Ci)	Containment Sump Concentration (μCi/cc)	Reactor Coolant Concentration (μCi/cc)	Containment Airborne Concentration (μCi/cc)	Waste Gas Concentration (μCi/cc)
Br-84	3.9 + 6	2.0 + 7	3.5 + 3	2.2 + 4	7.2 + 1	3.4 + 3
Kr-87	1.9 + 7	1.9 + 7	1.7 + 4	1.1 + 5	7.0 + 2	6.7 + 5
Te-133	2.3 + 5	-	2.1 + 2	1.3 + 3	-	-
Cs-134	8.9 + 4	-	8.0 + 1	5.0 + 2	-	-
Cs-136	2.5 + 4	-	2.2 + 1	1.4 + 2	-	-
Cs-137	3.7 + 4	-	3.3 + 1	2.1 + 2	-	-
Ba-139	7.9 + 5	-	7.0 + 2	4.4 + 3	-	-
Br-83	1.7 + 6	8.4 + 5	1.5 + 3	9.5 + 3	3.0 + 1	1.9 + 2
Kr-83m	3.3 + 6	3.3 + 6	3.0 + 3	1.9 + 4	1.2 + 2	1.1 + 5
Kr-85m	1.0 + 7	1.0 + 7	8.9 + 3	5.6 + 4	3.6 + 2	2.6 + 5
Kr-85	3.9 + 5	3.9 + 5	3.5 + 2	2.2 + 3	1.4 + 1	5.9 + 3
Kr-88	2.9 + 7	2.9 + 7	2.6 + 4	1.6 + 5	1.0 + 3	8.5 + 5
Rb-88	2.8 + 5	-	2.5 + 2	1.6 + 3	-	-
Rb-89	3.6 + 5	-	3.2 + 2	2.0 + 3	-	-
Sr-89	3.9 + 5	-	3.5 + 2	2.2 + 3	-	-
Sr-90	2.6 + 4	-	2.3 + 1	1.5 + 2	-	-
Y-90	2.6 + 4	-	2.3 + 1	1.5 + 2	-	-
Sr-92	4.8 + 5	-	4.3 + 2	2.7 + 3	-	-
Y-92	5.2 + 5	-	4.7 + 2	3.0 + 3	-	-
Sr-93	5.8 + 5	-	5.2 + 2	3.3 + 3	-	-
Y-93	5.8 + 5	-	5.2 + 2	3.3 + 3	-	-

TABLE 3-1 (C[illegible])

Isotope	Liquid Source Activity (Ci) [1]	Gaseous Source Activity (Ci) [2]	Containment Sump Concentration (μCi/cc)	Reactor Coolant Concentration (μCi/cc)	Containment Airborne Concentration (μCi/cc)	Waste Gas Concentration (μCi/cc)
Mo-99	7.9 + 5	-	7.0 + 2	4.4 + 3	-	-
Tc-99m	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-
Ru-103	5.8 + 5	-	5.2 + 2	3.3 + 3	-	-
Rh-103m	5.8 + 5	-	5.2 + 2	3.2 + 3	-	-
Ru-106	2.2 + 5	-	1.9 + 2	1.2 + 3	-	-
Rh-106	2.2 + 5	-	1.9 + 2	1.2 + 3	-	-
Te-132	5.8 + 5	-	5.2 + 2	3.3 + 3	-	-
I-132	3.1 + 7	1.5 + 7	2.7 + 4	1.8 + 5	5.6 + 2	2.7 + 4
Te-134	8.4 + 5	-	7.5 + 2	4.8 + 3	-	-
I-134	4.7 + 7	2.3 + 7	4.2 + 4	2.6 + 5	8.5 + 2	4.0 + 4
Xe-138	7.9 + 7	7.9 + 7	7.0 + 4	4.4 + 5	2.9 + 3	3.4 + 6
Cs-138	7.9 + 5	-	7.0 + 2	4.4 + 3	-	-
Ba-140	7.3 + 5	-	6.6 + 2	4.2 + 3	-	-
La-140	7.9 + 5	-	7.0 + 2	4.4 + 3	-	-
Ce-143	6.3 + 5	-	5.6 + 2	3.6 + 3	-	-
Pr-143	6.3 + 5	-	5.6 + 2	3.6 + 3	-	-
Ce-144	4.8 + 5	-	4.3 + 2	2.7 + 3	-	-
Pr-144	6.3 + 5	-	5.6 + 2	3.6 + 3	-	-
Sr-91	4.8 + 5	-	4.3 + 2	2.7 + 3	-	-
Y-91m	-	-	-	-	-	-
Y-91	5.0 + 5	-	4.5 + 2	2.8 + 3	-	-
Zr-95	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-
Nb-95m	-	-	-	-	-	-
Nb-95	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-
Zr-97	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-

TABLE 3-1 (C[illegible])

Isotope	Liquid[1] Source Activity (Ci)	Gaseous[2] Source Activity (Ci)	Containment Sump Concentration (µCi/cc)	Reactor Coolant Concentration (µCi/cc)	Containment Airborne Concentration (µCi/cc)	Waste Gas Concentration (µCi/cc)
Nb-97m	-	-	-	-	-	-
Nb-97	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-
Ru-105	4.5 + 5	-	4.0 + 2	2.6 + 3	-	-
Rh-105m	4.5 + 5	-	4.0 + 2	2.6 + 3	-	-
Rh-105	2.9 + 5	-	2.8 + 2	1.6 + 3	-	-
Te-131	3.8 + 5	-	3.4 + 2	2.1 + 3	-	-
I-131	2.2 + 7	1.1 + 7	1.9 + 4	1.2 + 5	3.9 + 2	1.9 + 4
Xe-131m	3.4 + 5	3.4 + 5	3.0 + 2	1.9 + 3	1.2 + 1	5.8 + 3
I-133	4.1 + 7	2.0 + 7	3.7 + 4	2.3 + 5	7.4 + 2	3.5 + 4
Xe-133m	2.0 + 6	2.0 + 6	1.8 + 3	1.1 + 4	7.2 + 1	3.7 + 4
Xe-133	7.9 + 7	7.9 + 7	7.0 + 4	4.4 + 5	2.9 + 3	1.4 + 6
I-135	3.6 + 7	1.8 + 7	3.2 + 4	2.0 + 5	6.6 + 2	3.1 + 4
Xe-135m	2.2 + 7	2.2 + 7	2.0 + 4	1.2 + 5	8.0 + 2	9.1 + 5
Xe-135	1.5 + 7	1.5 + 7	1.4 + 4	8.6 + 4	5.5 + 2	2.6 + 5
Ba-141	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-
La-141	6.8 + 5	-	6.1 + 2	3.9 + 3	-	-
Ce-141	7.3 + 5	-	6.6 + 2	4.2 + 3	-	-

(1) Based on 100% noble gas core inventory, 50% halogen core inventory, and 1% of all others core inventory.

(2) Based on 100% noble gas core inventory and 25% halogen core inventory.

4.0 REVIEW OF AREAS REQUIRING ACCESS FOR POST-ACCIDENT OPERATIONS

4.1 AREA A: Hydrogen Recombiner Control Panel

Area "A" as identified on Figure 4-1 is the general location of the hydrogen recombiner control panel on the 253' elevation of the Intermediate Building. Post-accident hydrogen formation in the containment vessel may require operation of the hydrogen recombiner system. The system is started and controlled by operator action at the recombiner panel. This task would require that an individual spend approximately 30 minutes at the control panel.

Access to the control panel could be required shortly after an accident. The exposure rate at one hour after the accident in the general area of "A" would come from the direct dose from the Containment Building (2.2 R/hr) and from sample lines approximately 30 ft. away (0.95 R/hr) totaling about 3.1 R/hr. The estimated dose to an individual in this area for the necessary occupancy would be 1.6 rem. If occupancy was required at one day after the accident, the exposure rate is estimated to be 0.12 R/hr which results in a 30-minute dose of 0.06 rem.

The access route to area "A" is route "1" described in Table 4-1. The estimated dose to an individual as he traverses this route is 0.08 rem after at one hour the accident and is less than 0.005 rem at one day.

It should be noted that the Chemical Drain Tank, located in Area "A", could receive small quantities of discarded primary coolant samples from the Nuclear Sample Room sink under the existing sample disposal piping arrangement. The most significant quantity of sample fluid would be from the dissolved O_2 analysis which is a radiologically

impractical test to perform under the assumed accident conditions, and therefore was not addressed in this analysis (see Section 4.3).

While the total calculated dose from performing the described task is less than 3 rem shortly after the accident, recommendations for sample line and Nuclear Sample Room shielding in the sections which follow would provide further dose reduction, particularly during access to and away from this area.

4.2 AREA B: <u>Post-Accident Containment Air Sample Penetration No. 203</u>

Area "B" is the general location of containment penetration #203 in the Intermediate Building, 271' elevation (See igure 4-2). After an accident, a containment air sample could be collected at this location, in addition to alternative sampling penetrations which are later addressed in Areas "F" and "J".

Each post-accident containment air sample penetration contains a sample inlet line and a return line to the Containment Building. Both lines have manual valves locked closed. Upon opening the valves, a sample rig would be attached to the lines consisting of tubing, a gas collection bulb and a vacuum pump. The sample is collected by circulating containment air through the collection bulb for approximately 5 minutes. The time required to collect the air sample has been conservatively estimated to be 15 minutes.

The radiation level at one hour in Area "B" will result from the direct dose from the containment (9.5 R/hr), from the nuclear sample lines carrying liquid samples (3 ft. away 60 R/hr), from the 35 ml gas sample bulb (1 ft. away - 1.2 R/hr) and the containment air sample lines (3 ft. distance - 0.2 R/hr). The summation of these various radiation sources (at one hour after the accident) is 70 R/hr. The

resulting dose is estimated to be 17.8 rem for collecting an air sample at this location. These radiation levels drop significantly with time (See Table 4-2). The exposure rate decreases to 7.8 R/hr in one day.

The access route to Area "B" is designated as route #2 in Table 4-2, and would entail some exposure contributed by direct radiation from the Containment Building, sample piping and sample coolers located in the Nuclear Sample Room. This access route is not anticipated to add significantly to the radiation dose of the individual collecting the sample (0.08 rem). The gas sample would be placed in a shielded container for transport to the Primary Chemistry Laboratory.

The dose to an individual performing this task could be reduced by several means. First, because of the high specific activity, the volume of the air sample collection container can be decreased. Lead shielding can also be used around the sample container; one inch of lead around the sample would decrease the radiation levels at one hour from 1.2 R/hr to 0.32 R/hr, a reduction factor of about 4. The 15-minute exposure period assumed for performing this task is likely to be conservative by as much as a factor of 2, and considers the fact that work would be performed in respiratory protection equipment. The individual taking the sample could readily leave the high dose rate area while the sample was being collected, as is normal practice. Therefore, utilizing a reduced sample size, sample container shielding and shortened stay-time, this task could be accomplished with a dose of approximately 5 rem.

It is important to note, however, that under the accident conditions assumed in this analysis, priority should be given to the collection

and analysis of a primary coolant sample and to onsite and offsite radiological monitoring efforts. High airborne levels inside containment would be deduced from containment area radiation and airborne radioactivity monitors. Therefore, containment air sampling, while important, would be less critical after the onset of an accident than the other types of sampling. Should a containment air sample be deemed necessary, alternative sampling locations are available in the plant, and could afford further dose reduction [Area "F"].

4.3 AREA C: Nuclear Sample Room

The Nuclear Sample Room, Intermediate Building, elev. 271 (Figure 4-2) has been designated as Area "C". Primary system liquid and gas samples would be needed following an accident to assess the extent of fuel damage and to evaluate primary coolant chemistry. Potential sample sources could include: The "B" primary loop hot leg; the pressurizer liquid and steam spaces; and the residual heat removal loop. These samples are collected under a ventilated hood in the Nuclear Sample Room. Liquid sample lines discharge to a sample sink for collection in a suitable sample collection bottle, and gas samples are collected in a stainless steel pressure vessel which is installed with quick-disconnect fittings. Samples would be collected and transferred to the adjacent Primary Chemistry Laboratory through a passbox.

The sample size for normal operations is 250 ml. Under the accident conditions being considered, the sample size will be reduced significantly. However, shielding of the sample will be required to

minimize the radiation dose to workers. This evaluation assumes the first primary system sample will be taken within one hour after an accident, in accordance with NRC guidelines. At this time, a 75 to 100 ml. sample (75 ml. indicated henceforth) gives an exposure rate at 1 ft. from the container of 600 R/hr. With 4 inches of lead shielding, the exposure rate is reduced to 1.4 R/hr. Sample lines in the area contribute 60 R/hr three feet away at one hour following the accident. In addition, the dose rate from a contaminated sample cooler (located behind a 6-inch concrete shield wall in the Nuclear Sample Room) would be 30 R/hr. It has been estimated that sample collection takes a worker 6 minutes resulting in 70 rem with no shielding or 9 rems when the sample bulb is shielded. The same operation one day after the accident will result in 8.7 and 1.1 rem, for samples without shielding and with shielding, respectively. Since access to the Nuclear Sample Room may be required within one hour after an accident, consideration must be given not only to shielding the sample collection vessel but also the sample lines, and providing additional shielding around the sample coolers. One inch of lead shielding will reduce radiation levels from exposed sample piping and the coolers by a factor of 7 thus resulting in acceptable personnel doses. Actual piping locations and available space in the nuclear sample room need to be reviewed to utilize shielding to the extent practicable.

Dissolved oxygen analysis is currently performed using a leuco reagent color test, under the ventilated hood in the Nuclear Sample Room. Although the test for oxygen could be performed following the accident, several radiological considerations would argue against attempting the analysis in this manner shortly after an accident. First, it is

unlikely that significant dissolved oxygen levels will occur in the coolant due to the presence of hydrogen. Second, the leuco test procedure involves relatively large volumes of primary sample which cannot be significantly diluted or reduced. This could involve significantly higher direct radiation dose rates from the sample bottle (greater than 1000 R/hr at 1 ft.) and disposal problems from the flushed sample liquid into the sample sink. In addition, significant quantities of primary gas will evolve using this method and could aggravate airborne radioactivity levels in surrounding work areas. It is recommended therefore that an alternative means be investigated for performing dissolved oxygen analysis.

The access route to the Nuclear Sample Room, Route #2 (See Table 4-1) will add approximately 0.08 rem to the worker dose if the task is done during the first hour after the accident. Utilization of this access route at later times will not add significantly to worker dose.

It is seen therefore that sampling of primary coolant can be performed within acceptable dose levels, if measures are taken to provide shielding around the sample container, sample lines and sample coolers inside the Nuclear Sample Room. An alternative approach to adding shielding to the sample coolers could be eventual relocation in a more remote or better shielded portion of the plant.

4.4 AREA D: Primary Chemistry Laboratory

The Chemistry Laboratory in the Service Building has been designated as Area "D". Radiological and chemical analyses are performed in the Primary Chemistry Laboratory once samples are transferred from the Nuclear Sample Room. A series of procedures are then utilized

for drawing liquid and gas aliquots for counting, and for performing additional required tests including boron, pH, conductivity, hydrogen and other trace chemicals.

Occupancy in this area is assumed to be primarily frequent after an accident. Radiation levels in Area "D" are primarily attributed to sample lines in the Nuclear Sample Room. A one foot shield wall separates Areas "C" and "D". Radiation levels next to the wall separating these areas will read about 0.2 R/hr within one hour after the accident, 0.03 R/hr after one day, and about 0.002 R/hr after two days. General areas in the Chemistry Laboratory will read about .03 R/hr after one hour, about 0.004 R/hr after one day and about 0.002 R/hr after two days.

4.4.1 Isotopic Analysis

Primary coolant sample processing results in high localized doses for the workers involved. (See 7.0 Appendix for Sample Preparation Time Study.) Handling the 75 ml. primary coolant sample and subsequent diluted samples results in 6.7 rem (40 seconds exposure to the unshielded 75 ml. primary coolant sample) and 0.2 rem (from handling 1 ml. of the primary coolant sample diluted to 50 ml.).

The 75 ml. sample could be shielded while in the sample hood. The only unshielded time would be during the transfer from the passbox to the sample hood. This transfer could be done in 10 sec. using an extension tool to keep the sample vessel at least 6 feet from the worker. Under these conditions, the exposure rate from the sample vessel would be 19 R/hr at one hour after the accident resulting in a dose to the worker of less than 0.1 rem. In the radio-

chemistry sample hood with four inches of lead, the sample vessel will read 1.4 R/hr and, therefore, result in about 0.01 rem to the worker. The total dose to the worker for processing the sample for isotopic analysis, utilizing shielding to the extent possible, is about 0.3 rem.

4.4.2 Boron Analysis

For boron analysis the 75 ml. primary coolant sample would be shielded while in the sample hood. By providing shielding around the titration rig, the total dose to the worker should not exceed the dose of 0.3 rem anticipated for the isotopic analysis. (See 7.0 Appendix on Sample Preparation Time Study.)

4.4.3 Primary Gas Analysis

The primary gas analysis involves a transfer rig mounted on the wall next to the sample passbox. (See 7.0 Appendix on Sample Preparation Time Study.) Gases from the 75 ml. primary coolant sample vessel are drawn into a gas collection container from which samples are taken for hydrogen and isotopic analyses. As indicated previously the radiation level from the 75 ml. primary coolant sample vessel is about 600 R/hr at 1 foot at one hour after the accident. In reviewing the procedures, it has been determined that an individual would be located approximately an arm's length from the sample for about 4 minutes which would results in a dose to that worker of 40 rem. Utilizing a shadow shield of 2 inches of lead, the exposure rate would be reduced to 20 R/hr and the dose to the worker to about 1.3 rems. If this task were done at one day post accident, the exposure level, using the 2 inch lead shield, would be 2.7 R/hr and the dose to the worker about 0.2 rem.

Although it is apparent that the required handling and analysis of samples can be performed in the Primary Chemistry Laboratory within dose guidelines, several procedural and shielding modifications should be considered. Recommended shielding additions to the sample lines and coolers in the Nuclear Room will also reduce dose levels considerably in the Primary Chemistry Laboratory. Shielded sample containers are also necessary, both for temporary sample storage during procressing and for receiving discarded samples. Disposal of samples into the sample sink, as is normal practice, may pose immediate airborne contamination problems in the Chemistry Laboratory. All handling of open sample containers should be done behind shielding under the Laboratory's ventilation hood. This would also apply to use of the gas partitioner (primary H_2 analysis) as some volume of the injected sample will diffuse into the Chemistry Laboratory atmosphere. Portable shadow shields should be considered for work in close proximity to any exposed sample container (e.g., primary gas sample bomb, boron analysis).

4.5 AREA E: Count Room

The Count Room, designated as Area "E", would be primarily affected by post-accident radiation sources in the adjacent Nuclear Sample Room. Two feet of concrete shielding separate these two rooms. The Count Room is equipped with a gamma pulse-height analyzer system and counting safe for isotopic analysis of primary liquid and gas samples, and plant effluent releases. Additional counting equipment is also available in the Count Room (e.g., GM and proportional counters) for analyzing general plant samples for airborne and surface contamination.

Radiation levels from sample lines and sample coolers in the adjacent Nuclear Sample Room would approach 0.09 R/hr one hour after the accident within three feet of the shield wall and about 0.01 R/hr in general areas within the Count Room. A relatively minor dose rate contribution, 0.002 R/hr, would be caused by direct radiation from the Containment Building. By the end of the first day after an accident, these levels are decreased by over a factor of 10.

A combination of two alternatives should be considered for post-accident in-plant sample counting. First, provisions should be made to mobilize count room equipment in the event actual direct radiation background and airborne contamination levels prohibit adequate counting capability. There appear to be no major problems in accomplishing this as Ginna Station. Most of the count room equipment could be relocated in a lower background area onsite (e.g., warehouse) within the first two hours following an accident. In addition, a trailer now equipped for routine environmental counting, could be utilized if necessary for sample analysis.

In addition, recommended shielding of Nuclear Sample Room sample lines and coolers will reduce dose rates in the Count Room. Although certain sensitive counting equipment may still be adversely affected by elevated background levels, the 4-inch thick counting safe housing the GeLi detector would allow gamma isotopic analysis to be performed.

4.6 AREA F: Post-Accident Containment Air Sample Penetration No. 305

Area "F" is in proximity to post-accident containment air sample lines at Penetration 305 located on the 253 ft. elevation of the Intermediate Building (See Figure 4-1). For Area "F" liquid sample

lines will not be a problem and the direct dose from containment will be less since this penetration is below the operating floor. Table 4-2 gives the dose rates in this area as a function of time. As indicated for Area "B", the task of collecting an air sample from containment is conservatively assumed to require about 15 minutes of worker exposure time. Without shielding the gas sample bulb, the worker would receive about 0.9 rems collecting the sample. By using one inch of lead around the sample collection bomb, the dose to the worker would be reduced to 0.7 rems. Reduction in the sample size and of the length of the sampling lines can also be used to decrease the dose to the worker.

The access route utilized to get to Area "F" will be #4 as described in Table 4-1. The radiation dose accessing Area "F" will be small (0.01 rem), provided the sample collection vessel is adequately shielded.

4.7 AREA G: Radwaste Control Panel

Area "G", the radwaste control panel, is located on the basement floor of the Auxiliary Building (See Figure 4-3). Immediate post-accident access to the waste disposal panel and gas analyzer is not required; however, certain longer term actions may make access necessary to this area. The waste disposal panel contains pressure gauges for the tanks using cover gas and also for the gas decay tanks and vent header. Alarms for tank and vent header pressure and gas analyzer oxygen are locally indicated with a waste disposal panel alarm given on the Main Control Board. All gas system manual operations and releases are controlled locally at the waste panel. In addition, various equipment associated with the liquid waste system are manually controlled at this location.

Before recirculation through the RHR pumps (at least 30 minutes after the accident) the radiation level in Area "G" will be background. During recirculation, RHR piping, safety injection pumps and pipes and containment spray pumps and pipes will make radiation levels on the basement floor of the Auxiliary Building very high.

It is anticipated than when occupancy is required at the radwaste control panel, an individual would spend 2 minutes in this area once per shift. From Table 4-2, the radiation levels at this location are estimated to be 3,000 R/hr one hour after the accident. A two-minute occupancy in this area would result in a 100 rem dose. After one, three and 10 days, the dose to a worker in two minutes would be 10 rems, 5 rems and 2 rems, respectively.

Access to the radwaste control panel is via route #5 (Table 4-1). The dose received along this route is estimated to be 0.35 rem during the first hour of the accident and less than 30 mrem 1 day following the accident.

Because of the magnitude of these levels and the extent to which shielding would be required to reduce these levels, it appears probable that a modification to the plant procedures and controls should be investigated to transfer certain radwaste equipment indications and control functions from the radwaste control panel to another location in the plant.

4.8 AREA H: Safeguards Bus 16

Area "H" is on the intermediate level of the Auxiliary Building around Safeguard Bus #16 (Figure 4-4). It is not expected that any immediate access would be required to the areas of the safeguards buses or their associated motor control centers.

Radiation levels in Area "H" will be from the Safety Injection pumps and associated piping directly below on the basement floor. The shine through the 18" concrete floor has been estimated to be about 60 R/hr at one hour after the accident. This radiation level drops to 2.4 R/hr at one day and 0.4 R/hr at 10 days (See Table 4-2).

In the event the CVCS system is used to strip gas from the primary coolant, the waste gas piping from the volume control tank to the gas compressor and into the waste gas decay tanks will be contaminated. The waste gas decay tanks, once filled with accident source activity, will also contribute to the radiation level in Area "H", although to a lesser extent.

Radiation levels 30 feet from the 1" waste gas piping have been estimated to be 100 R/hr at one hour after the accident. After one day, these levels decrease to 5 R/hr, approximately double the radiation level coming from the floor below. The waste gas decay tanks have been estimated to contribute 2.8 R/hr to Area "H" at one hour into the accident and 0.09 R/hr after one day.

The waste gas piping is one inch in diameter and could be locally shielded. One inch of lead shielding would decrease the radiation level by a factor of about 4, and 2 inches of lead by a factor of 10.

Access to Area "H" at one hour after an accident is assumed to be via route #7 (Table 4-1) and will result in an additional 4.4 rem for ingressing and egressing the area. The majority of this dose comes from the waste gas piping (4 rem). If no waste gas stripping is in progress, then the access dose would be only 0.4 rem. If the waste gas piping is contaminated with stripped gases, shielding with the equivalent of 1" of lead on waste gas piping and 1/2" of lead on RHR sample lines results in a dose to individuals accessing Area "H" of just over 1.5 rem. If access to this area is required at 1 day post-accident, then the radiation dose for ingress and egress would be 0.14 rem and 0.007 rem for unshielded and shielded waste gas and RHR sample piping, respectively.

4.9 Area I: <u>Safeguards Bus 14</u>

The safeguards bus #14 is on the 271' elevation. It has been designated as Area "I" (Figure 4-5) and is directly over Area "H". Occupancy in this area will be the same as explained for Area "H".

The radiation level in Area "I" will be essentially from the direct dose from containment. At one hour, the radiation level has been estimated to be about 5 R/hr, after one shift 0.7 R/hr and only 0.04 R/hr after one day.

Access to Area "I" is via route #6 (Table 4-1) and will add 0.35 rem to the worker's dose at one hour following an accident and only 0.005 rem after one day.

4.10 AREA J: <u>Post-Accident Containment Air Sample Penetration</u>

Area "J", defined in Figure 4-4, provides another location where a containment air sample can be collected. The task performed at this

location would be the same as in Areas "B" and "F". Area "J" radiation level will be from the direct dose from containment and from two 6" diameter safety injection pipes running several feet below the floor slab.

The exposure rate from these sources is estimated to be 35 R/hr at one hour post accident dropping to 1.2 R/hr in one day. The 35 ml. sample vessel will not add a significant amount to the worker's dose. At one hour, the sample vessel will be reading about 1R/hr and at 1 day 0.13 R/hr. The dose to the worker at 1 hour after an accident (assuming 15 minutes occupancy) would be about 9 rem, at one day about 0.3 rem and at 30 days, 0.01 rem.

Access to Area "J" will be via route #8 (Table 4-1) and will add 3 rem to personnel doses at one hour after the accident (waste gas piping contaminated and not shielded; if not contaminated, then the dose would be 0.4 rem. One inch of lead shielding around contaminated waste gas piping and 1/2 inch lead shielding on RHR sampling lines would reduce the access dose to 1 rem. Access doses at 1 day would be 0.1 rem if waste gas stripping is in progress and is negligible without gas stripping.

It is clear that access to Area "J" for containment air sampling shortly after an accident would be less preferable due to high direct radiation levels and probable airborne contamination than would be Area "F". Nevertheless, the area could be accessed after 1 day with proper coordination of waste gas stripping operations to minimize dose. Also, sample container shielding recommendations previously made for sample transport should also be considered.

4.11 AREA K: Auxiliary Building HVAC Filters

Area "K", the Auxiliary Building HVAC filters, 253 ft. elevation (Figure 4-4) will have the same exposure rates as Area "H".

Should access be required in Area "K" to change filters, it has been estimated that it would take approximately 20 man-hours to complete this task. Table 4-3 provides area dose rates and access doses similar to those calculated for Area "H" following a postulated accident. At one hour following the onset of the accident, dose rates would be prohibitive for the filter change task to be completed. However, it is likely that such a task would not be undertaken until later recovery operations were begun. If it is assumed that Area "K" filters are changed 10 days following the accident, then direct radiation levels would result in personnel exposures of less than 5 rem (assuming 4 workers available for this task). This also accounts for the likelihood that the contaminated filters themselves would be a direct source of radiation, on the order of 1 R/hr (ref. 5).

Several means are available to reduce exposure in the event such a task needs to be performed. For instance, new filters can be unboxed in a lower dose area (Aux. Bldg. operating floor) and lowered to the 253 ft. elevation by crane through the floor access hatch. Also, spent filters could be shielded with lead blankets and left on the 253 ft. elevation for decay prior to further handling or removal. Modifications to waste gas piping involving shielding or relocation could also further reduce worker dose in the event primary coolant degassing operations are coincident with filter changes.

4.12 AREAS L AND M: Spent Fuel Pool and Auxiliary Building HVAC Filters

Areas "L" and "M" are filter units for the Spent Fuel Pool and Auxiliary Building HVAC systems (Figures 4-5 and 4-2, respectively). These areas will be affected by radiation levels from containment and from sample lines running along the west and south walls. It is not anticipated that access will be required in these areas immediately after an accident, but should conditions dictate, access may be needed to change out filters.

The exposure rates for Areas "L" and "M" are provided in Table 4-2 as a function of time following the assumed accident. Access to these areas is via route #2 (Table 4-1) and would result in an approximate 0.08 rem dose one hour after an accident. After one day the access dose becomes negligible (less than 0.01 rem).

If one assumes the same exposure times used for Area "K" (e.g. 20 man-hours at 10 days following an accident), both filter change tasks would be able to be performed within dose levels of less than 5 rem.

4.13 AREA N

Area "N", as identified on Figure 4-1, includes the area for the Control Access HEPA and charcoal filters on the 352 ft. elevation. Access would be required in this area only to change filters, and would be via route #9 (Table 4-1). The radiation level in Area "N" will be the summation of the direct dose from the containment and from primary coolant sample lines approximately 30 feet away.
The radiation exposure rates are given in Table 4.2 as a function of time. The dose rate in this area decreases from 1.7 R/hr after the

first hour to slightly more than 0.1 R/hr one day following an accident. It is seen that even with an assumed task performance time of 20 man-hours, the task can be accomplished within acceptable dose guidelines after one day.

4.14 CONTROL ROOM

Continuous occupancy is required in the Control Room after an accident. The Control Room is located on the 289 ft. elevation as shown in Figure 4-6. The direct radiation from the Containment Building will result in radiation levels in the Control Room of approximately 0.018 R/hr at one hour after an accident, quickly dropping off to 0.008 R/hr at 4 hours and to near background levels shortly thereafter. The integrated dose (6 months) to operators in the control room (assuming 100% occupancy) due to direct radiation from containment has been calculated to be 0.2 rems. The majority of this dose comes within a few hours after the accident.

TABLE 4-1

ACCESS ROUTES TO VITAL AREAS

	Access Route	Description
1.	To H_2 Recombiner Panel	From Service Building (S.B.), 271 elev. to Intermediate Building (I.B.) South Side; down staircase next to personnel hatch to the 253' elev.
2.	To Air Sample Penetration 203 and Nuclear Sample Rm.	From S.B. 271 elev. to I.B. then south by nuclear sample room and containment penetrations.
3.	Primary Chemistry Lab. and Count Room	In S.B.
4.	Air Sample Penetration 305	S.B. 271 elev. to Turbine Building (T.B.) to staircase in NW I.B. down to 253' elev. T.B. East to double door to I.B.
5.	Radwaste Panel	S.B., 271 elev. to I.B. South by Nuclear Sample Room, along SFP to Auxiliary Building (A.B.) to East staircase down to 236 elev.
6.	Bus 14	Same as "5" but stays on 271' to column lines 8a-9a and L-N.
7.	Bus 16 and HVAC Filters	Same as "5" but utilizes West staircase in A.B. to the 253 elev. to column lines 8a-9a, L-N.
8.	Air Sample Penetration (behind RWST)	Same as "7" except to column lines 7a-8a, L-N.
9.	Control Access Filters	Same as "2" but utilizes stairs in SW I.B. to go to the 253 ft. elevation.

TABLE 4-2
EXPOSURE RATES FOR VITAL
AREAS AS A FUNCTION OF TIME
EXPOSURE RATE (R/hr)

VITAL AREA		1 HOUR	1 DAY	30 DAYS	6 MONTHS
A		3.1	0.12	0.01	(1)
B	(unshielded air sample)	71	7.8	0.47	(1)
	(shielded air sample 1" Pb)	70	7.7	0.46	(1)
C	(unshielded liquid sample)	690	87	5.8	1.1
	(shielded liquid sample)	91	1.1	0.48	(1)
D	(next to wall)	0.22	0.03	(2)	(1)
	(general area)	0.03	0.004	(2)	(1)
E	(hear wall)	0.09	0.003	(2)	(1)
	(general area)	0.01	(2)	(2)	(1)
F	(unshielded air sample)	3.6	0.17	(2)	(1)
	(shielded air sample 1" Pb)	2.7	0.07	(2)	(1)
G		3100	340	21	4.4
	(excluding waste gas system)	60	2.4	0.16	(1)
	(including waste gas system - unshielded pipes)	180	7.4	0.28	(1)
	(including waste gas system - 1" Pb on pipes)	93	2.7	0.14	(1)
I		5.4	0.04	(1)	(1)
J	(unshielded air sample)	35	2.6	0.10	(1)
	(shielded air sample Pb)	34	2.4	0.10	(1)
K	(excluding waste gas system)	60	2.4	0.16	(1)
	(including waste gas system - unshielded pipes)	180	7.4	0.28	(1)
	(including waste gas system - 1" Pb on pipes)	93	2.7	0.14	(1)
L & M		8.4	0.11	0.004	(1)
N		1.7	0.11	0.01	(1)
Control Room		0.018	(2)	(2)	(2)

Notes: 1) not calculated

2) negligible (<0.001 R/hr)

TABLE 4-3

VITAL AREA RADIATION DOSE SUMMARY

Area	Time After Accident	Occupancy	Dose In Area (rem)	Access Dose To And From (rem)	Total (rem)
A	1 hr.	30 min.	1.6	0.08	1.7
	1 day		0.06	neg.	0.06
B	1 hr.	15 min.	17.8[1]	0.08	17.9
	1 day		2.0[1]	neg.	2.0
C	1 hr.	6 min.	9.0[1]	0.08	9.1
	1 day		1.1[1]	neg.	1.1
D[3]	1 hr.	Frequent	0.03[2]	0.05	N/A
	1 day		.004[2]	neg.	N/A
E	1 hr.	Frequent	.01[2]	0.05	N/A
	1 day		.003[2]	neg.	N/A
F	1 hr.	15 min.	0.7[1]	0.01	0.71
	1 day		0.07[1]	neg.	0.07
G	1 hr.	2 min.	100	0.35	100
	1 day		10	<0.03	10
H	1 hr.	- -	60[2]	0.4[4]	N/A
	1 day		2.4[2]	neg[4]	N/A
I.	1 hr.	- -	5.4[2]	0.4	N/A
	1 day		0.04[2]	neg	N/A
J	1 hr.	15 min.	8.8[1]	0.4[4]	9.2
	1 day		0.3[1]	neg[4]	0.3
K	1 hr	- -	60[2]	0.4[4]	N/A
	1 day		2.4[2]	neg[4]	N/A
L&M	1 hr	- -	8.4[2]	0.08	N/A
	1 day		0.1[2]	neg.	N/A

TABLE 4-3 (Cont.)

Area	Time After Accident	Occupancy	Dose In Area (rem)	Access Dose To And From (rem)	Total (rem)
N	1 hr.		1.7^2	0.08	N/A
	1 day		0.1^2	neg.	N/A
Control Room	1 hr.	Continuous	0.018^2	neg.	N/A
	4 hr.		0.008^2	neg.	N/A
	>8 hr.		Bkg^2	neg.	N/A

1 - Shielding utilized around sample vessel.

2 - General area dose rates (R/hr.).

3 - General areas in Chemistry Laboratory. See Sections 4.4.1, 4.4.2, and 4.4.3 and Appendix (7.0) for sample processing.

4 - No waste gas stripping in progress.

5.0 REVIEW OF RADIATION QUALIFICATION OF SAFETY EQUIPMENT

5.1 IDENTIFICATION AND EVALUATION OF EQUIPMENT

Table 5-1 identifies the location of the safety-related equipment considered in the review. The equipment considered is consistent with SEP environmental qualification and safe shutdown reviews previously submitted to the Commission. Table 5-1 also presents the required time each item of equipment has to be capable of performing its safety function; the integrated dose for each given time period and location, and the radiation qualification of the equipment. Integrated doses were not calculated for components which perform their safety function prior to post-accident recirculation or which are located in areas remote from the systems which may contain radioactive material after an accident. These components would not have any substantial increase in integrated exposure as a result of the NUREG-0578 requirements.

5.2 RECOMMENDATIONS

A comparison of the calculated and qualified integrated doses in Table 5-1 indicates several components which could potentially be exposed to high radiation levels that do not have qualification values available. These include CV pressure transmitters (PT-945 and 946) and the RHR, containment spray, and safety injection pumps. Evaluation of these components should be made to determine the appropriate qualified radiation dose.

TABLE

R. E. GINNA

Equipment Qualification

Equipment Name	Identification Number	Location			Time After Accident Safety Function Required	Required Operating Duration	Calculated Integrated Dose (Rads)	Qualified Integrated Dose (Rads) (ref. 6)
		Building	Elev.	Col. Lines				
NaOH Addition Valves	V836 A, B	Aux.	236	$8A+9A, L+N_1$	$\leq$ 2 min.	30 min.	(2)	N/A (5)
Feedwater Isolation Valves	V4269, 4270	Turbine	271	4+5, F+E	5 sec.	5 sec.	(2)	N/A
Main Steam Isolation Valves	V3516, 3517	IB	278	$4+6, F+F_5$	5 sec.	5 sec.	(2)	N/A
BA Supply to SI Pumps	V826 A, B, C, D	Aux.	252	$9A+10A, L+N_1$	$\leq$ 1 min.	$\sim$10 min.	(2)	N/A
RWST Supply to SI Pumps	V825 A, B	Aux.	236	$8A+9A, L+N_1$	$\leq$ 10 min.	$\sim$10 min.	(2)	N/A
RWST Isolation Valves	V896 A, B	Aux.	236	$8A+9A, L+N_1$	$\leq$ 30 min.	30 min.	(2)	N/A
Aux. Feedwater to S/G	V4007, 4008	IB	256	3+4, G+H	$\leq$ 1 min.	LT	(3)	2.0 + 8
Service Water to Aux. FW Pumps	V4027, 4028	IB	256	3+4, G+H	$\leq$ 1 min.	LT	(3)	2.0 + 8
CV Sump Isolation Valve	V850 A, B	Aux.	219	$5A+6A, N+N_1$	$\leq$ 30 min.	LT (1)	2.8 + 6	2.0 + 8
RHR Pump Discharge	V857 A, B, C	Aux.	236	$8A+9A, L+N_1$	$\leq$ 30 min.	LT (1)	2.8 + 6	2.0 + 8
Containment Spray Discharge	V860 A, B, C, D	Aux.	236	$8A+9A, L+N_1$	$\leq$ 1 min.	LT (1)	2.8 + 6	2.0 + 8
RWST to RHR Suction	V856	Aux.	236	$6A+7A, N+N_1$	$\leq$ 1 min.	LT (1)	4.8 + 4	2.0 + 8
Standby Aux. FW Pumps	IC + ID	Aux. Add	272	9A+11A, Q+R	$\leq$ 10 min.	LT	(3)	N/A
RHR Pumps	1A + 1B	Aux.	219	$5A+6A, N+N_1$	$\leq$ 1 min.	LT (1)	2.8 + 6	N/A
Containment Spray Pumps	1A + 1B	Aux.	236	$8A+9A, L+N_1$	$\leq$ 1 min.	LT (1)	2.8 + 6	N/A
CCW Pumps	1A + 1B	Aux.	271	6A+8A, Q+N	Continuous Operation	LT	(3)	N/A
Aux. FW Pumps	1A + 1B	IB	252	3+4, G+H	$\leq$ 1 min.	LT	(3)	N/A
SI Pumps	1A, 1B, 1C	Aux.	236	$8A+9A, L+N_1$	$\leq$ 30 sec.	LT (1)	2.8 + 6	N/A
Service Water Pumps	1A, 1B, 1C, 1D	Screen House	253	5+6, A+C	Continuous Operation	LT	(3)	N/A
CV Press. Transmitter	PT - 947, 948	IB	272	3+4, J+K	Continuous Operation	LT	(3)	N/A
CV Press. Transmitter	PT - 945, 946	Aux.	256	5A+7A, L+N	Continuous Operation	LT (1)	<1.0 + 3	N/A
CV Press. Transmitter	PT - 949, 950	IB	256	4+5, G+H	Continuous Operation	LT	(3)	N/A
RWST Level	LT - 920, LIC - 921	Aux.	236	$7A+8A, L_1+N_1$	Continuous Operation	30 min.	(2)	3.0 + 4
BAST Level Transmitter	LT - 102, 106, 171, 172	Aux.	272	$9A+11A, L+N_1$	Continuous Operation	$\sim$10 min.	(2)	3.0 + 4
A - Steamline Press. Transm.	PT - 468, 469, 482	IB	256	$5+6, F_5+F_1$	Continuous Operation	LT	(3)	3.0 + 4
B - Steamline Press. Transm.	PT - 478, 479, 483	IB	253	3+4, G+H	Continuous Operation	LT	(3)	3.0 + 4
Batteries	A + B	Control	252	$11+13, F+G_2$	Continuous Operation	LT	(3)	N/A

TABLE 5-1

Equipment Name	Identification Number	Building	Elev.	Col. Lines	Time After Accident Safety Function Required	Required Operating Duration	Calculated Integrated Dose (Rads)	Qualified Integrated Dose (Rads)
Diesel Generators	A + B	DG Bldg.	252	11+13, $A+A_2$	< 30 sec.	LT	(3)	N/A
I and C Cabinets	-	Control	272	12+13, F+B	Continuous Operation	LT	(3)	N/A
Reactor Trip Breakers	-	IB	256	6+7, F+G	≤ 30 sec.	seconds	(3)	N/A
Safeguard Bus	16	Aux.	256	8a+9a, $L+N_1$	Continuous Operation	LT (1)	<1.0 + 3	N/A
Safeguard Bus	14	Aux.	272	8a+9a, $L+N_1$	Continuous Operation	LT (1)	1.0 + 2	N/A

1) Long term integrated exposure is calculated for a period of six months considering direct gamma radiation.

2) Safety function performed prior to post-accident recirculation phase, therefore, no increase in total dose from present considerations.

3) Located in area with a negligible increase in total dose from present considerations.

4) Dose calculated for time period 0.5 hours to 24 hours.

5) N/A - not available

6.0 REFERENCES

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2. E. D. Arnold and B. F. Maskervitz, "SDC - A Shield Design Calculation Code for Fuel Handling Facilities,"ORNL-3041, March 1966.

3. C. M. Lederer, J. M. Hollander, and J. Perlman, "Table of Isotopes," Sixth Edition, John Wiley and Sons, Inc., 1967.

4. A. Tobias, "Data for the Calculatoin of Gamma Radiation Spectra and Beta Hearing from Fission Products (Revision 3), RD/B/M2669, June 1973.

5. Personal communication, E. E. Gutwein to C. Montgomery, Gilbert Associates, Inc., December 18, 1979.

6. Letter from L. D. White, Jr. to D. L. Ziemann, "Environmental Qualification of Electrical Equipment R. E. Ginna Nuclear Power Plant", December 1, 1978.

7.0 APPENDIX 1: LIQUID AND GASEOUS SAMPLE COLLECTION AND PREPARATION

7.1 SAMPLE COLLECTION: NUCLEAR SAMPLE ROOM (RG&E Procedure S-5)

7.1.1 Liquid (O_2 and Radionuclide) Sample Collection

Task	Time	Worker Location
o Set up Purge System	2 min.	1
o Purge	15 min.	2
o Sample flow to sample sink	30 sec.	1
o Sample flush to BOD bottle (O_2 Leuco test)	5 min.	2
o Sample for radioactivity analysis, 100 ml. (dumps bottle, rinse, fill bottle)	2 min.	1
o Place sample in pass box	30 sec.	1
o Realign and close valves	1 min.	1
TOTALS	6 min.	1
	20 min.	2

7.1.2 H_2 and Radioactive Sample Collection

Task	Time	Worker Location
o Vent and Drain System		
o Insert Samples - Sample bomb in quick release fittings	.5 min.	1
o Open isolation valves		
o Start Purge		
o Purge	15 min.	2
o Close isolation valves		
o Vent and Drain System to Sink	3 min.	1
o Remove sample bomb		
o Transfer Sample Bomb to Pass Window	15 min.	1
o Realign Valves	1 min.	1
TOTALS	9.25	1
	15.0	2

7.2 CONTAINMENT AIR SAMPLE COLLECTION: I.B. 253 ELEV., NORTH SIDE (RG&E PROCEDURE RD-1.1)

Task	Time	Worker Location
o Line up valves and connect 35 cc sample collection bulb	2 min.	3
o Purge sample through container	5 min.	4
o Realign valves and disconnect sample container	1 min.	3
o Transfer sample to shielded cask	30 sec.	3
o Carry shielded sample back to the H.P. Lab. and Count Room	2 min.	5
o Transfer sample to count safe	30 sec.	3
o Count Sample	200 sec.	5

7.3 SAMPLE PREPARATION (PRIMARY CHEMISTRY LABORATORY AND COUNT ROOM)

7.3.1 Isotopic Analysis of Primary Coolant (RG&E Procedure PC-5)

Task	Time	Worker Location
o Remove 100 ml sample from pass box to hood; transfer 20-30 mls of 100 ml sample to beaker and add acid and boiling beads, place remaining 70 to 80 ml sample behind lead shield. GAI to specify amount of shielding required - curve showing mR/hr vs. Pb thickness	30 sec.	3
o Boil Sample; Cool in Water Bath	10 sec. 7 min.	3 4
o Pipe 1 ml to 50 ml volumetric bottle and dilute to 50 mls and mix.	1 min.	3
o Transfer 50 ml. diluted sample to count room and place in steel counting safe. Note: 4" of steel surround detector; count room has 2 ft. shield walls.	30 sec.	3

Task	Time	Worker Location
o Count Sample-(If conditions in the count room are such that have a high background or airborne contamination, the counting trailer for environmental samples can be used.) The sample is counted 15 minutes after collection and again 1 week later.	200 sec.	5

7.3.2 Boron Analysis In Primary Chemistry Laboratory (RG&E Procedure PC-14)

Task	Time	Worker Location
o 10 ml of primary coolant sample is placed in a beaker and moved to the titration rig.	30 sec.	3
o Mannitol is added to the beaker	10 sec.	3
o Automatic titration of NaOH to sample along with pH determination	2 min.	4
o Presently sample is dumped down sink in Chemistry Laboratory presenting potential airborne problems for accident level specific activities.		

7.3.3 Primary Gas Analysis In Primary Chemistry Laboratory (RG&E Procedure PC-4)

Task	Time	Worker Location
o Connect 100 cc steel sample bomb in transfer rig mounted on wall next to sample pass window. Transfer bomb is 300 cc glass cylinder with valves.	2 min.	3
o Transfer 100 cc sample to 300 cc glass collection bulb (purge)	10 min.	4
o Technician checks transfer operation.	5 sec.	3

Task	Time	Worker Location
o With syringe remove 5 cc from glass collection bulbs to counting vial.	30 sec.	3
o Transfer vial to Count Room and place in counting safe.	30 sec.	3
o Count Sample	200 sec.	5
o Remove liquid from transfer rig lines.	1 min.	3

7.3.4 Hydrogen Analysis In Primary Chemistry Laboratory (RG&E Procedure PC-4)

Task	Time	Worker Location
o After gas sample has purged 10 min. (see DC-2) draw off 1 cc in syringe from 300 cc glass collection bulb.	30 sec.	3
o Transfer gas to partitioner (gas chromatograph)	30 sec.	3
o Reading from G.C. gives hydrogen, nitrogen and oxygen.	3 min.	4
o The gas then dissipates from G.C. to laboratory.		

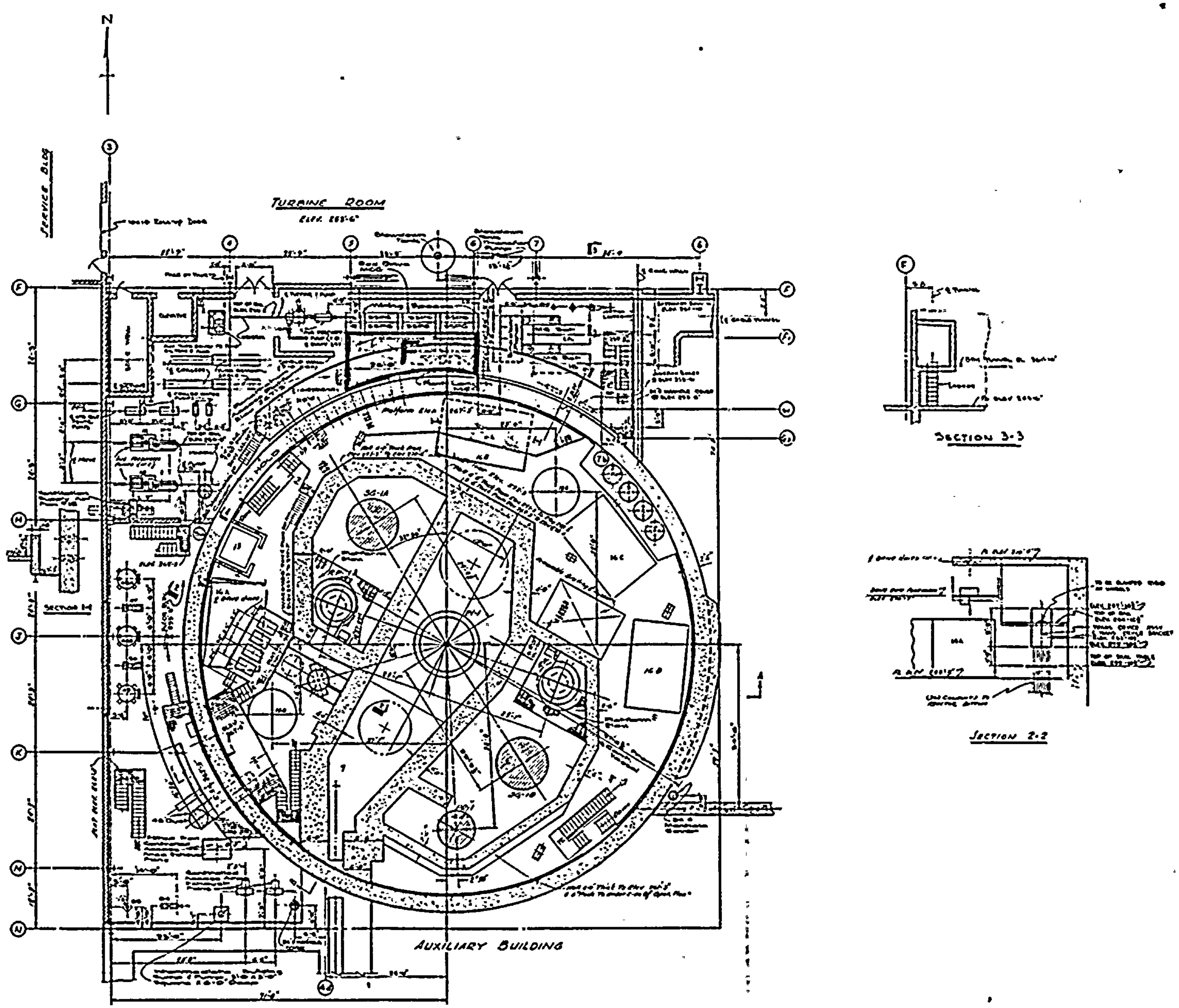
[1] Inside Nuclear Sample Room

[2] Outside Nuclear Sample Room in Low Radiation Area

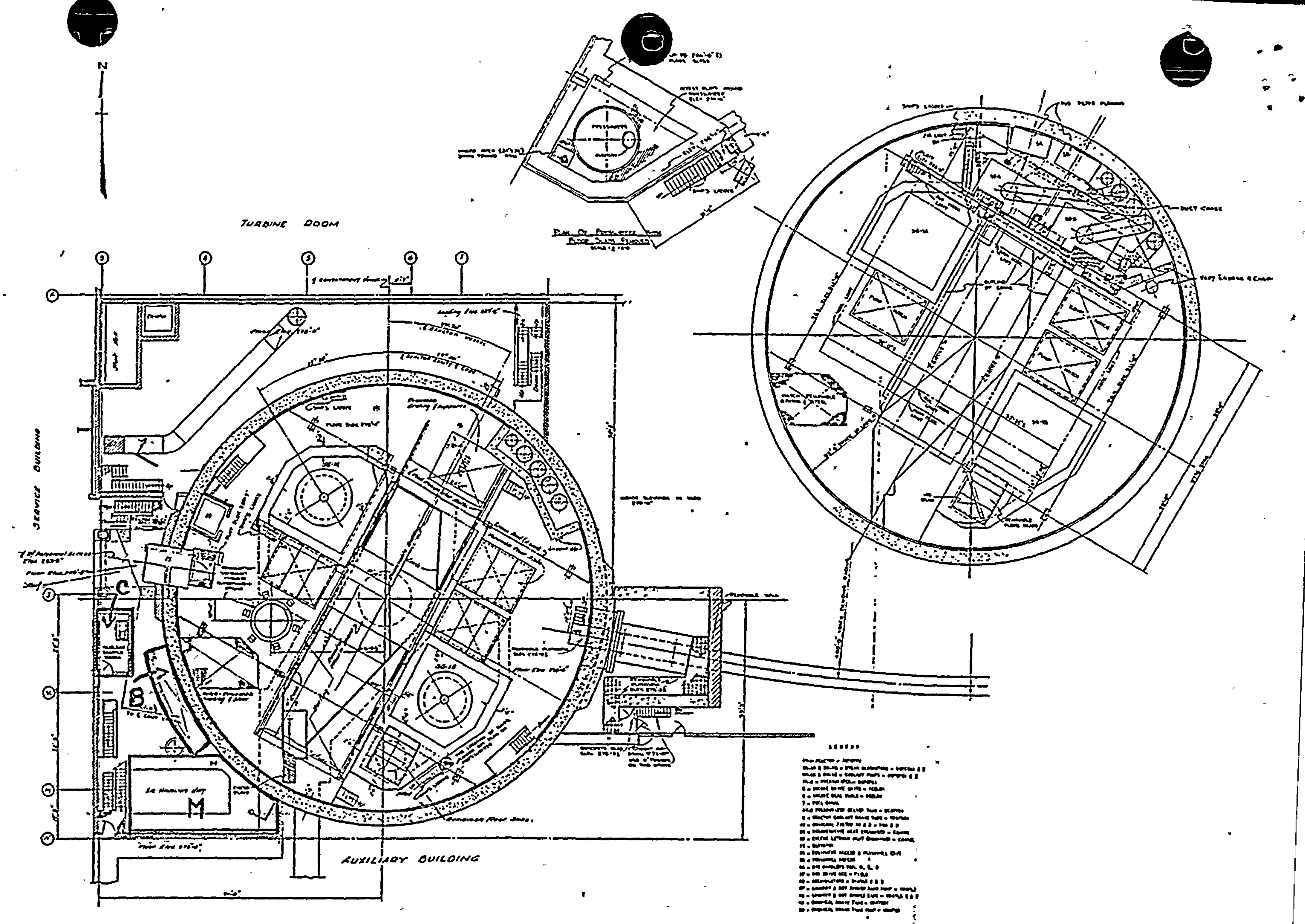
[3] Within Arm's Reach of Unshielded Sample

[4] 10 ft. Away from Unshielded Sample

[5] Sample Shielded

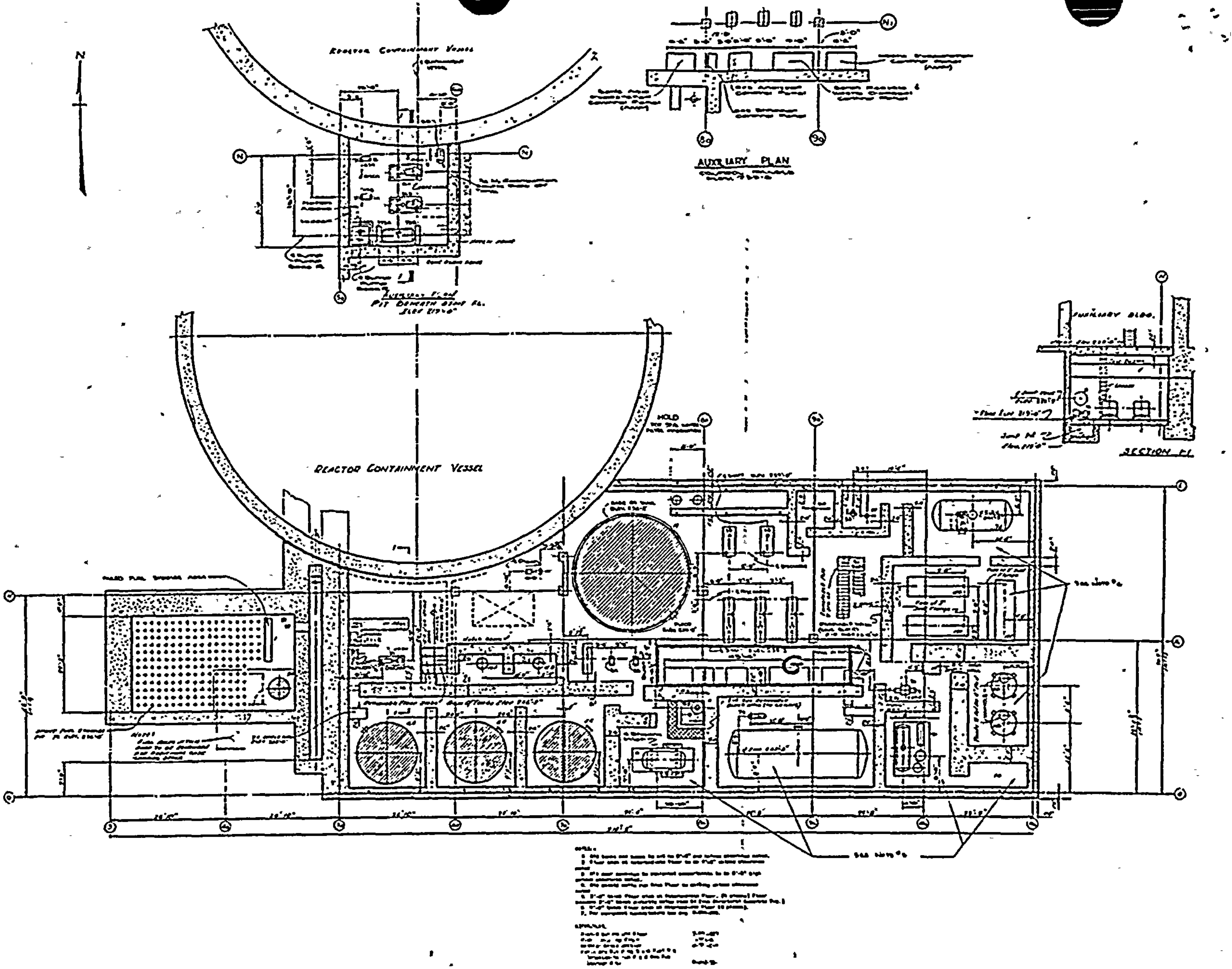


REACTOR CONTAINMENT VESSEL PLAN ABOVE INTERMEDIATE FLOOR
(D-001-022)
FIG. 4-1



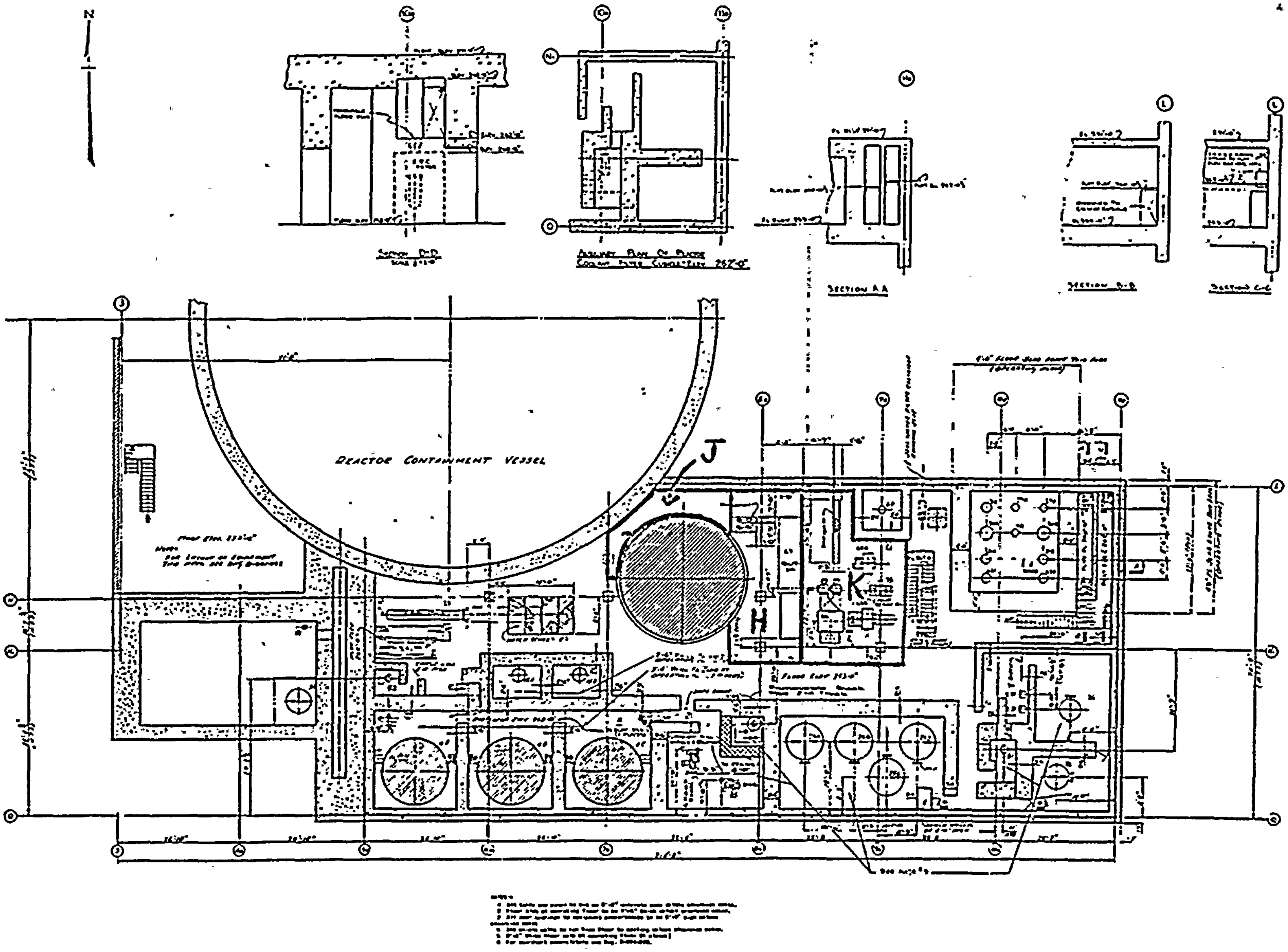
REACTOR CONTAINMENT VESSEL PLAN ABOVE OPERATING FLOOR
(D-001-032)

Fig. 4-2

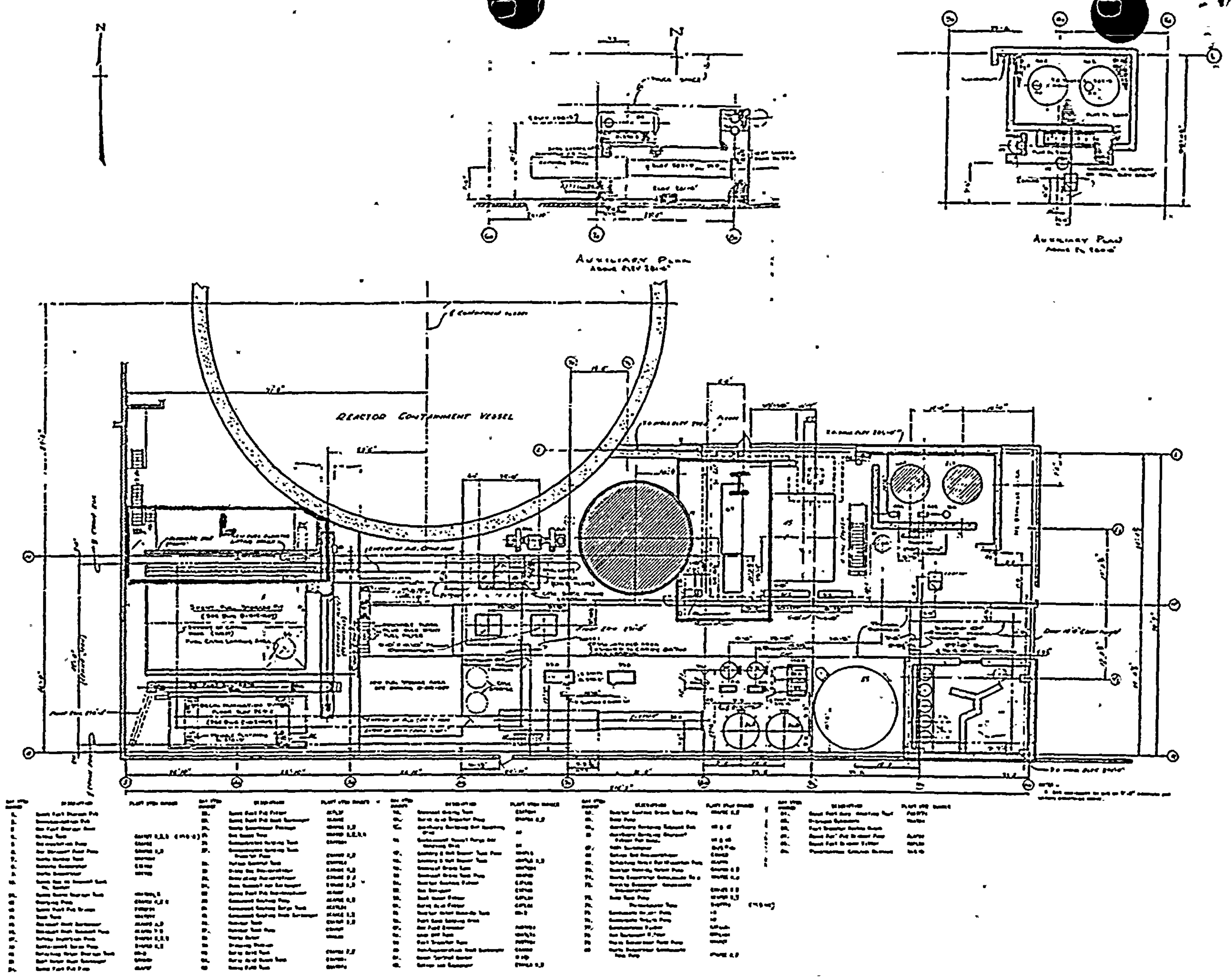


AUXILIARY BUILDING PLAN - BASEMENT FLOOR
(D-001-013)

Fig. 4-3

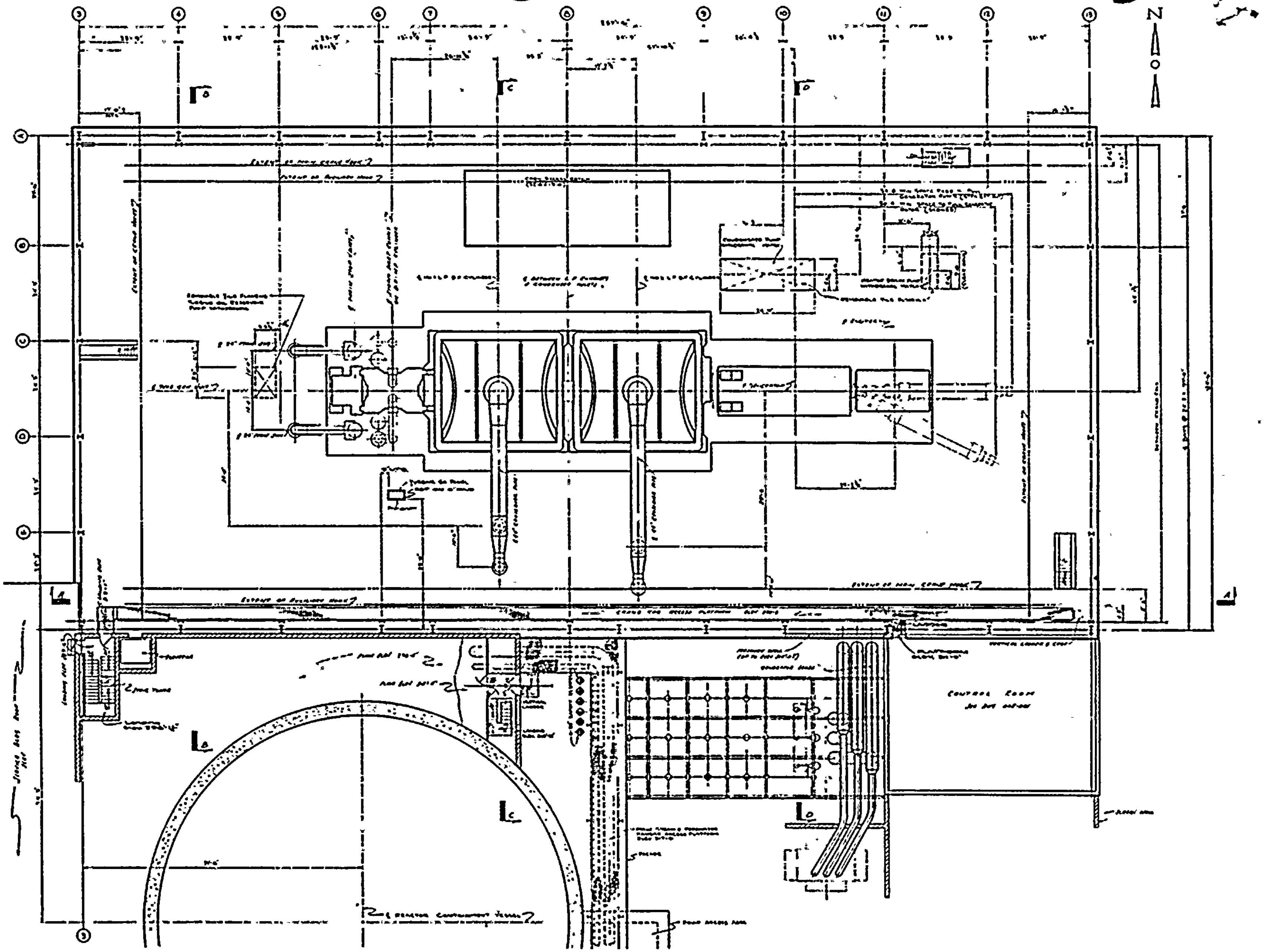


AUXILIARY BUILDING PLAN - INTERMEDIATE FLOOR
(D-001-023)
Fig. 4-4



AUXILIARY BUILDING PLAN - OPERATING FLOOR
(D-001-033)

Fig. 4-5



TURBINE BUILDING PLAN ABOVE OPERATING FLOOR ELEV. 289'-6"
(D-001-031)

FIG. 4-6