NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-335

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO:	FROM:	
Mr. Don K. Davis	Florida Power & Light Co. Miami, Fla. R. E. Uhrig	DATE OF DOCUMENT 8/3/77 DATE RECEIVED 8/8/77

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DESCRIPTION

(1-P)

PLANT NAME:
St. Lucie Unit No. 1
RJL 8/8/77

ENCLOSURE

Consists of requested additional information concerning the installation of additional neutron shielding in the reactor vessel cavity at Unit No. 1.....

(15-P)

ACKNOWLEDGED
DO NOT REMOVE

FOR ACTION/INFORMATION

SAFETY			ENVIRONMENTAL
~~ASSIGNED AD~~			ASSIGNED AD: V. MOORE (LTR)
BRANCH CHIEF: (7)	DAVIS		BRANCH CHIEF:
PROJECT MANAGER:			PROJECT MANAGER:
LICENSING ASSISTANT:			LICENSING ASSISTANT:
			B. HARLESS

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NRC FORM 195 (2-76)

P. O. BOX 013100, MIAMI, FLORIDA 33101

FLORIDA POWER & LIGHT COMPANY

August 3, 1977
L-77-245

Office of Nuclear Reactor Regulation
Attention: Mr. Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Davis:

RE: St. Lucie Unit 1
Docket No. 50-335
Neutron Shielding

On November 29, 1976 (L-77-406), we submitted a plan for installing additional neutron shielding in the reactor vessel cavity at St. Lucie Unit 1. Your letter of April 29, 1977 requested additional information about our plan. The information you requested is attached.

Very truly yours,

R. E. Uhrig
Vice President

REU/MAS/pm

Attachment

cc: Mr. Norman C. Moseley, Region II
Robert Lowenstein, Esquire

ATTACHMENT

RE: St. Lucie Unit 1
Docket No. 50-335
Neutron Shielding

TABLE OF CONTENTS

I. NRC QUESTIONS OF 4/29/77

1. Clarify if the shield support structure has been designed to withstand the following load combination:

$$1.6S = D + E'$$

where D = moments and forces due to dead load of support structures, bags and contained water.

2. Clarify if the seismic excitation along three orthogonal directions was imposed simultaneously for the design of the shield support structure. The peak response from each direction may be combined by the square root of the sum of the squares (SRSS). Provide the vertical and two horizontal floor response spectra used in the analysis and describe the basis of their development.

3. Provide clear and legible copies of Figures 1 and 2. You may send full size drawings directly to the NRC Project Manager. Also, clarify the location of the sections shown in Figure 19.

4. Although the pipe break opening time is currently under review by the NRC staff, longitudinal break opening time of 5 milliseconds for a 30" diameter pipe would be acceptable without further justification. The longitudinal break opening time utilized in your report is significantly greater than 5 milliseconds. Therefore, you should evaluate the effect of a 5 millisecond break time or provide further justification for your originally proposed break time.

5. The report is not clear with respect to neutron and gamma dose rates. Throughout the report, the unit MR/hr is used for neutron dose rate. This unit is applicable only for x and gamma radiation. The unit for neutron dose equivalent rate is mrem/hr. To avoid possible confusion, the report should be revised to better characterize the gamma exposure rate, neutron dose equivalent rate and the summation of the two to provide dose equivalent rates (See Figure 17).

6. Provide an occupation radiation exposure budget (man-rem) for the proposed shield. The budget should separate the neutron dose from the gamma exposure where applicable and should include the following:

   (a) Man-rem doses received outside containment (e.g. streaming through containment penetrations such as the equipment hatch)

   (b) Man-rem doses to personnel inside containment during reactor operations considering routine maintenance and inspection procedures.

   (c) Man-rem exposures to personnel inside containment during refueling after the shield support structure is removed to its storage position. This is needed since the submittal only considered the exposure to personnel during removal and replacement of the shield. You should address the expected exposure to personnel inside containment from the support structures activation products during refueling operations while the structures are in the storage position.

7. It is not clear from the shielding analysis why additional neutron attenuation, provided by a thicker water shield, will not provide a significant dose rate reduction. The report states that despite the neutron dose rate attenuation from the proposed one foot thick shield, the dose rate at the operating level of the containment appears to be dominated by the neutrons which stream through the cavity depressurization and ventilation openings. The report should quantify this statement. Therefore, with reference to Figure 17, specify the fraction of the tabulated "shielded mr/hr" dose rate that is due to streaming from the aforementioned openings.

8. Provide an analysis of the exposure dose rate (mr/hr) evolved from 2.2 MEV neutron capture gamma-rays formed from neutron capture of the hydrogen in the water shield. The analysis should address the capture gamma-ray effects from all neutrons incident on the water shield including the incident thermal neutrons ($10^7 n/cm^2$-sec) and those fast neutrons interacting in the water shield that are eventually slowed down and captured.

9. Provide neutron streaming data taken during the power assention test program.

II. RESPONSE TO 4/29/77 NRC QUESTIONS ON NEUTRON SHIELDING

1. The shield support structure will be designed for the load combination
$$1.6\ S = D + E'$$
where the dead load D includes the weight of support structures, bags and contained water.

2. The shield support structure design will consider seismic excitation along three orthogonal directions imposed simultaneously. The peak response from each direction will be combined by SRSS. Attached are the vertical and horizontal (OBE) response spectra used in the analysis. The vertical spectra curve applies to all elevations and was used at the support elevation. In the horizontal directions, spectra curves at the support elevation were not available, so the maximum "g" envelope of the curves from the next upper (El 44.00') and lower (El 24.00') elevation was used. At a given elevation, the same curve applies to both E-W and N-S horizontal directions. The magnitude of the DBE response is defined as twice that produced by OBE excitation. The basis of the development of the floor response spectra is described in FSAR Section 3.7.1.

3. G-size prints of drawings SK-8770-AS-154 Sh 1 and 2 (Figures 1 and 2) have been transmitted to the NRC Project Manager, E. Reeves, under separate cover. Attached are marked-up copies of figures 18 and 19. The corrections shown on these figures will clarify the section locations.

4. The time required by the jet caused by a longitudinal break in the cold leg to reach the bottom of the shield support structure is estimated to be within a 7 or 8 msec range on the basis of a distance of 9 ft of travel. In our opinion the real opening time of the longitudinal break (to full open) will be in the range of 20 msec. Our opinion is based on the Battelle Memorial Institute tests results as stated in our prior submittal.

   Nevertheless, were a break opening time of 5 msec to be assumed as computed in CENPD 168 for a smaller break area, it would mean that the source of the jet would be fully open before the jet hits the shield, instead of having an initially smaller jet hitting the shield.

   In our analysis no credit was claimed for a reduced area of the jet as the breaks develops. The fully developed jet was used, and in this context the choice of the break opening time is immaterial. However credit was claimed for a reduction in reservoir pressure prior to the arrival of the fully developed jet at the shield. The choice of a break opening time does influence the time of depressurization of the 30" line. CE has shown that while the time required to depressurize a 30" line from the 2360 psi operating pressure to 1100 psi for a slot break is reasonably insensitive to the flow area opening time, it is roughly equal to half the opening time for break opening times between 7 and 13 msec. For a 5 msec opening time then it may be safely assumed that the depressurization time would be no longer than that required to depressurize the line for the longer opening times of 7 and 13 msec. These times are 4 and 6 msec respectively. Even for a 20 msec opening break the depressurization time would be of the order of 8 msec. Therefore it can be concluded that the reservoir feeding the jet when the jet hits the shield will be at the saturation pressure. Our choice of a 20 msec opening time results in a conservatism. A faster opening time would lead to faster depressurization and increased assurance that the jet hitting the shield would be fed by a reservoir at approximately 1100 psi.

5. The unit of neutron dose rate used in the calculation is mRem/hr.

6. Table 1 presents the occupation radiation exposure budget (man-rem) determined for the proposed shield assuming an 80% plant factor.

   The estimated yearly man-rem saved would by itself not be sufficient to warrant the expenditure of capital required for the shield design, fabrication, and installation, particularly as the exposures are very sensitive to the occupancy time assumed for various areas. In fact it is doubtful that as much time would be spent on the containment operating floor as that assumed, particularly with high dose rates, since little activity is required at this level, with most of the jobs being required at the lower levels. Thus yearly man-rem saved has probably been overestimated. The primary reason for installation of the shield is to minimize the neutron dose rates which would otherwise severely hamper potential maintenance and repair operations inside containment.

7. Larger depths of water would indeed provide larger attenuation of neutrons streaming directly upward or scattered upward through the water bags. Since no occupancy is present directly above the water bags, dose rates directly above them were not computed.

   The response at the refueling machine detectors (point no. 38 of Figure 17) is dominated by the neutrons which are reflected from the shield or miss the shield entirely and stream through the openings between the shield and the concrete walls. For this point 99% of the neutron dose rate is caused by streaming through the opening.

   For the other detectors, the dose rate due to neutron bypassing the shield is somewhat less than 99% but of the same order.

   This effect had been noted in prior neutronic analyses which employed a similarly configured shield, i.e., roughly the same extent of coverage at the same elevation above the flange, but of different material and thickness (PERMALI in 2-1/2 ft. thicknesses). This thicker shield, with roughly double the direct neutron attenuation through it, resulted in dose rate reduction at the refueling machine of approximately a factor of 20.

When the opening between the shield and concrete walls were further reduced, the reduction factor increased from 20 to more than 30, signifying that it is the streaming through the openings that dominates the neutron dose rates.

8. The total flux measured at St Lucie at a location immediately below the shield, was determined to be less than 10^7 n/cm^{2}sec. The measurement at this location, however, had a large uncertainty associated with it.

   To conservatively estimate the capture gamma production in the water bags, a conservative flux impinging on the bag has been derived by weighting the average thermal flux (E<0.45 ev) at the seal ring elevation, which was measured with better accuracy, by the solid angle subtended by the shield. The average thermal flux (E < 0.45 ev) measured at the seal ring elevation is 1.5x10^8 n/cm^{2}sec. Weighted by the solid angle subtended by the shield, the impinging thermal flux on the shield is computed to be approximately 1.5x10^8 $\frac{(1 - \cos 70^o)}{2}$ = 5 x 10^7 n/cm^{2}sec.

   The capture gamma source density is then computed utilizing a thermal capture cross section of 0.33 barns (see attached figure). Its value is 1.1 x 10^6 γ/cc sec.

   The capture gamma dose rate contribution at point #38 of Figure 17 is computed, again conservatively, by assuming a concentrated point source of strength equal to 7.0 x 10^{12} γ/sec located at distance of approximately 1500 cm. A 66% reduction is achieved by self attenuation of the capture γ's in the water. The resultant dose rate estimated in the very conservative manner outlined above is less than 250 mr/hr.

   Since the measured total flux is a factor of 5 less than that conservatively estimated for the thermal flux impinging on the bags, the actual dose rate, including the contribution from the neutrons above 0.45 ev is expected to be less than 50 mr/hr.

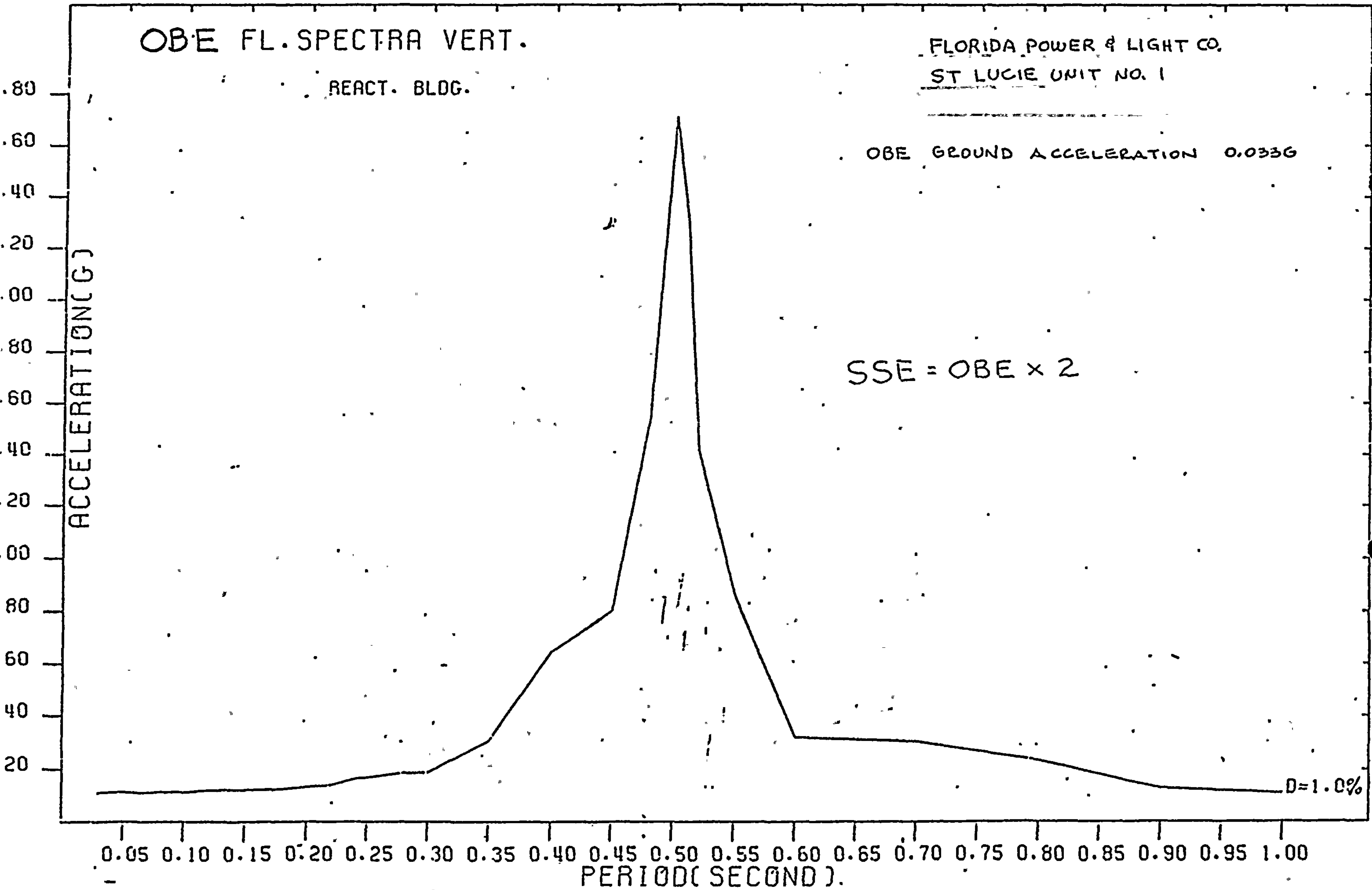
9. A report of neutron streaming data taken during the power ascension test program was sent to the NRC on April 25, 1977 (FPL letter L-77-126 from R. E. Uhrig to Dennis L. Ziemann).

TABLE 1

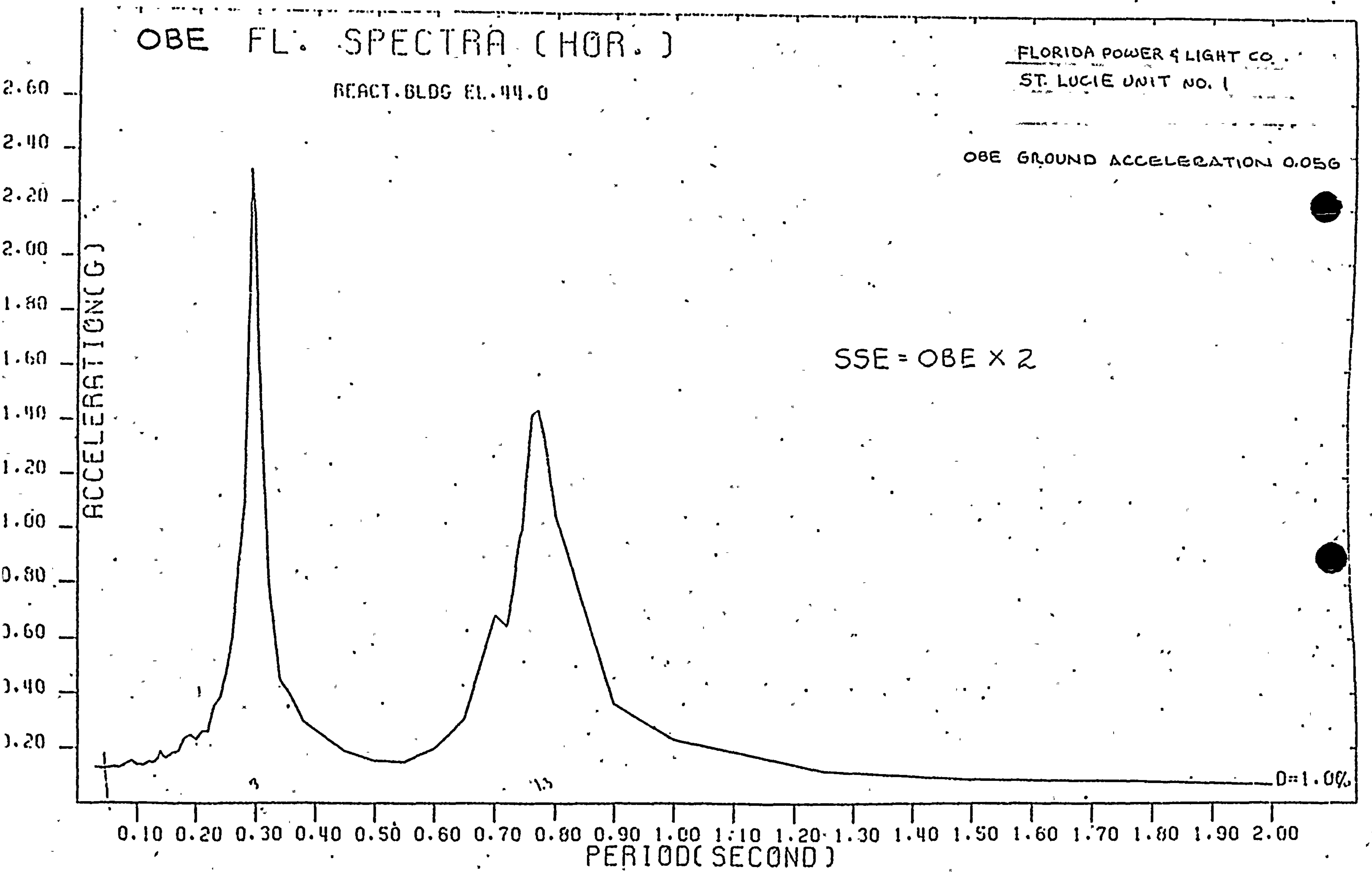
OCCUPATION RADIATION EXPOSURE BUDGET

	Avg. Neutron Dose Rate (mRem/hr)	Avg. Gamma Dose Rate (mr/hr)	Estimated Exposure Rate (man-hr/wk)	Exposure (man-rem/yr)
A. OUTSIDE CONTAINMENT				
1. No Shield	2.5	2.5	5	1.2
2. Shield	0.5	0.5	5	0.21
B. INSIDE CONTAINMENT				
1. Operating Floor - No Shield	2500	500	0.7	109
- Shield	150	100	0.7	9.1
2. Other Areas - No Shield	100	50	1.4	11.
- Shield	25	50	1.4	5.5
3. Refueling, Removal & Replacement of Shield	0	33	18 (b)	0.60
4. Stored Activated Support Structure(a)	0	0.5	720	0.72
C. ESTIMATED MAN-REM SAVED DUE TO SHIELD	-	-	-	105

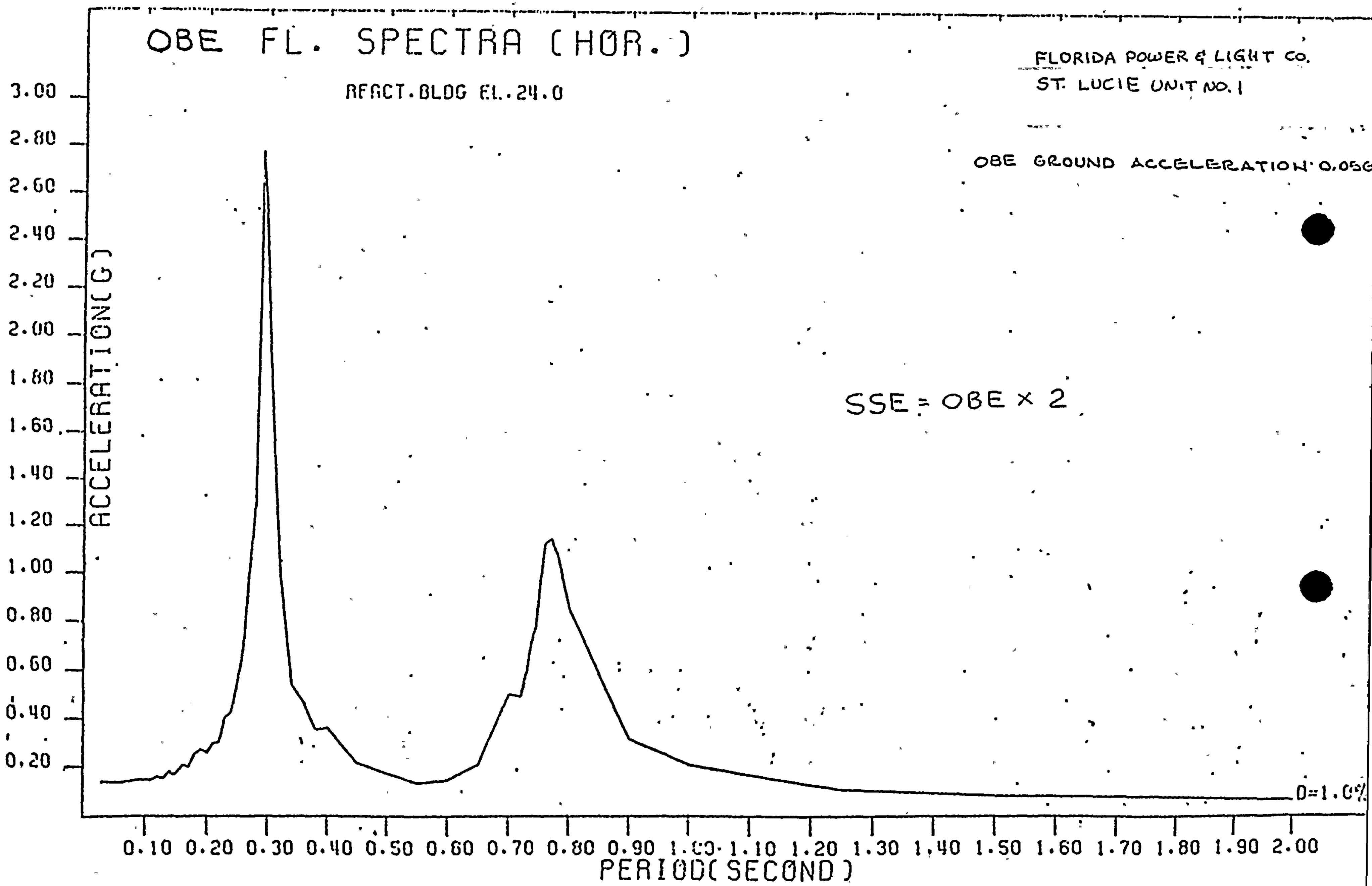
a)Assumes 4 people present for 15 days
b)one operation per year



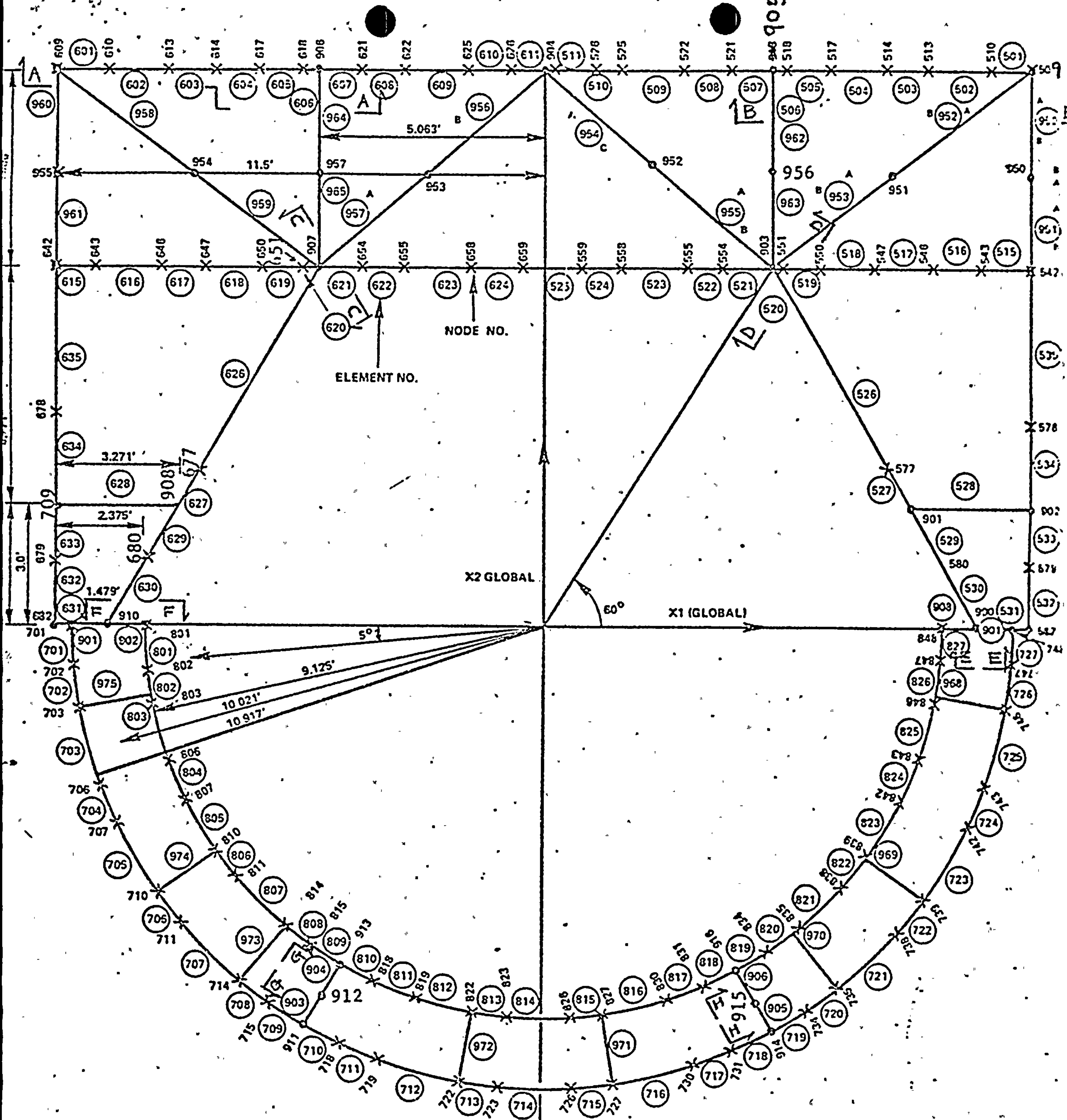
OBE FL. SPECTRA VERT.
REACT. BLDG.
FLORIDA POWER & LIGHT CO.
ST LUCIE UNIT NO. 1
OBE GROUND ACCELERATION 0.033G
SSE = OBE × 2
ACCELERATION(G)
.80
.60
.40
.20
.00
.80
.60
.40
.20
.00
80
60
40
20
D=1.0%
0.05 0.10 0.15 0.20 0.25 0.30 0.35 0.40 0.45 0.50 0.55 0.60 0.65 0.70 0.75 0.80 0.85 0.90 0.95 1.00
PERIOD(SECOND).



OBE FL. SPECTRA (HOR.)
REACT.BLDG EL.44.0
FLORIDA POWER & LIGHT CO.
ST. LUCIE UNIT NO. 1
OBE GROUND ACCELERATION 0.05G
SSE = OBE X 2
ACCELERATION(G)
2.60
2.40
2.20
2.00
1.80
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
D=1.0%
0.10 0.20 0.30 0.40 0.50 0.60 0.70 0.80 0.90 1.00 1.10 1.20 1.30 1.40 1.50 1.60 1.70 1.80 1.90 2.00
PERIOD(SECOND)



OBE FL. SPECTRA (HOR.)
REACT.BLDG EL.24.0
FLORIDA POWER & LIGHT CO.
ST. LUCIE UNIT NO. 1
OBE GROUND ACCELERATION 0.05G
SSE = OBE × 2
ACCELERATION(G)
3.00
2.80
2.60
2.40
2.20
2.00
1.80
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
D=1.0%
0.10 0.20 0.30 0.40 0.50 0.60 0.70 0.80 0.90 1.00 1.10 1.20 1.30 1.40 1.50 1.60 1.70 1.80 1.90 2.00
PERIOD(SECOND)

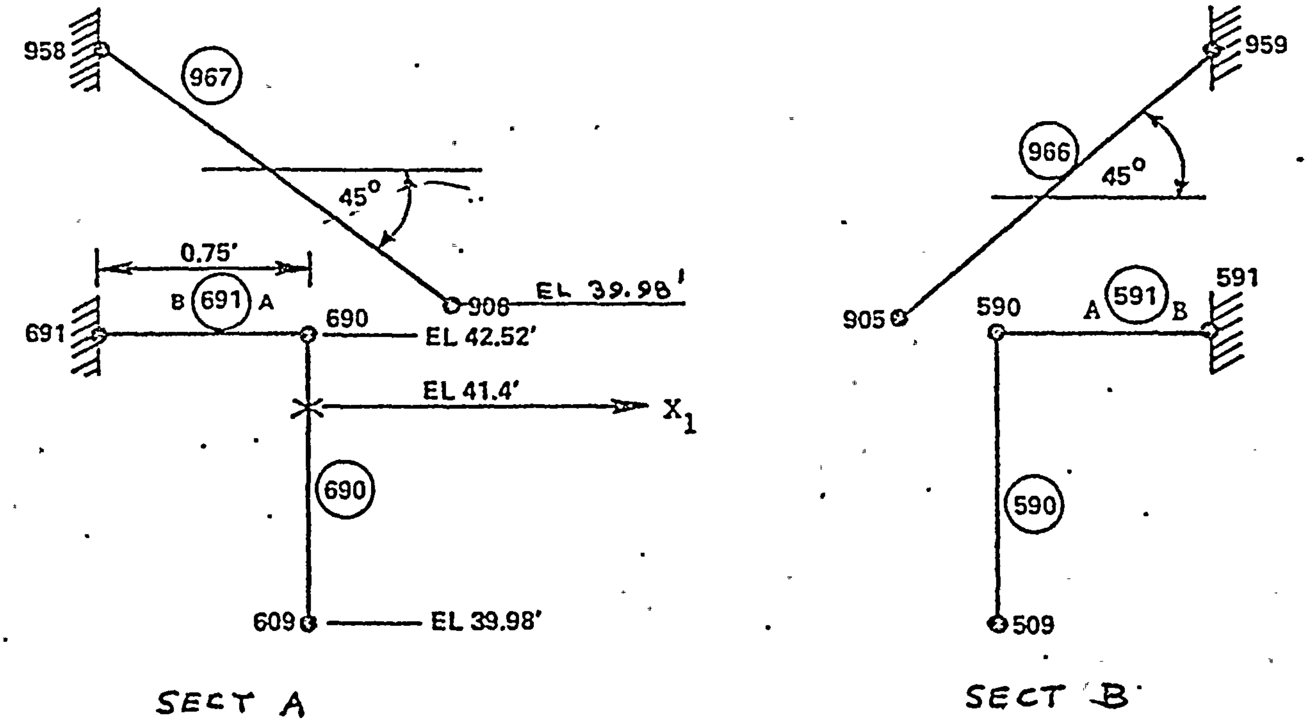


PLAN (EL 39.98')

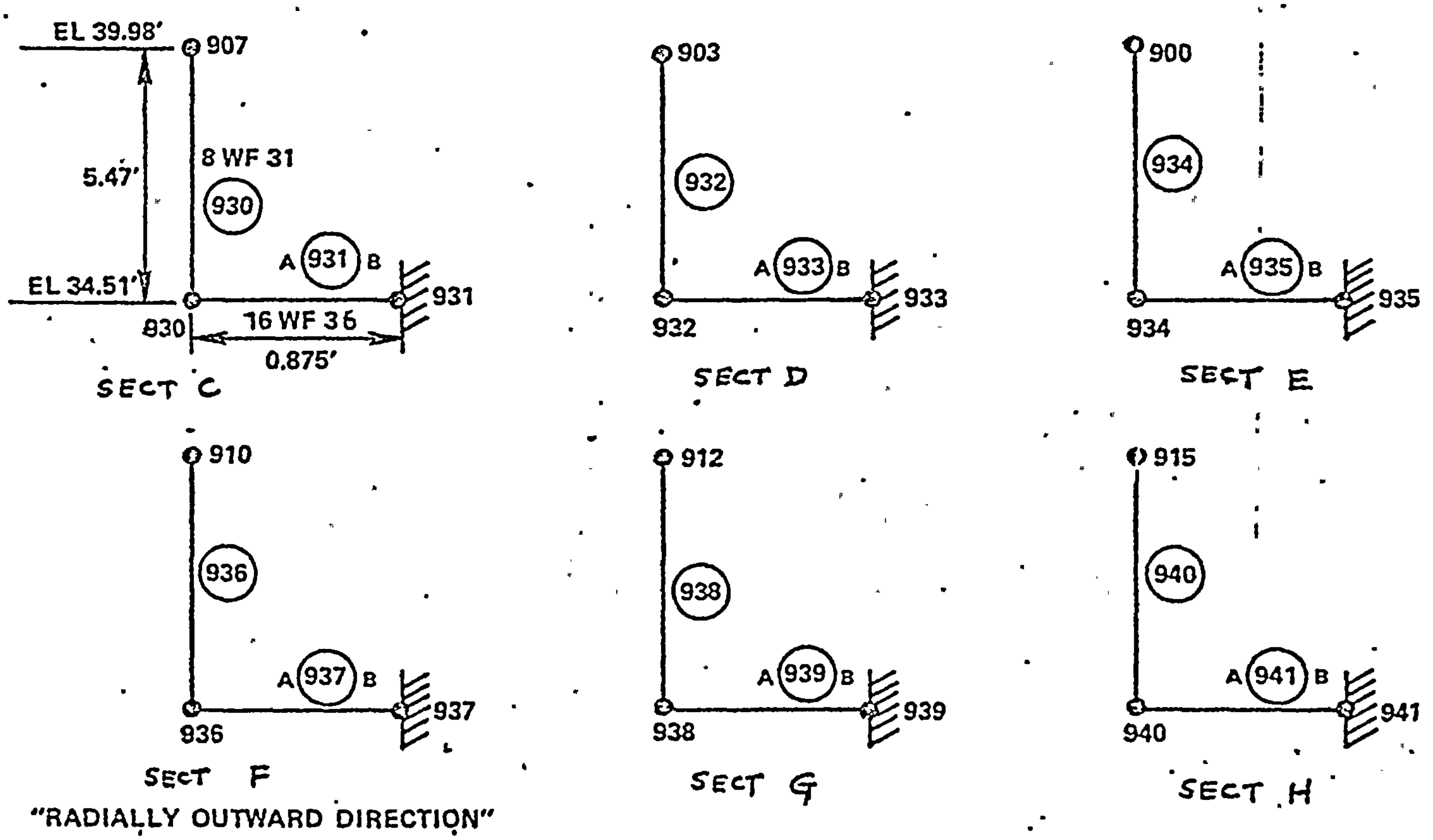
MODEL—NEUTRON SHIELD

FIGURE 18

SECTION "A-A"

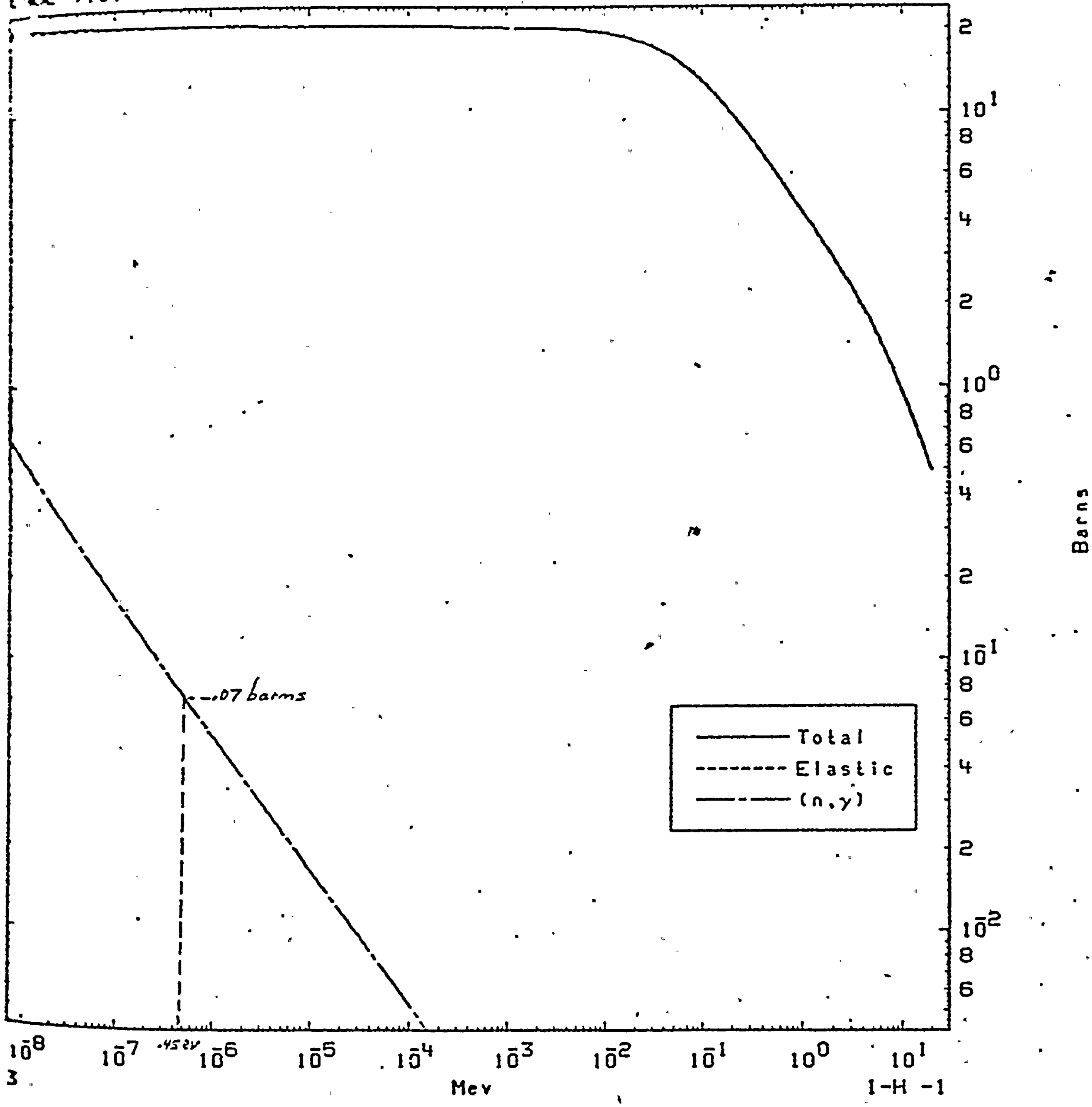


SECTION "B-B"



MODEL—NEUTRON SHIELD

FIGURE 19


2
10^1
8
6
4
2
10^0
8
6
4
2
10^{-1}
8
6
4
2
10^{-2}
8
6
Barns
--.07 barns
Total
Elastic
(n,γ)
10^{-8}
10^{-7}
.45ev
10^{-6}
10^{-5}
10^{-4}
10^{-3}
10^{-2}
10^{-1}
10^0
10^1
Mev

III. SCHEDULE

Completion of design	-	July 15, 1977
Material purchase	-	August 15, 1977
Material delivery	-	November 15, 1977
Installation	-	First scheduled unit shutdown of sufficient duration after material delivery.