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AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Forwards responses to NRC 800929 request for addl info re industrial exposure,offsite doses & accident evaluation. Questions were discussed at public meeting between respective staffs on 800917.

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November 4, 1980
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US NRC DISTRIBUTION SERVICES BRANCH

Office of Nuclear Reactor Regulation
Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Varga:

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250 & 50-251
Steam Generator Repair

Enclosed you will find Florida Power & Light Company's responses to the requests for additional information regarding (a) Industrial Exposure, (b) Offsite Doses, and (c) Accident Evaluation, which were enclosed with your letter of September 29, 1980. These questions were also discussed at a public meeting between our respective staffs in Homestead on September 17, 1980.

Please notify us if we can supply additional information.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/LFR/ah

Enclosures

cc: J. P. O'Reilly, Director, Region II
Harold F. Reis, Esquire
Norman A. Coll, Esquire
Mark P. Oncavage
Neil Chonin
Henry Harnage
Burt L. Saunders

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Accident Evaluation

Question 1:

The answer to previous question (1) (a) concerning the ability of the air handling system to maintain negative pressure at all points in the containment indicates that the air flow at the open equipment hatch would be about 300 fpm. The wind at Turkey Point blows faster than this value about 95% of the time. Respond to the previous question, taking into account turbulent air flow conditions inside containment due to wind speeds in excess of 3 mph external to the containment.

Response:

As previously described in response to question (1)(a), the permanent 35,000 CFM containment purge system is being supplemented by an 18,000 CFM temporary exhaust fan, with HEPA filter, to bring the total exhaust capacity during the repair to 53,000 CFM. This airflow will maintain a negative pressure throughout containment even with the equipment, personnel, and emergency hatches open. Should wind conditions in excess of 3 mph arise outside containment, the effect on the containment atmosphere will be localized to only those areas immediately adjacent to the containment openings. Since there is no significant radiation work planned for these areas, the consequences of these localized perturbations in pressure will be negligible. Additionally, the work of most radiological significance (i.e., cutting of the channel head) will be done well inside containment and within the confines of temporary steel enclosure structures. As further described in response to Question 2 below, each of these enclosure structures has a separate exhaust and HEPA filter system which will maintain an additional negative pressure in these areas.

Also, another design feature is being added to enhance the ability to maintain negative pressure should problems arise. This feature consists of a movable barrier which will be installed at the equipment hatch. This barrier will be designed so that the opening in the containment hatch can be quickly reduced. Should a problem arise, such as the postulated drop of a lower assembly, the permanent purge system will be shut off (leaving only the 18,000 CFM HEPA-filtered purge system running) and the barrier opening will be reduced to 17.4 square feet or less. Calculations show that an opening of this size will enable a negative pressure to be maintained with only the smaller purge system operating.

Accident Evaluation

Question 2:

In answering previous question (1)(b), Turkey Point has not considered that there may be accidents inside or outside of the tented and filtered area which may damage the isolation barrier. Since, as long as the equipment hatch remains open, isolating the vent will not stop the release, provide to NRC a review of the planned activities to be conducted within tents that could release radioactive material (i.e., decon, cutting, welding) to assure that no potential for damage to the tent or filter exists or that appropriate procedures will exist to mitigate the effects of barrier damage.

Response:

In response to the first part of this question, there were many considerations which went into the design of the isolation barriers, including accidents. Hence, the design of the barriers has been finalized to be not a tent, but rather a galvanized steel enclosure structure which makes use of existing concrete walls where practicable. Each enclosure structure (one for each steam generator) will have two windows, a ventilation inlet with roughing filter, a HEPA filter exhaust system, and a double door access area. The structure will be joined to the concrete walls and steam generators by angle irons and sealed with an air setting sealing compound for relative air tightness.

Concerning the second part of this question, the potential for damage always exists, however, both the equipment involved in activities which may affect the barrier, and the barrier itself will be designed commensurate with the likelihood and consequences of an equipment malfunction, and in accordance with ALARA principles. The cutting of the channel head will be done by an automatic cutting machine which follows a track placed around the steam generators. The welding of the new steam generators onto the channel head is done after decontamination is performed. Therefore, the welding is a relatively clean process and does not require enclosure structures. The decontamination processes under consideration (i.e., grit blast, chemical, or electropolishing) utilize a liquid medium which, were a malfunction to occur, would result in a liquid spill, and would not create an airborne problem inside the containment, nor have any offsite consequences. The spill would be handled using normal plant health physics procedures which will be in force throughout the repair program and which enforce the ALARA principle. Hence, while we do not envision any damage to the isolation barrier due to work inside or adjacent to it, appropriate actions will be taken should any damage to the barrier or equipment occur.

Accident Evaluation

Question 3:

In answer to previous question (3) concerning lifing of the steam generator lower assembly before a plate was welded on the bottom, Turkey Point indicated that a lift of about 30 feet would be made without a cover plate welded in place. Turkey Point quoted two sentences from an article in "Proceedings of the American Power Conference, 1971" to show that the crud in the primary system is tenacious. This is certainly true, however, the article was written from the point of view of one who wishes to decontaminate an object not from the point of view of one who has to deal with any radioactivity removed, and the word "negligible" must be evaluated in that context. The previous response from Turkey Point is inadequate; provide an evaluation of the lower assembly drop accident. The staff notes that attempts to decontaminate the steam generator prior to the lift may increase or decrease the amount of crud that could be removed by mechanical shock.

Response:

The lifting of the steam generators has been carefully engineered to ensure that a steam generator drop accident will not occur. Bechtel personnel, both engineers and superintendents, all of whom are experienced in making heavy lifts, are responsible for the design, analysis, planning, rigging, review, and lifting of the steam generators. Since 1970, Bechtel has been responsible for more than 50,000 tons of special rigging and lifting in 42 power plant units, both nuclear and fossil, including the following:

Arkansas Nuclear One 1 & 2	5,000 tons
Calvert Cliffs 1 & 2	4,500
Davis-Besse 1	2,200
Fast Flux Test Facility	1,600
Limerick 1 & 2	3,500
Midland 1 & 2	6,000
Millstone 2	2,000
Rancho Seco	2,000

Additionally, most recent experience has included the successful receipt, rigging, and placement of the six replacement steam generators on site as shown in Figure 3.1-2 of the SGRR.

Furthermore, as a part of the planning process, and also while performing the steam generator lift, the rigging personnel utilize process sheets. These documents provide a step by step listing of the work to be performed. They include drawings which depict rigging arrangements and specify size, location, etc. of cables, slings and other lifting devices. The process sheets have sign offs for inspection, witness and hold points where engineers and quality control inspectors verify that all work is being done in accordance with the drawings and rigging procedures.

Question 3, page 2

Additionally, the following requirements will be met by the equipment to be used in the lifts:

Crane Hooks	- Proof loaded by the manufacturer to 133% of the rated load.
Wire Ropes	- Factor of safety of 5 times the rated load capacity.
Wire Slings	- Factor of safety of 5 times the rated load and proof loaded by the manufacturer to 2 times the rated load.
Overhead Cranes	- Test to 125% of the rated load.

If, despite these precautions and careful engineering, a steam generator drop accident were to be postulated, the effects would still be well within 10 CFR 20 guidelines as described below.

There is no intention to decontaminate the tube bundle section of the steam generator. Therefore, the crud in the tubes will not be loosened by decontamination. The crud in the tubes is tightly adherent since it has withstood the hydraulic erosion effects of water at 2200 psig and 600° F flowing at 14 feet/sec during the life of the plant. Therefore it is not likely that very much crud would be loosened in a drop of a steam generator.

The crud forms particles that are generally too heavy to become airborne very readily even if they should come loose. Therefore, it is not expected that any appreciable percentage would become airborne in the event that a steam generator might be dropped. Additionally in the early stages of the lift, a herculite cover will be placed on the bottom of the assembly to catch any crud that might be knocked loose. This cover would have to be knocked away before any crud could exit the steam generator during a drop accident.

In the Environmental Impact Appraisal, the NRC chose to assume that approximately 0.1% might become airborne. However, for purposes of conservatism, we assumed that 1% of the steam generator tube bundle activity might become airborne in the event of a drop of a tube bundle section.

By analysis of the lower assembly drop accident, it has been determined that the critical organ dose (lung) is 0.45 mrem. The whole body dose due to submersion in a semi-infinite cloud is negligible. These results are based on 3.14 E-2 Ci being released to the environment through HEPA filters. An accident X/Q of 5.5 E-5 $^{S}/_{m^3}$ taken from the NRC Safety Evaluation Report was used.

Question 3, page 3

However, even if a higher percentage of the activity is assumed to become airborne, the site boundary dose still does not become significant since the dose only goes up in direct proportion to the number of curies released. For example, even if the extremely conservative assumption is made that 100% of the activity becomes airborne, the site boundary dose is only 45 mrem (lung), well below the limit of 1,500 mrem/yr. in the Radiological Effluent Technical Specifications (RETS) for the Limiting Conditions for Operation (LCO) for normal releases, and far below the accident release limits.

In addition, it should be noted that the concept of dispersion of particulate releases to the site boundary should include the use of a cloud depletion or deposition factor. The use of an accident X/Q implies a very low wind velocity which enhances deposition. Therefore for particulate releases, it is very conservative to calculate a site boundary dose using an accident X/Q without depletion.

Accident Evaluation

Question 4:

NRC notes that no response to previous question (5) has been received. The method of decontamination is particularly important in evaluating the lower assembly drop accident as noted in present question (3).

Response:

FPL has not completed evaluation of Decontamination Equipment and has not awarded a contract for a specific system. However, the following generic statements can be made irrespective of the decon system used:

The connections for the decon system will consist of high-pressure hose, doubly contained to prevent airborne in case of leakage, and not standard rubber hose. In addition any high-pressure portion of the system will be downstream of the filter separation equipment which will remove the majority of contaminated crud and grit.

While the exact method of decontamination has not been determined, the consequences of a decontamination system failure can be evaluated by considering a worst case situation. For purposes of conservatism, it was assumed that all 45 curies in the steam generator channel head might be instantaneously spewed into the air inside the containment. No credit for HEPA filters was taken and the NRC accident X/Q of 5.5×10^{-5} sec/m^3 from the Steam Generator Repair Report, Safety Evaluation Report and Environmental Impact Appraisal was used. The resulting site boundary dose is 645 mrem (lung) which is well within the 10 CFR 20 limits for normal releases and is far below the 10 CFR 100 limits for accidents.

Question 5:

The X/Q value to be used in a radiological assessment depends on the period of time over which releases are made. For accident conditions, as well as for some "normal" releases that occupy only a short time period (e.g. cutting or welding operations), an annual average X/Q is not appropriate.

Response:

In calculations to be used for safety evaluations, accident X/Q values are appropriate because they maximize doses. However, in calculations which are used for environmental impact evaluations, realistic assumptions and data should be used. The value of X/Q used in determing lung doses at the site boundary was 1.02 E-6 $^{S}/_{m^3}$, the annual average. Using this X/Q, a dose of 0.00607 mrem was calculated for the SG drop accident and represents a realistic dose. A conservative X/Q of 5.51 E-5 $^{S}/_{m^3}$ (taken from the Safety Evaluation Report for Turkey Point Plant Units 3 and 4) results in a dose of 0.45 mrem.

For releases made over several months which are considered normal releases (e.g. from channel head cutting or decontamination)the average annual X/Q was applied. This is consistent with the NRC interpretation given in NUREG-0133 that the annual average X/Q be used where releases, even though they might be short term individually, are sufficiently random in timing that they average out over the time period. The generators will be removed sequentially so that the releases will be randomly distributed during the outage.

Question 1:

Footnote 1 to Table 5.2-1 states that the estimated activities per unit area of corrosion products on the steam generator side of the plenum are based on "actual Turkey Point data". Describe the "data" indicating: (a) When it was taken?; (b) What actual physical measurements were made?; and (c) Why it is applicable to the cuts that will be made?

Response:

The data is from Westinghouse report ST-RES-FJF-3030 dated July 7, 1977. The data was obtained by performing a radioisotopic analysis of the Turkey Point 3A steam generator manway diaphragm on October 25, 1975 after 90 hours of decay. This activity was then extrapolated to account for nine years buildup of radioisotopes. The diaphragm is stainless steel similar to the stainless steel cladding of the steam generator channel head. The diaphragm is immersed in reactor coolant as is the rest of the channel head.

Since the cuts will be made on the channel head in the vicinity of the diaphragm, it is expected that the activity on the channel head cut area will be very similar to that on the diaphragm.

Question 2:

Footnote 5 to Table 5.2-1 lists the area for 6 components. What type of area is listed? For example, is it surface area, area assumed to be vaporized in cutting? Is there any relation between this area and the area referred to in your response to Question A-35?

Response:

Question A-35 and the response to it concerned cutting the steam generators into small pieces for disposal. This method of disposal is no longer being considered. Therefore, there is no relation between the response to question A-35 and these areas.

The areas in Table 5.2-1 are totals for the various steam generator components. They were provided along with their relative crud concentrations for information purposes. The activity vaporized in cutting would consist of only that within one inch on either side of the circumferential cut on the channel head and the cut between the divider plate and the tube sheet. No cutting of the tube sheet or tube bundle is being considered.

Question 3:

Table 5.2-2 lists estimates of airborne releases to the environment. Identify the assumptions and basis for assumptions that were used to make these estimates. For example: How many different types of cuts will be made? What is the length and width of each cut? What surface decontamination factors were assumed for the various cuts? What activities per unit area were estimated to be on each cut area? How many filters and what filter efficiencies were used for particulates and radioiodines?

Response:

The assumptions made for airborne releases are given in Section 5.2.2 of the Steam Generator Repair Report. A decontamination factor of 12 for the channel head is assumed before any cut is made (Section 5.2.2.1e). Activities per unit area are obtained by applying a DF of 12 to the activities in Table 5.2-1.

The airborne releases are assumed to result from cutting and the complete decontamination of the entire channel head and divider plate surface area (Section 5.2.2.1e). During cutting of the channel head, the channel head will be surrounded by a contamination control envelope. The atmosphere from this envelope will be exhausted through a 99% efficient HEPA filter and will then flow through ducting directly to the containment purge system.

Question 4

Table 5.2-2 lists estimates of airborne releases to the environment. Give the vent release points and the height of the releases. Over what time period (e.g. a few days, a few months, a year) will most of the airborne activity be released? If most of the airborne activity is released over a short time period, then annual average meteorological dispersion factors might not be appropriate for estimating doses to the maximum individual.

Response:

Releases of radioactivity will take place over a period of a few months. The release points are the plant vent (elevation 200') and the auxiliary building roof (elevation 63'). The worst annual average X/Q has been used in calculating doses since the releases are considered normal and span a time period of several months. NUREG-0133 allows the use of the annual average X/Q even if the releases are short term as long as they are sufficiently random. The removal of each steam generator is not predicated on a particular time of day, or day of the month, but rather on the completion of necessary preliminary construction operations, all of which are also independent of time. This, coupled with normal variations of the repair schedule, ensures the randomness of the repair program with respect to time.

However, the use of an accident X/Q value of 5.51 E-5 s/m^3 from the Safety Evaluation Report and from the Environmental Impact Assessment will not increase doses even to the 10 CFR 50, Appendix I level for normal operating releases. While conservative assumptions are applicable to safety evaluations, more realistic assumptions should be used for environmental impact evaluations.

Question 5:

Why is the value for estimated release of iodines in Table 5.2-3 (i.e., 0.0032 Ci) less than the sum of the iodine releases in Table 5.2-2 (i.e., 0.0105 Ci)?

Response:

The value of estimated releases of iodine in Table 5.2-3 (i.e. 0.0032 Ci) should be changed to 0.0105 Ci. Even so, the estimated releases due to steam generator removal and replacement will still be on the order of one-fifth of the normal releases if the plant were operating.

Question 6:

Table 5.2-5 contains estimates of specific activities in laundry waste water. State the basis for these estimates. For example, what physical measurements were made to arrive at these estimates? Why are these estimates applicable to the steam generator repair at Turkey Point?

Response:

Table 5.2-5 contains estimates of specific activities in laundry waste water. This data was obtained from actual Turkey Point laundry water during the course of a year including normal operation, refueling, steam generator tube eddy current testing, and tube plugging. These estimates are, therefore, considered applicable to the steam generator repair operation.

Question 7:

Identify (i.e., give distance and direction from the site) "the worst site boundary location" referred to on p. 5-22. Provide a table that lists the nearest full time residence, garden, cow and meat producing animals in each sector.

Response:

The attached Table provides the requested information out to 5 miles. The worst site boundary location is shown in the Turkey Point FSAR, Figure ID-1 (location at 360 degrees north). The distance to this location is 4164 feet (see FSAR, page 2.13-1). An aerial view is shown on FSAR Figure 2.2-2.

LAND USE CENSUS
TURKEY POINT PLANT
JUNE 3, 1980

I.

METEORLOGICAL SECTOR	RESIDENCES	GARDENS	MILK ANIMALS
North	1.9 Miles (Note: 1)	None	None
NNW	None	3.8 Miles	None
NW	3.5 Miles	3.1 Miles	None
WNW	None	3.6 Miles	None
West	None	None	None
WSW	None	None	None
SW	None	None	None
SSW	None	None	None
South	None	None	None
SSE	None	None	None
SE	None	None	None
ESE	None	None	None
East	None	None	None
ENE	None	None	None
NE	None	None	None
NNE	None	None	None

Note 1: Residence refers to trailers located in the Biscayne National Monument section of Bayfront National Park. Based on interviews with park rangers; trailers were not occupied at the time of the survey; but persons have lived there in the past and he expects the trailers to to be reoccupied before the end of the year.

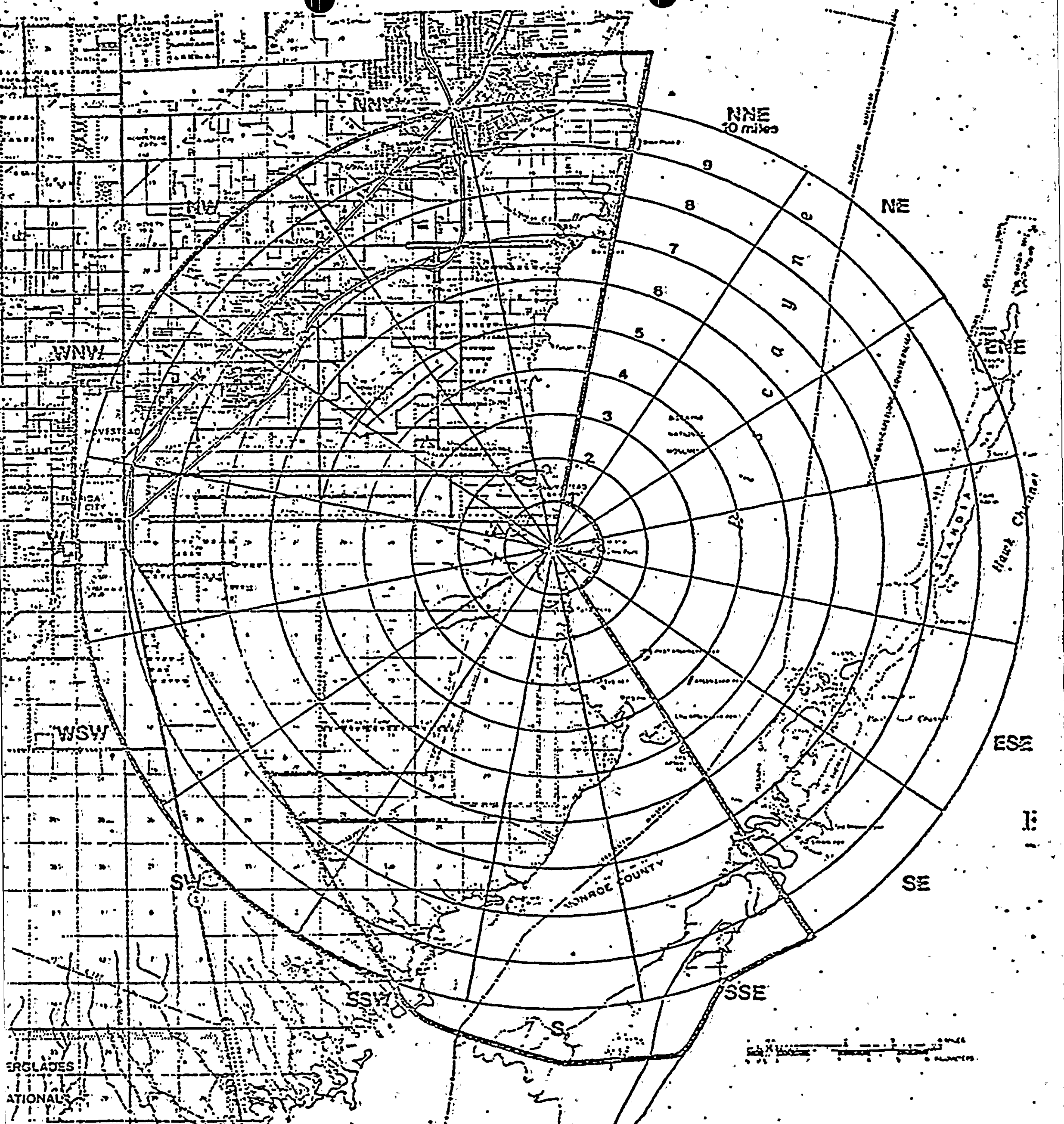


FIGURE 3-1

EVACUATION STUDY AREA (10 mile EPZ)

o Residence

Agriculture (Gardens)

△ Other Occupied Areas

Offsite Dose

Question 8:

The response to Question A-49 is incomplete. What is FPL's estimate of the total volume and curies of solid wastes which are to be disposed of including transuranics, I-129, Fe-59 and Ni-63?

Response

The total volume of solid wastes which are to be disposed of including the steam generator lower assemblies is approximately 39,200 cubic feet per unit with a total activity of about 880 curies per unit as shown in Table 8-1 below.

Excluding the steam generator lower assemblies, the totals are approximately 28,690 cubic feet and 270 curies.

The total activity will contain only minute traces of Iron-55, Nickel-63, Iodine-129, and transuranics, as follows:

A. Iron-55 and Nickel-63 are activation products in the reactor vessel caused by neutron irradiation from the core. The steam generators however are not subjected to significant neutron irradiation, therefore there will be insignificant activation of the steel to produce Iron-55 and Nickel-63.

B. Iodine-129 has a very low fission yield so that the total core inventory is approximately 0.2 curies. Almost all of this remains in the core enclosed by the cladding on the fuel. Only an insignificant amount might be transported to the steam generators.

C. Transuranics are also held in the fuel by the cladding on the fuel. Very little of the transuranics are transported to the steam generators. From Table 5.2-1 it can be seen that there will be less than 3×10^{-4} nCi/gm long lived transuranics which is far below the limit of 10 nCi/gm for recoverable waste.

TABLE 8-1

SOLID WASTE (PER UNIT)

	Quantity (ft^3)	Activity (Ci)
Steam Generator Lower Assemblies	10,500	750
Channel Head Decon	1,260	(135, included in 750 above)
Concrete	1,620	< 1
Reactor Coolant Cutting and Weld Preparation	7	~ 1
Rags, Paper and Clothing	22,000	~50
Sand, Miscellaneous Concrete, Tools, and Scaffolding	3,000	~50
Evaporator Bottoms	700	~ 5
Spent resin	100	~25
	39,187	882

Offsite Dose

Question 9:

On the bottom of p. 3-22 it is stated that the "dose equivalent to an individual at the north site boundary for a full year would be approximately 5.2×10^{-3} mrem. State the basis (i.e., equations and values used in equations) for this estimate of direct radiation dose from the onsite storage of the steam generator lower assembly.

Response:

The annual dose to an individual at the north site boundary was determined using a point-kernel method to calculate the direct dose rate from a cylindrical source through slab shields.

This method utilized the semi-empirical methods developed by Rockwell (1) for calculating the direct gamma dose rates from a homogeneous volumetric cylindrical source through slab shields. Individual buildup factors for the source materials and the shield wall concrete were taken from the work of Capo (2). Broder's method (3) was used to accommodate multi-layer shield buildup.

The values of the source terms for the analyses were based on the results of an experimental field survey of the 4-A steam generator in a drained condition one month after shutdown.

No credit is taken for the 2.63" thick steel shell surrounding the tubes.

The dose rate at t=0 is integrated over the total number of hours per year to arrive at the annual dose. No credit for decay has been taken after placement of the steam generators in the compound. It was assumed that all six steam generators were placed in the compound at the same time. Actual credit for decay would result in a dose reduction by another factor of one third.

The basic point kernel equation is:

$$\dot{D} = S_v \cdot \frac{1}{4\pi r^2} \cdot B \cdot Ke \cdot \sum_i e^{-\mu i X i}$$

where $\dot{D}$ = dose rate
S_v = volumetric source strength
r = distance from source to dose point
B = buildup factor calculated with Capo's values utilizing Broder's method to accommodate multi-layer shield buildup @ E=1.3 MeV
Ke = tissue response function (flux to dose conversion factor) @ E=1.3 MeV
μ = Linear attenuation coefficient for the ith shield @ E=1.3 MeV
x = thickness of the ith shield
i = material index, i.e. concrete and air

It was assumed that the sole contributor to the measured dose rates was Cobalt-60. Based on a 1.3 MeV energy gamma ray, the dose rate can be approximated using the values given below:

$$S_V = 1.806\ E4\ \frac{MeV}{cm^3\ sec}$$

$$r = 4164\ ft$$

$$Ke = 1.653\ E{-3}\ \frac{mRem/hr}{MeV/cm^2\ sec}$$

$$\mu = 0.131\ cm^{-1}\ (concrete)$$

$$= 7.29\ E{-5}\ cm^{-1}\ (air)$$

$$X = 24''\ (concrete)$$

$$= 4164'\ (air)$$

Attenuation by the steel tubes in the steam generator should also be taken into account.

References

1. T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., New York (1956).

2. M. A. Capo, Polynomial Approximation of Gamma Ray Buildup Factors For a Point Isotropic Source, APEX 510.

3. R. G. Jaeger, et. al., Engineering Compendium on Radiation Shielding, Shielding Fundamentals and Methods, I (1968).

INDUSTRIAL EXPOSURE

QUESTION

1. Your August 8, 1980 response to some of the questions in item 1 of our July 29, 1980 letter on how you will incorporate the following provisions of Regulatory Guide 8.8, Revision 3 (June 1978) in the steam generator replacement project is incomplete. The following positions and request for additional information are need to complete our review:

   C. Regulatory Position

   2. Facility and Equipment Design Feature

      b. Radiation Shields and Geometry

         (2),(5) Verify that the temporary shielding discussed in Sections 3.3.5.1.a and 3.3.5.2 of your report includes shield plugs for openings to the primary system, such as primary coolant pipes and steam generator openings after cutting the steam generator.

RESPONSE

1.C.2.b. The temporary shielding includes shield plugs for primary coolant pipe openings to be used as needed to maintain exposures "as low as reasonably achievable" (ALARA).

QUESTION

1.C.2. d. Control of Airborne Contaminants and Gaseous Radiation Sources

(1) Verify that ventilation systems will be operated so that airflow for all buildings associated with the steam generator replacement will be from less contaminated areas to more contaminated areas.

Provide details of the local contamination containment areas you plan on erecting to control airborne activity.

RESPONSE

Airflow for all buildings assoicated with the steam generator replacement will be from less contaminated areas to more contaminated areas.

For details on local contamination containment areas, see the response to Item 2 of Enclosure 3.

INDUSTRIAL EXPOSURE

QUESTION

2. Verify that your training program contains instructions in accordance with Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

RESPONSE

The FPL Training program contains instructions in accordance with Regulatory Guide 8.13.

QUESTION

3. It appears that your DF of 12 is based on engineering judgment, and not based on previous decontamination of a steam generator. In the event that your DF will not provide this decontamination, discuss your alternative procedures for cutting the channel head and divider plates. In addition, provide the details of the dose analysis that lead to the results listed in Table 331.13-1 for the decontamination, cutting and welding of the steam generator (including setup and cleanup after the job is completed). Include in your response your analysis of occupancy, task performance time, radiation fields and occupational dose.

RESPONSE

The alternative procedure will be to provide shielding in the form of lead sheet, blankets, and brick, and to conduct all operations as remotely as feasible to maintain exposures ALARA.

Details of FPL's dose analysis are provided in attached Table 3-1

TABLE 3-1

(Reference: FPL Response to NRC Question 331.13, Table 331.13-1)

TASK #14:

	Pipe Cut 2 Cuts/Pipe Man-Hours	Pipe Cut 2 Cuts/Pipe Man-Rem	Channel Head Cut Man-Hours	Channel Head Cut Man-Rem
a) Remove Pipe Sections	1,730	172	-	-
b) Prep for Weld	2,850	318	1,700	50
c) Weld	5,155	478	6,230	133
d) Clean-up	1,105	46	-	-
e) Erection & Removal of Facilities	27,010	141	27,010	141
Totals	37,850	1,155	34,940	324

TASK #15: COMMON TO BOTH METHODS

	Man-Hours	Man-Rem
a) Temporary Power	6,705	170
b) Clean-up	42,309	201
c) Construction & Removal of Facilities	12,416	45
Totals	61,430	416

TASK #16: COMMON TO BOTH METHODS

	Man-Hours	Man-Rem
a) FP&L Personnel		
1) HP	16,125	127
2) QC	16,125	127
3) Supporting	6,052	49
b) Bechtel Personnel		
1) Engineers	16,116	71
2) QA	10,070	44
3) Supervisors	4,052	18
Totals	68,540	436

QUESTION

4. Your response to question 10 of our July 29, 1980 letter is incomplete. Provide a drawing showing location of new facilities including change room, access control station, laboratory facilities and decontamination facilities.

RESPONSE

A conceptual outline (including drawings) is attached. It is for planning purposes only and is subject to change.

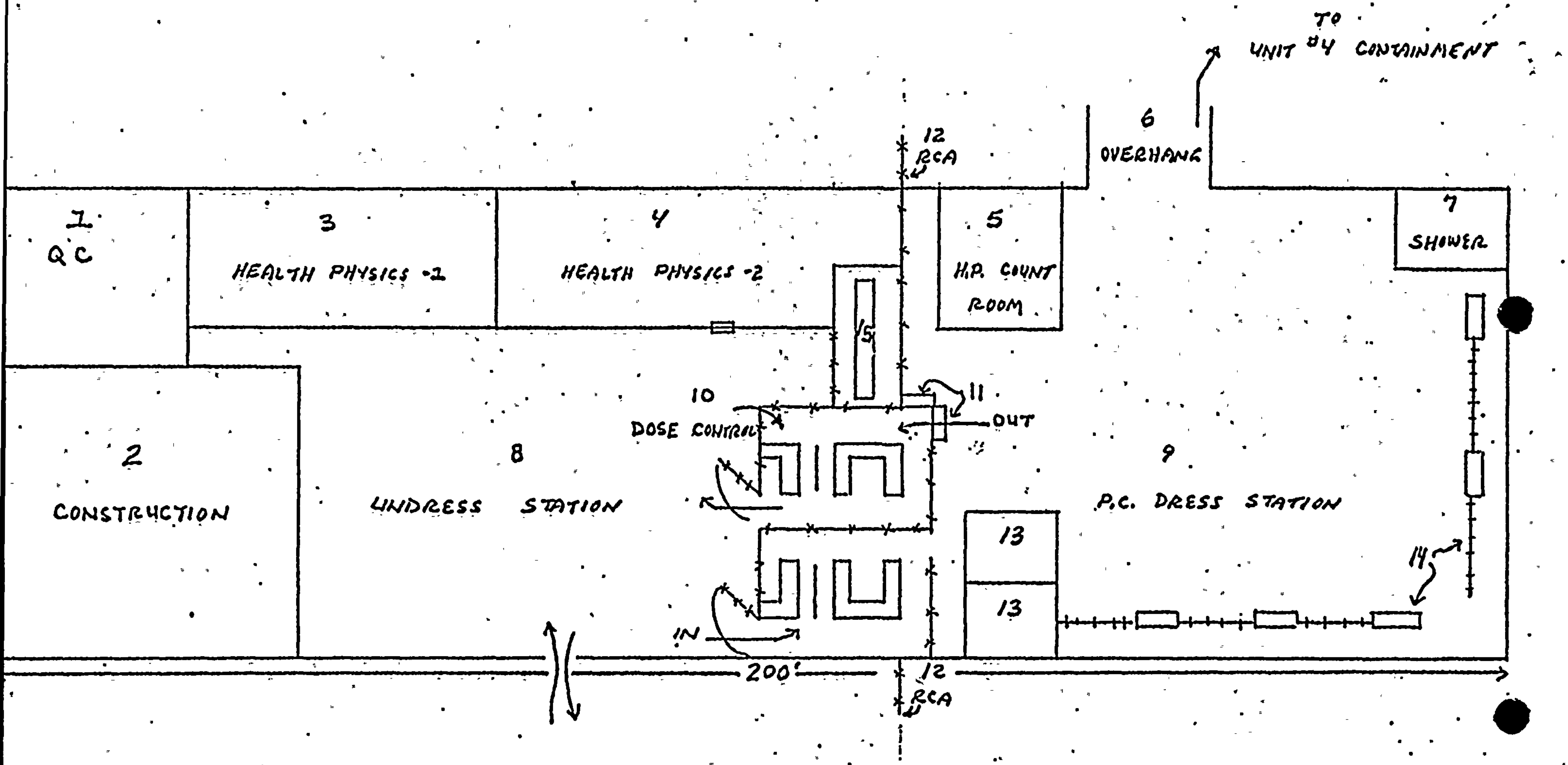


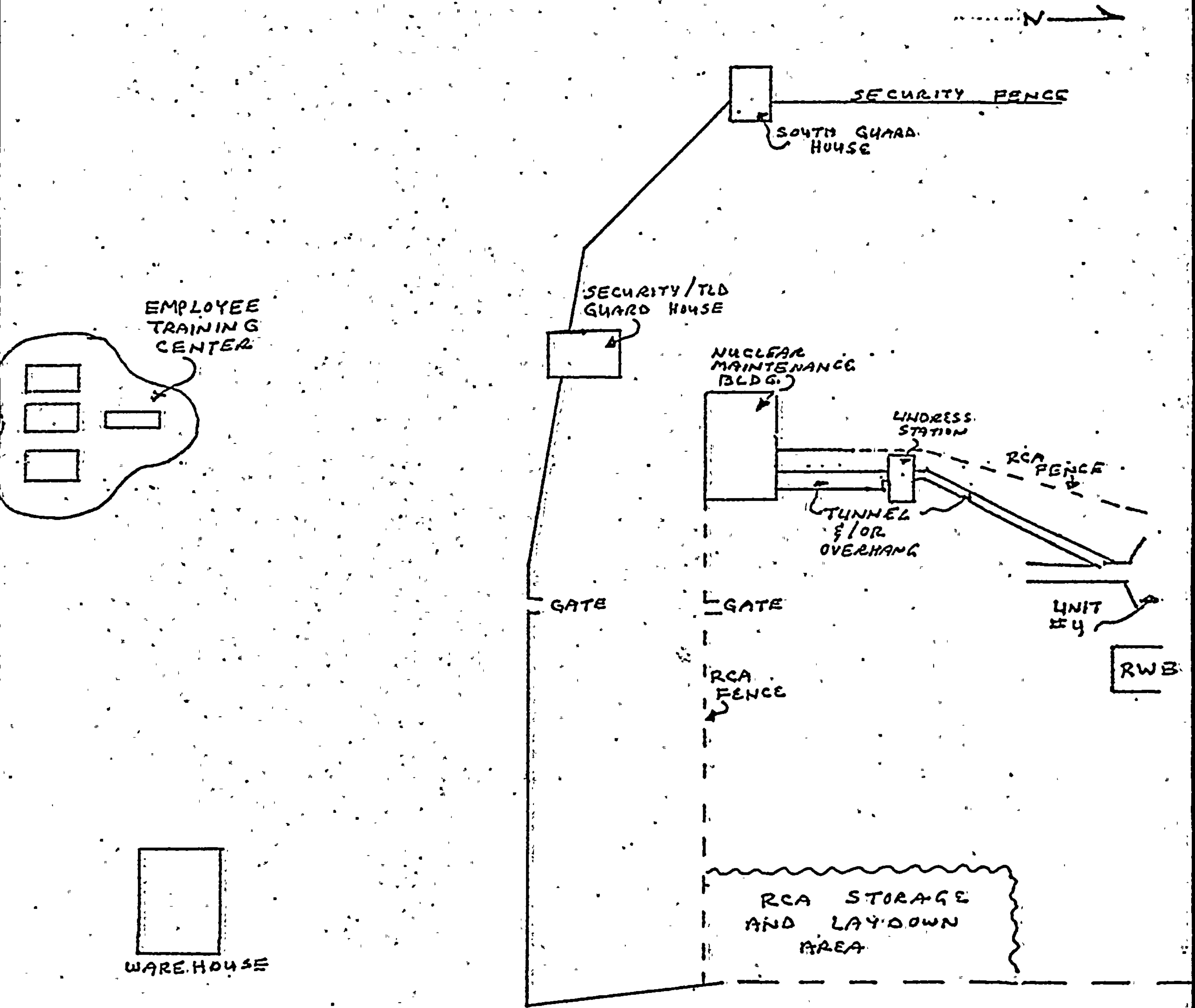
FIGURE 4

NUCLEAR MAINTENANCE BUILDING

NOTE:
NOT TO SCALE

FIGURE 1

GENERAL LAYOUT OF BUILDINGS ASSOCIATED WITH HEALTH PHYSICS ACTIVITIES



NOTE:

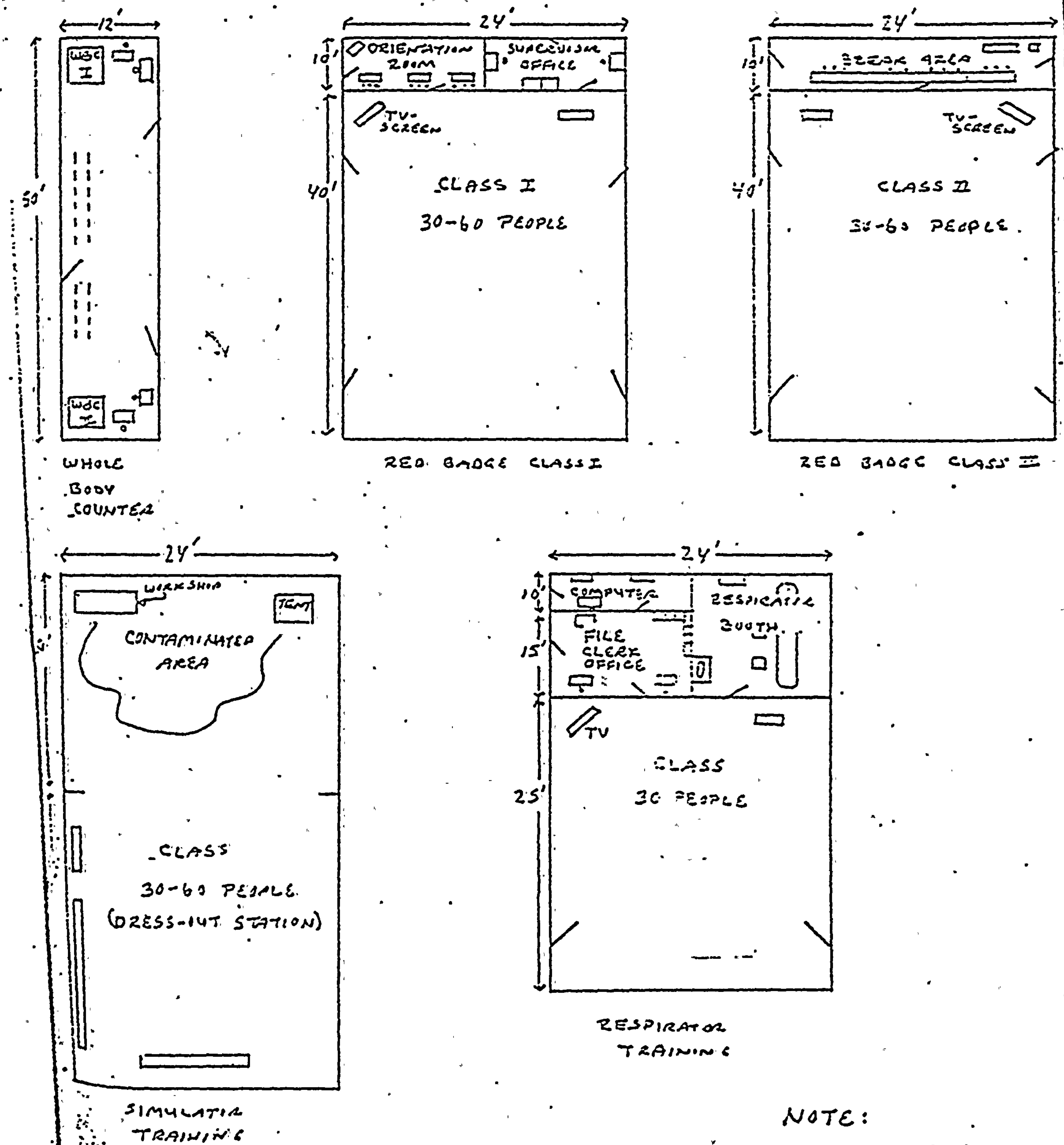
ATTACHMENT TO QUESTION 4, INDUSTRIAL EXPOSURE

Figure 1 reveals the general layout for the Unit #4 Steam Generator Replacement Outage with the locations of major buildings associated with health physics activities presented. These buildings consist primarily of:

1. Employee Training Center (Figures 1 & 2)
2. Security/TLD Guard House (Figure 3)
3. Nuclear Maintenance Building (Figure 4)
4. Tunnel and/or Overhang

FIGURE 2

EMPLOYEE TRAINING CENTER



NOTE:
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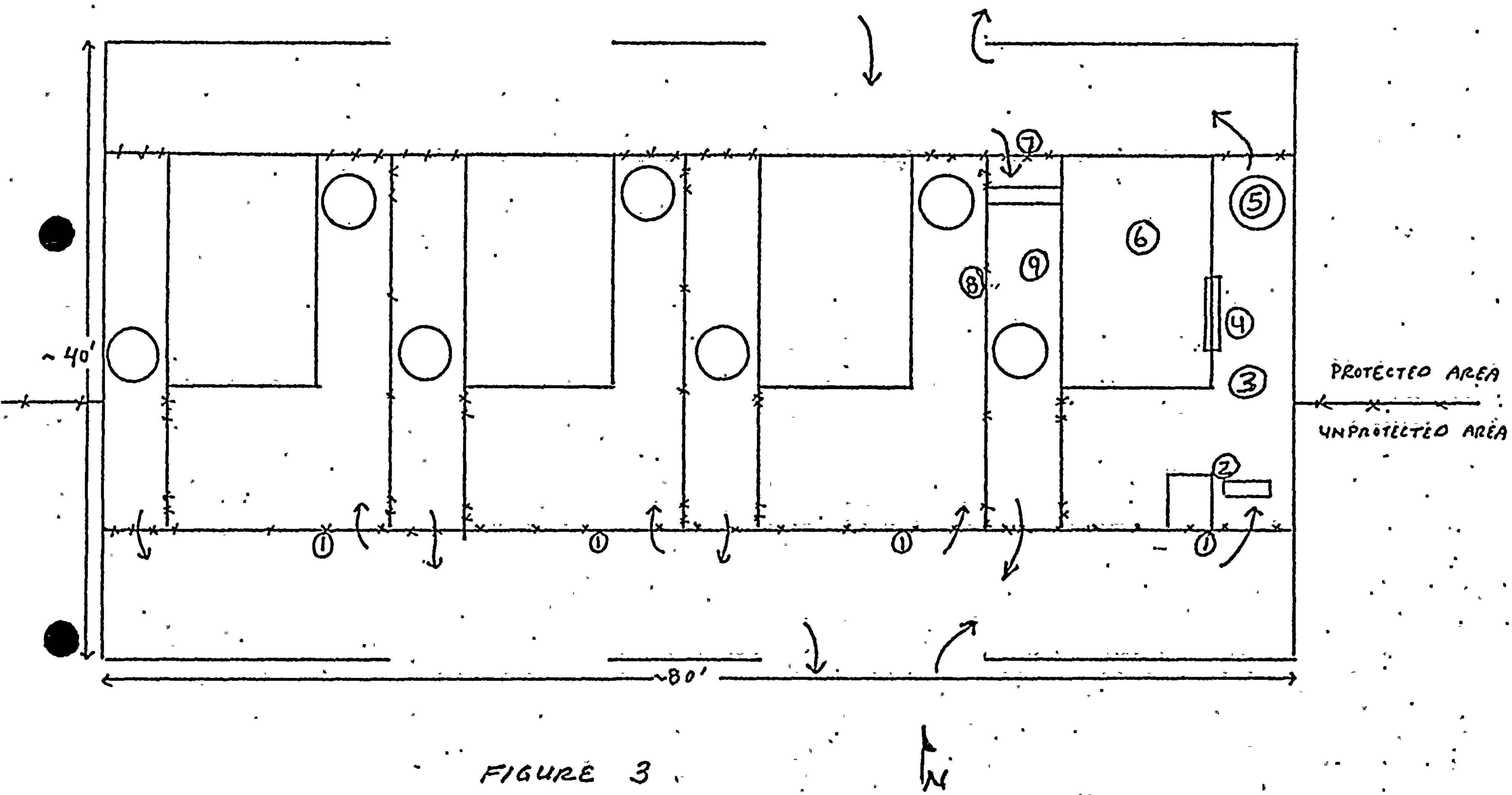

FIGURE 3

SECURITY / TLD GUARD HOUSE

FIGURE 3

(1) Sign posted at entrance door informing people as to which lane to enter for security card key and TLD badge issuance. Both card key and TLD will have the same number. If individual forgets his TLD number, it may be quickly determined by the daily health physics exposure print-out.

(2) Metal Detector/Lunch Box Search Point- A guard will insure that all personnel go through the metal detector and search articles brought into the Plant prior to issuing badges. People will line up in hallway to pass through this point rather than outside (hopefully) due to weather.

(3) Long hallway, single lane, for people to line up for badge issuance.

(4) Individual asks guard for badge (TLD #). Guard obtains badge and verifies (by picture) properly issued badge. He then punches individual through guardhouse (gate terminal, item 5).

(5) Gate terminal.

(6) Security/TLD badges will be racked up numerically in booth for quick issuance. Only those badges posted will be in this booth.

(7) Upon completing daily work assignment, individual enters guardhouse at the same booth as he entered at beginning of day.

(8) Barricade between "In" and "Out" lanes prevent individuals from leaving the guardhouse without returning their TLD/ security badge.

(9) Individuals (single lane) drop badge in designated container, guard punches out individual (by hand rather than card) after verifying that card has been returned with TLD in place. Individual then passes through gated terminal. Portal monitor will also be located at this point to insure that any potentially contaminated individual does not leave the property site. Friskers will also be available when portal monitors are not functioning properly.

General Comments:

1. Normal Distribution Through Booths - rather than issuing TLD badges in a numerical order, starting with 1 and on up, numerical brackets are used in accordance with the booth numbering system:

0 - 750
751 - 1500
1501 - 2250
2251 - 3000

As individuals are initially issued their TLD badge, the scheme used is:

Employee #1 - 1
Employee #2 - 751
Employee #3 - 1501
Employee #4 - 2251
Employee #5 - 2
Employee #6 - 752

etc.

If the flow path between booths becomes out of distribution (which can be monitored daily by computer log ins), TLD badges from those booths that are proportionately low can be brought up to par as new people get badged.

2. Lost Card Key - permanently lost card keys can never have that number reissued, therefore the same TLD number cannot be reissued. This does not present a major problem since only 41 card keys have been lost during the past two years (over 4000 different cards issued).

3. Lost TLD - lost TLD's require an immediate investigation by Health Physics. If warranted, a new TLD will be issued with the same number as the lost one. Investigations will be handled at dose control (in RCA) or visitor station (2° Side).

4. Faulty or Broken Card Key - it requires anywhere from 30 to 90 days to replace faulty or broken card keys. Since the card key number is out of service until replaced, the corresponding TLD number cannot be reissued until then.

5. Monthly TLD/Card Key Changeout - since TLD's are routinely changed out on a monthly basis, the card keys will also only be changed out monthly. Thus, people terminated during the month will have their TLD and card key held until the end of the month before reissuing that number.

6. 2000 TLD's In Circulation - from Surry's estimates, they had up to 2000 TLD's in circulation during a one month period.

NUCLEAR MAINTENANCE BUILDING

Figure 4, along with its attachment, reveal the proposed spacial layout in the Nuclear Maintenance Building (NMB). This layout takes into account existing plumbing and wall port requirements.

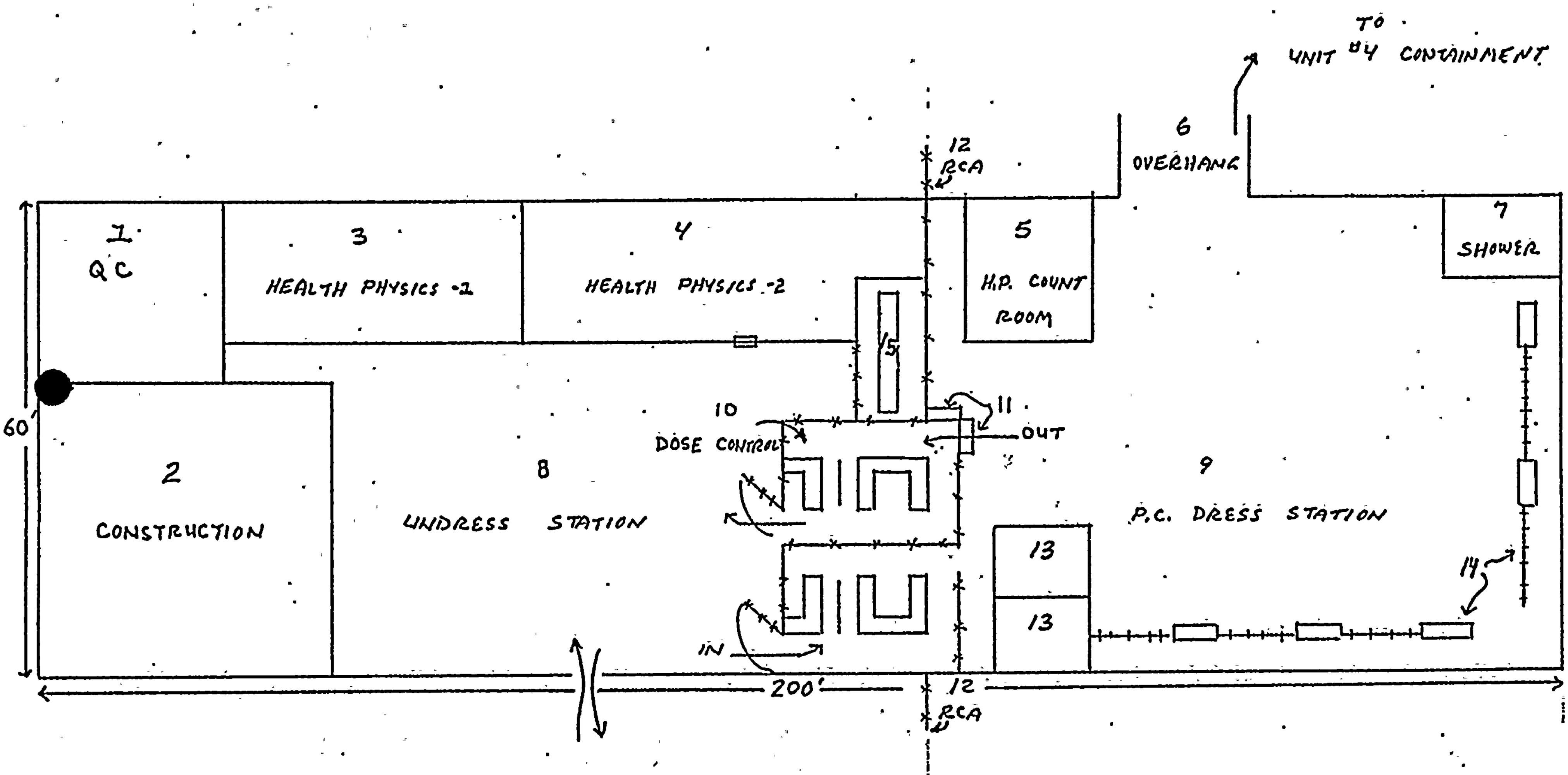

FIGURE 4

NUCLEAR MAINTENANCE BUILDING

NOTE:
NOT TO SCALE

FIGURE 4 - ATTACHMENT

A brief explanation of each room or area in the 60' X 200' NMB is presented along with their approximate spacial requirements.

1. Quality Control (Q.C.) Office - 25'X25'- meeting room and office space for Q.C. supervisors and foremen.

2. Construction Office - 36'X39' - meeting room and office space for construction supervisors and foremen.

3. Health Physics Office #1 - 18'X40' - break area and office space for health physics technicians (permanent and contract).

4. Health Physics Office #2 - 18'X45' - comprises three different rooms; 14'X20' office space for FPL operations (H.P.) supervisors and foremen; 16'X18' H.P. control room for filing of radiation area surveys, airborne measurements, RWP's and other H.P. forms; and 11'X18' storage room of H.P. supplies.

5. Health Physics Counting Room - 16'X18' - office space for analyzing contamination surveys, both smearable and airborne, storage of supplies, work tables and benches.

6. Overhang - from NMB to Unit #4 Equipment Hatch.

7. Shower - 10'X15' - personnel decontamination shower with radioactive drain to existing Radioactive Waste Building.

8. Undress Station - 36'X60' - street clothes undress station designed to handle up to 150 persons at one time with more than 400 lockers.

9. Protective Clothing (P.C.) Dress Station - 50'X50'- designed to handle up to 180 persons at one time.

10. Dose Control - 10'X13' - three access and egress lanes to control personnel radiation exposure.

11. Portal Monitors - 2 - whole body radiation monitoring prior to egressing the Radiation Controlled Area.

12. RCA Fence - fence line delineating the Radiation Controlled Area (RCA) from the protected area.

13. Bathrooms (2) - 9'X12' each

14. Protective Clothing Racks and Bins - approximately 60' of P.C. coverall racks and 30' of P.C. rubber boots, gloves, etc. bins for dress out; 4' lanes behind racks and bins for storage and handling of clean P.C.'s.

15. Electrical Panel Box - for future use.

The flow path followed in Figure 4 consists of:

1. Enter the NMB and proceed to Health Physics (H.P.) office area to review updated exposure print-out, RWP boards, and daily interaction with H.P. prior to work (window).

2. Prebriefing meeting (when warranted) with craft foreman and/or supervisor and health physics is accomplished in Q.C., construction, or H.P. room prior to undressing from street clothes.

3. Undress and place personal clothing and valuables in lockers.* Bench space is available for 150 people to undress simultaneously. Locker space is available for more than 400 people.

4. After undressing proceed to Dose Control (in) and pick up zeroed-out self reading pocket dosimeter (SRPD). Verify personal dose and margin remaining on limit, and RWP number prior to entering the RCA. If individual is restricted due to dose or invalid RWP number, entrance into RCA is prohibited.

5. Individual gets dressed in PC's and proceeds to overhang entrance for work in Containment #4 or RCA. Bench space is available for approximately 180 people to dress out simultaneously.

6. After completion of work (breaks, etc.) individual exits tunnel and/or overhang, undresses, frisks, and proceeds to Dose Control at exit of RCA. Individual turns in his SRPD, states the RWP number he is authorized to work under, and his TLD number after being monitored in the portal monitor.

7. Individual then gets dressed at end of shift or break.

8. The hot shower is situated near the exit of the tunnel.

*. Protection of personal valuables and an undressing station for female workers need to be addressed by Power Plant Construction.

INDUSTRIAL EXPOSURE

QUESTION

5. In previous correspondence you have stated that the elements of your ALARA program are incorporated in your Health Physics Manual. Our review of your manual shows two elements of Regulation Guide 8.8 that are not incorporated in your HP manual or SGRR Report. They are:

a. You did not specify who was responsible for ALARA coordination. It is our position that you have a qualified radiological engineer assigned the responsibility and authority to function as ALARA coordinator (see Regulatory Guide 8.8 Section C.3.a.(1)). It is anticipated that this function will be the individual's primary function. You should expand your description of the radiation protection organization to include such a position.

RESPONSE

a. An ALARA program is being developed by the Corporate Staff. The Regulatory Protection Program will be revised and the formal ALARA program implemented in early 1981.

An ALARA engineer is budgeted for 1981 and will be integrated into the radiation protection organization.

INDUSTRIAL EXPOSURE

QUESTION

5. b. Post-operational debriefings and feedback of experience (including doses) into new operations (See Regulatory Guide 8.8, Section 3.C.3.C).

It is our position that these sections of Regulatory Guide 8.8 be incorporated into your ALARA program. You should revise your radiation protection program to incorporate these elements of Regulatory Guide 8.8.

RESPONSE

b. Post-operational debriefings will be used for jobs incurring major radiation exposure. Feedback of dose experience is presently used via computer programs and regular printouts of doses.

SUPPLEMENTARY INFORMATION FOR FPL'S RESPONSE TO DEMINERALIZER QUESTION NO. 5, FPL LETTER L-80-231, JULY 22, 1980

WATER QUALITY VALUES FOR THE SUPERNATANT DISCHARGED FROM THE RECEIVING VESSEL TO THE DISCHARGE CANAL

DISSOLVED OXYGEN	0.08 ppm
SUSPENDED SOLIDS	0.5 ppm

TABLE 2-3

Summary of Sensible Heat Sources
D. C. Cook Units 1 & 2

Primary Water Sources (initially at rated power temperature and inventory)

- RCS fluid
- Pressurizer Fluid (liquid and vapor)

Primary Metal Sources (initially at rated power temperature)

- Reactor coolant piping, pumps and reactor vessel
- Pressurizer
- Steam generator tube metal and tube sheet
- Steam generator metal below tube sheet
- Reactor vessel internals

Secondary Water Sources (initially at rated power temperature and inventory)

- Steam generator fluid (liquid and vapor)
- Main feedwater purge fluid between steam generator and AFWS piping.

Secondary Metal Sources (initially at rated power temperature)

- All steam generator metal above tube sheet, excluding tubes.

Response to Question 3:

The AFS's in both Units 1 and 2 of the Cook Plant are identical and are designed to supply the necessary water to the steam generators under the transient and accident conditions described in our response to questions 1 and 2 above, assuming the most limiting single failure in the system. The calculated flow rate to the steam generators is given in Table 3-1. For each case considered in the flow design basis (Questions 1 and 2) and with the assumed single failure, the AFS supplies the necessary flow.

The flow numbers calculated for the given cases are listed in Table 3-1. In the blackout case the numbers account for the leak-off flow loss since the emergency leak-off valves may fail open under these conditions. As can be seen from the table, necessary flow requirements are met.

The minimum margin occurs under the feedline rupture condition with a turbine driven pump failure. Under this condition the margin between the minimum safeguards required flow and the actual system design flow is 125 GPM total for both motor driven pumps or 62.5 GPM each pump.

TABLE 3-1

Auxiliary Feedwater Flow To Steam Generators
D. C. Cook Units 1 & 2

Accident/Transient (See Note 1)	Single Failure			Number of Intact Loops	Required Flow
	Electric Train Failure	Turbine Driven Pump Failure	Motor Driven Pump Failure		
1. Loss of Main Feedwater	1380 gpm	980 gpm	1380 gpm	4	≥ 410 gpm
2. Station Blackout	1260 gpm	860 gpm	1260 gpm	4	≥ 410 gpm
3. Cooldown	1380 gpm	980 gpm	1380 gpm	4	Variable
4. Rupture of Main Feedline	990 gpm	575 gpm	990 gpm	3	≥ 450 gpm
5. Rupture of Main Steamline (Total System Flow)	1335 gpm	920 gpm	1335 gpm	3	--
a. Minimum flow to 3 intact loops	990 gpm	575 gpm	990 gpm	-	≥ 450 gpm
b. Maximum flow to 1 faulted loop	345 gpm	345 gpm	345 gpm	-	≤ 580 gpm

NOTE 1: Items 1 through 5a are minimum expected flows to intact loops; Item 5b is maximum possible flow to the faulted loop.