POWER AUTHORITY OF THE STATE OF NEW YORK

10 COLUMBUS CIRCLE NEW YORK, N. Y. 10019

(212) 397-6200

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Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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U.S. NUCLEAR REGULATORY COMMISSION

Attention: Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Post Accident Shielding Analysis
NUREG-0578, Item 2.1.6.b
NUREG-0737, Item II.B.2

References: 1. NRC letter, D.G. Eisenhut to all Licensees of Operating Plants, dated October 31, 1980
2. PASNY letter, J.P. Bayne to T.A. Ippolito (JPN-81-26) dated April 17, 1981

Dear Sir:

Enclosed is the subject analysis which you requested in Reference 1. This analysis addresses all of your concerns except equipment qualification. As stated in Reference 2, equipment qualification is being addressed in the Authority's on-going efforts in response to I.E. Bulletin 79-01B.

The shielding analysis contains recommendations to augment shielding in the reactor building to reduce post-accident radiation fields and enhance accessibility. The Authority has reviewed these recommendations and determined that their implementation is not required to achieve compliance with the NUREG-0737 guidelines.

Should you or your staff have any questions, please contact us.

Very truly yours,

J. P. Bayne
Senior Vice President
Nuclear Generation

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POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

NUCLEAR DESIGN AND ANALYSIS DEPARTMENT

POST ACCIDENT
SHIELDING ANALYSIS

RESPONSE TO NRC-NUREG-0578
ITEM 2.1.6.b

By R.E. Deem *Rolf E. Deem* Date *March 29, 1981*

Supervising Nuclear Engineer
Nuclear Design and Analysis

James A. FitzPatrick Nuclear Power Plant

Post Accident
Shielding Review

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DESIGN REVIEW OF PLANT SHIELDING FOR PLANT ACCESS AND SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS

NRC Position:

With the assumptions of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core halogens, 100% of the core noble gases, and 1% of the core solid fission products are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the areas around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrumentation areas, in which personnel occupancy may be unduly limited or safety-related equipment unduly degraded by the radiation fields during the post-accident operations of these systems. Each licensee shall provide for adequate access to vital areas and protection of safety related equipment by design changes, increased permanent shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

NRC Clarification:

Assure adequate shielding of radiation from systems outside containment which may contain primary coolant or gases to assure access to vital areas or equipment. Field run piping is also to be evaluated where necessary (NRC letter of October 30, 1979). In the case of depressurized reactor coolant, the noble gas component of the radiation source may be excluded (NUREG-0737).

SHIELDING REVIEW AND ANALYSIS

Objective:

To investigate the adequacy of plant shielding against radiation from sources which are outside the primary containment, with the aim of determining area dose rates and location specific dose rates where access may be deemed necessary during post-accident conditions.

Scope of Review:

With the above assumptions and from consideration of post-accident operations and maintenance requirements, the following areas and equipment locations were considered to be of primary interest in the shielding analysis and review:

1. Control Room
2. Technical Support Center
3. Primary Coolant System Sampling Stations
4. Motor Control Centers

Scope of Review: (Continued)

5. Electrical Control Panels

6. Manual isolation and control valve locations for vital equipment

7. Chemistry Laboratory and counting room

8. Areas where emergency maintenance may be required, such that non-repair of the equipment may result in an additional burden on vital equipment used for mitigation purposes or result in an increased radiological hazard to personnel.

9. General access areas which may be required during post-accident conditions.

Radiation Level Guidance:

Areas requiring continuous occupancy: less than 15 mR/hr. (i.e., Control Room)

Areas requiring possible frequent access: less tnan 100 mR/hr. (i.e., Radwaste Control Panels)

Other areas which may require access for emergency maintenance or operational needs: shielding as required to keep exposures less than 10 CFR Part 20.

Systems considered to be the dominant sources:

1.	HPCI -	High Pressure Coolant Injection System
2.	LPCI -	Low Pressure Coolant Injection System (one mode of the RHR system)
3.	RHR -	Residual Heat Removal System
4.	CSS -	Core Spray System
5.	CAD -	Containment Atmosphere Dilution System (including Torus/drywell purge & vent system)
6.	RCIC -	Reactor Core Isolation Cooling System
7.	SGT -	Standby Gas Treatment System

Systems considered to be the dominant sources: (Continued)

8. Primary coolant sampling lines and other radioactive lines smaller than 2 inch nominal outside diameter. Where larger radioactive piping or equipment is in close proximity, such that it is the dominant source (i.e., smaller lines are less than 10% of the source), the sampling lines are not considered.

Shielding Source Terms:

Pressurized Liquid Systems

Noble Gases:	100% of core inventory
Halogens:	50% of core inventory
Remaining Fission Products:	1% of core inventory

Depressurized Liquid Systems

Halogens:	50% of core inventory
Remaining Fission Products:	1% of core inventory

Containment Air

Noble Gases:	100% of core inventory
Halogens:	25% of core inventory

A thorough review of all design-basis accident blowdown analyses indicate that the primary system will be depressurized to below 200 psia in under 25 minutes. Therefore, the depressurized liquid system source is utilized for all analyses performed during the mitigation phase of the accident where a liquid radiation source is present.

Dilution Volumes:

Liquid source volume dilution:

Primary System	80,000 gallons
Torus	750,000 gallons
Condensate Storage Tanks	200,000 gallons
Total -	1,030,000 gallons

Containment air source volume dilution:

Drywell Air Volume -	154,476 cubic feet
Torus Air Volume -	113,089 cubic feet
Total -	267,565 cubic feet

A thorough review of plant technical specifications and system descriptions indicates that at the onset of an accident, the initial supply of water to the emergency core cooling systems (HPCI and RCIC) is provided by the Condensate Storage Tanks (CST). With these systems operable, as they would be for any type of design-basis accident (large pipe break, small pipe break or failed-open safety relief valve), the CST volume would be injected into the primary system within 43 minutes. Consequently, the dilution effects of the CST are not factored into the analysis before 45 minutes into the mitigation phase of the accident.

Assumptions:

The following assumptions were made for this review, in light of the Lessons Learned from the TMI-2 accident:

1). No primary containment (drywell and torus) entry will be made during the mitigation phase of the postulated accident.

2). Containment isolation is in effect immediately after the accident commences; main steam lines are isolated and the main turbine has tripped.

3). The Reactor Water Clean-Up System is isolated immediately after the accident commences.

4). No radwaste processing of high-level primary fluids will take place during the mitigation phase.

5). The length of the mitigation phase is taken to be six (6) months. This time will also serve as the limit for integrated airborne dose rates in vital areas.

6). For system activity calculations, an instantaneous release of core radioactivity in the percentages given, as specified previously, is utilized. Concentrations will be assumed uniform at t=0 and no credit taken for decay (prior to t=0) or diffusion.

7). The status (mode) of various plant systems with regard to overall plant conditions is taken into account where access to vital areas or emergency operations or maintenance is required.

Assumptions: (Continued)

8). Primary coolant system sampling lines are only considered to be a significant source when personnel access is required to be in close proximity to them.

9). The airborne whole body radiation dose contribution from postulated primary containment design leakage was not included as a source in the results of the sheilding review. (Table VII and Figures 1 through 12)

Shielding Analytical Procedure:

The activity concentrations for the sources used were taken from Reference 4 and are shown in Tables I, II, and III. These sources were then converted to a multigroup energy source using the ACTGEN computer code (Reference 6, Appendix A) for various times during the mitigation phase of the accident. The energy sources are shown in Tables IV, V, and VI. The QAD-QC computer code (Reference 5, Appendix B) was then used to determine the dose rates for various locations and areas throughout the plant. When there is a dose point location that involves multiple sources, the dose rate contribution from each source is calculated and the contributions summed to determine the location dose rate. No credit for shielding from other equipment or obstructions was taken and all piping was assumed to be bare (no insulation) in the models used in the analysis. Air dose rates in the Control Room were evaluated using the 0-2 hour X/Q and breathing rate throughout the six (6) month period. The Standby Gas Treatment System is assumed operating and charcoal filter efficiencies for halogen removal are taken as 0.9 for the Standby Gas Treatment and Control Room intake systems. The primary containment design leak rate of 0.5%/day was also assumed. For the Technical Support Center, similar assumptions and parameters were used, except that no emergency charcoal filters for halogen removal are assumed to be available at the air intake.

Air dose rates in the reactor building were calculated assuming the reactor building ventilation system is isolated and operating in the recirculation mode. The Standby Gas Treatment system is also in operation. There are only roughing filters on the reactor building ventilation system and they are assumed to provide no filtration for the reactor building atmosphere.

The DRAGON-3 computer code (Ref. 8) was used to calculate these air dose rates and all integrated dose rates.

Analytical Results:

Utilizing the above assumptions and procedure, dose rates were calculated at selected locations in the plant. Table VII depicts the dose rates calculated at these locations. These dose point locations are shown on Figures 1 through 12. Also shown on these figures are the primary radiation sources considered in the analysis above the 272 ft. elevation in the Reactor Building.

Analytical Results: (Continued)

To determine area dose rates within the Reactor Building, the two hour radiation source was utilized (Table V and VI). With the conservative assumption of uniform mixing at the onset of the accident and other assumptions, it is felt that the two hour source would be representative of the maximum radiation fields encountered in the Reactor Building. The area dose rates are shown on Figures 1 through 12 according to the following criteria:

Area Designation		Dose Rate
A.....	Less than	100 mR/hr.
B.....	Greater than	100 mR/hr.
	but less than	5 R/hr.
C.....	Greater than	5 R/hr.
	but less than	20 R/hr.
D.....	Greater than	20 R/hr.

Air dose rates and the 6 month integrated doses for the Control Room and Technical Support Center are shown in Tables VIII and IX. Table X depicts the air dose rates in the Reactor Building, which are not included on Figures 1 through 12 or on Table VII.

Discussion of Results By Elevation

Reactor Building, Elevation 227' (Figure 1)

The Cresent Area is inaccessible for the duration of the mitigation phase (6 months) of the postulated accident. This is due to the high concentration of safety related pumps and systems piping which contain primary coolant. An operational review of the equipment in this area indicate that access is not vital.

Reactor Building, Elevation 272' (Figure 2)

The major source contributors on the 272 ft. elevation (Reactor Building Elevation 272' Figure 2) are the containment and core spray piping, the Reactor Building/Torus vacuum breaker, and the torus purge line. All areas directly above the torus are influenced by the large volumetric liquid and gaseous source contained in the torus. Despite the attenuation afforded by the 2 ft. concrete floor, the dose rate due to the torus is approximately 86 R/hr at 2 hours. The gaseous portion of the torus source is the predominant dose contributor from the torus. This is due to the self-shielding characteristic of the liquid volume source. Consequently, since the high energy gaseous source decays rapidly, the torus dose rate contribution to the 272 ft. elevation is no longer significant after 24 hours (i.e., 1.4 R/hr). Consequently, access to certain areas of this elevation will improve substantially after 24 hours.

Reactor Building, Elevation 300' (Figure 3)

Sources which contribute to the dose rates on the 300 ft. elevation (Figure 3) are the core and containment spray lines. The west side of this elevation is relatively inaccessible due to the presence of the exposed core and containment spray lines in this area. The existing sampling station located in the southeast quadrant of this elevation is not readily accessible due to the close proximity of the RHR head spray piping. All contributing sources on this elevation are volumetric liquid sources. Accessibility to this elevation will be enhanced as the mitigation phase progresses, but not as rapidly as would be the case if gaseous sources were involved. The sources, however, do not in any way hamper accessibility requirements to the MG set room. It is envisioned that this area will provide the primary access route from the Administration Building into the Reactor Building, if required, on an emergency basis.

Reactor Building Elevation 326' (Figure 4)

The major sources contributing to the dose rates on the 326 ft. elevation (Figure 4) are the portions of the drywell vent and purge lines up to the outermost containment isolation valve, as well as the RHR head spray piping. The dose contributions due to the core spray and containment spray piping directly below this elevation on the west side of the Reactor Building

Reactor Building Elevation 326' (Figure 4) (Continued)

have been included in the stated dose rates. The results show that limited access to the 600 volt switchgear and MCC's on the west side of this elevation can be achieved on an emergency basis if required.

Reactor Building Elevation 344' (Figure 5)

The dose contributor on the 344 ft. elevation is the normal ventilation filter train of the Reactor Building ventilation system. When high radiation signals from the building's exhaust ducts trigger the intake and exhaust butterfly valves to close, the Reactor Building ventilation system goes into a recirculation mode. During post-accident operation, these filters could become a source of radioactivity due to the buildup of airborne particulate activity on the filters. This activity buildup would result from the conservatively assumed 0.5%/day primary containment design leakage rate. However, since these filters are intake roughing filters, their ability to entrain particulate activity is quite limited. Consequently, from a design basis analytical standpoint, the area immediately surrounding these filter banks is conservatively assumed to buildup to, but not exceed 20 R/hr., after the accident. There are no other significant dose contributors on this elevation of the Reactor Building.

Reactor Building Elevation 369' (Figure 6)

The 369 ft. elevation contains no major radioactivity sources as a result of the postulated accident. Therefore, this elevation is accessible during the mitigation phase. The removable shield above the reactor cavity on this elevation provides sufficient shielding to preclude the drywell from being a dominant source.

General

For all of the subject elevations in the Reactor Building, the drywell was not a contributing source since the thickness of the shield wall surrounding it is more than adequate.

Other radioactive sources and areas investigated included the halogen buildup on filters of the Standby Gas Treatment System, exterior areas of the plant, and the accessibility to the temporary post-accident primary system sampling station located in the Radwaste Building.

Variation in dose rates from halogen buildup accumulated on the Standby Gas Treatment System filters is shown on Figure 2 (B to D zone accessibility). The dose rates in this area peak approximately 2 weeks after the onset of the postulated accident, assuming the system is in continuous operation subsequent to t=0.

Exterior areas of the plant which would become radiological hazards are also depicted on Figure 2. This is due totally to the radiation sources in the Crescent Area below. Even with 3 feet 10 inches of concrete shielding, the heavy concentration of radiation sources in the Cresent Area make these exterior areas an accessibility zone B, 2 hours after the onset of the accident.

The temporary post-accident Primary Coolant Sampling System, located in the Radwaste Building was analyzed to determine accessibility. The only sources in this area are the sampling lines themselves. Results indicate that the dose rate would be apprximately 11 R/hr six feet from the sampling station 1 hour after the onset of the accident with 0.25 inch of lead on the horizontal sections of the sampling lines near the sample sink. At 2 hours, and with the same constraints, the dose rate is 8 R/hr. Without the shielding, the dose rates would be 19 R/hr and 14 R/hr, respectively.

The permanent location of the Post-Accident Primary Coolant Sampling System, represented by Detector #10 on Table VII and Figure 3, is adequate. The radiation sources in the Reactor Building provide no major contributing dose to this location to prohibit accessibility.

Recommendations:

Results of the shielding analysis and review have identified certain items in the plant that present radiological problems from the standpoint of accessibility and possible operational requirements. Implementation of the following proposed modifications would enhance accessibility in the plant during the mitigation phase of the accident.

Areas which were identified exterior to the plant to be a radiological hazard during the mitigation phase of the accident should be administratively controlled. An administrative procedure should be developed which addresses this situation if it is not currently addressed in existing administrative procedures.

The isolation valve on the torus purge line, located in the northwest quadrant of the Reactor Building on the 272 ft. elevation, should be relocated to an area below the 272 ft. elevation floor, or shielded up to the existing isolation valve with the equivalent of one foot of concrete. This modification would reduce the relatively high background radiation level in the radiochemistry laboratory (Detector #3 on Table VII and Figure 2), and allow access early in the mitigation phase of the accident to MCC's 162 and 143 located near the air lock on that elevation.

Recommendations: (Continued)

Another item of concern is the Reactor Building/Torus vacuum breaker located in approximately the same area of the reactor building. This vacuum breaker should also be relocated or shielded with the equivalent of one foot of concrete. Emergency access to the east side of the reactor building on this elevation would be enhanced at later times during the mitigation phase of the accident, if this large volumetric radiation source is significantly reduced. At 24 hour after the onset of the accident, the dose rate contribution from this piece of equipment is approximately 35 R/hr. If left as it now exists, it will replace the torus as the dominant source contributor in this area and when added to the other contributing sources, would hinder emergency access.

Conclusions:

The analysis and shielding review performed in this report meets the intent and commitment requirements of NUREG-0578, NUREG-0737, and other associated NRC guidance concerning post-accident plant shielding and access requirements. Thorough usage of design documentation (i.e., drawings, technical specifications, etc.) has been made to substantiate the shielding calculations. Verification of plant specific data was made by walkthroughs and discussions with plant operating personnel. Postaccident access requirements have been developed in conjunction with requirements specified by plant operating personnel. These post-accident access requirements have been based upon emergency situations which may necessitate entry into the Reactor Building after a postulated accident.

Conclusions (Continued)

With implementation of the two permanent shielding or relocation modifications, access to the 272 ft. elevation of the reactor building under emergency conditions would be improved. Radioactive sources have also been identified for possible application of temporary shielding to plant personnel if access to these areas is deemed necessary. Use of temporary shielding would be dependent however, on duration of access and plant conditions at that time, from both an operational and radiological viewpoint. In all cases, both interior and exterior to the Reactor Building, access will be administratively controlled. Health physics personnel will monitor and possibly survey all areas requiring access.

The Control Room will have a safe and low radiation level at all times. The Technical Support Center will also meet the 15 mR/hr criterion. Airborne radiation dose rates calculated for the Reactor Building could seriously hinder access. However, these results are based on an assumed conservative 0.5%/day primary containment leakage rate. It is felt that this is a highly conservative leakage rate and that the Standby Gas Treatment System would greatly reduce the airborne radioactivity concentration in the reactor building if a more realistic leakage rate is considered. Inleakage of air to the Reactor Building of approximately 6,000 CFM during operation of the Standby Gas Treatment System would also constitute a substantial additive dilution volume, thereby further reducing airborne radioactivity concentrations.

Conclusions (Continued)

As a result of this shielding analysis, in conjunction with the proposed modifications, it is felt that the Reactor Building has been made as radiologically accessible as is practical. Additional modifications to further reduce the calculated dose rates would necessitate extensive redesign of the major plant systems with little radiological benefit.

TABLE I

Pressurized Liquid Source *

	Isotope	Activity uCi/cc
Noble Gases	Kr-83m	1.87E3
	Kr-85m	4.05E3
	Kr-85	1.87E2
	Kr-87	7.49E3
	Kr-88	1.06E4
	Kr-89	1.25E4
	Xe-131m	1.12E2
	Xe-133m	1.25E2
	Xe-133	3.49E4
	Xe-135m	1.06E4
	Xe-135	6.13E3
	Xe-137	2.93E4
	Xe-138	2.74E4
Halogens	Br-83	9.36E2
	Br-84	1.47E3
	Br-85	2.03E3
	Br-87	3.44E3
	I-131	9.05E3
	I-132	1.31E4
	I-133	1.50E4
	I-134	1.94E4
	I-135	1.53E4
	I-136	6.26E3
Particulates	Se-81	9.359E0
	Se-83	9.359E0
	Se-84	9.359E0
	Rb-88	1.059E2
	Rb-89	1.373E2
	Rb-90	1.684E2
	Rb-91	1.745E2
	Rb-92	1.684E2
	Sr-89	1.372E2
	Sr-90	1.559E1
	Sr-91	1.804E2
	Sr-92	1.932E2
	Sr-93	2.306E2
	Sr-94	2.245E2

* At reactor power level of 2436 MW(t)

TABLE I (cont'd)

Isotope	Activity uCi/cc
Y-90	1.620E1
Y-91m	1.121E2
Y-91	1.932E2
Y-92	2.058E2
Y-93	2.370E2
Y-94	2.493E2
Y-95	2.816E2
Zr-95	2.918E2
Zr-97	2.867E2
Nb-95m	5.811E0
Nb-95	2.918E2
Nb-97m	2.867E2
Nb-97	3.046E2
Mo-99	3.123E2
Mo-101	2.918E2
Mo-102	2.739E2
Mo-105	1.059E2
Tc-99m	2.739E2
Tc-101	2.918E2
Tc-102	2.867E2
Tc-105	2.058E2
Ru-103	2.557E2
Ru-105	2.183E2
Ru-106	1.185E2
Ru-107	1.246E2
Rh-103m	2.493E2
Rh-105m	4.608E2
Rh-105	1.871E2
Rh-106	1.246E2
Rh-107	1.310E2
Sn-127	1.620E1
Sn-128	2.688E1
Sn-130	5.811E1
Sb-127	1.932E2
Sb-128m	2.816E1
Sb-129m	5.171E1
Sb-130	1.185E2
Sb-131	1.559E2
Sb-132	1.684E2
Sb-133	1.185E2
Te-127	1.932E1
Te-129m	8.115E0
Te-129	4.915E1
Te-131m	3.865E1
Te-131	1.497E2
Te-132	2.611E2
Te-133m	1.871E2
Te-133	1.121E2
Te-134	2.918E2

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TABLE I (cont'd)

Isotope	Activity uCi/cc
Cs-137	2.245E1
Cs-138	3.174E2
Cs-139	3.123E2
Cs-140	2.688E2
Cs-142	9.984E1
Ba-137m	2.06E1
Ba-139	3.13E2
Ba-140	2.92E2
Ba-141	3.05E2
Ba-142	2.44E2
La-140	3.12E2
La-141	3.05E2
La-142	2.69E2
La-143	2.56E2
Ce-141	3.05E2
Ce-143	2.62E2
Ce-144	2.12E2
Ce-145	1.69E2
Ce-146	1.31E2
Pr-143	2.56E2
Pr-144	2.19E2
Pr-145	1.69E2
Pr-146	1.37E2
Nd-147	1.06E2
Nd-149	6.87E1
Nd-151	3.31E1
Pm-147	4.18E1
Pm-149	8.74E0
Pm-151	3.56E1
Sm-153	5.13E1

TABLE II

De-Pressurized Liquid Source *

	Isotope	Activity uCi/cc
Halogens	Br-83	9.36E2
	Br-84	1.47E3
	Br-85	2.03E3
	Br-87	3.44E3
	I-131	9.05E3
	I-132	1.31E4
	I-133	1.50E4
	I-134	1.94E4
	I-135	1.53E4
	I-136	6.26E3
Particulates	Se-81	9.359E0
	Se-83	9.359E0
	Se-84	9.359E0
	Rb-88	1.059E2
	Rb-89	1.372E2
	Rb-90	1.684E2
	Rb-91	1.745E2
	Rb-92	1.684E2
	Sr-89	1.372E2
	Sr-90	1.559E2
	Sr-91	1.804E2
	Sr-92	1.932E2
	Sr-93	2.306E2
	Sr-94	2.245E2
	Y-90	1.620E1
	Y-91m	1.121E2
	Y-91	1.932E2
	Y-92	2.058E2
	Y-93	2.370E2
	Y-94	2.493E2
	Y-95	2.816E2
	Zr-95	2.918E2
	Zr-97	2.867E2
	Nb-95m	5.188E0
	Nb-95	2.918E2
	Nb-97m	2.867E2
	Nb-97	3.046E2
	Mo-99	3.123E2
	Mo-101	2.918E2
	Mo-102	[illegible]39E2
	Mo-105	1.059E2

TABLE II (cont'd)

Isotope	Activity uCi/cc
Tc-99m	2.739E2
Tc-101	2.918E2
Tc-102	2.867E2
Tc-105	2.058E2
Ru-103	2.557E2
Ru-105	2.188E2
Ru-106	1.185E2
Ru-107	1.246E2
Rh-103m	2.493E2
Rh-105m	4.608E2
Rh-105	1.871E2
Rh-106	1.246E2
Rh-107	1.310E2
Sn-127	1.620E1
Sn-128	2.688E1
Sn-130	5.811E1
Sb-127	1.932E2
Sb-128m	2.816E1
Sb-129m	5.171E1
Sb-130	1.185E2
Sb-131	1.599E2
Sb-132	1.684E2
Sb-133	1.185E2
Te-127	1.932E1
Te-129m	8.155E0
Te-129	4.915E1
Te-131m	3.865E1
Te-131	1.497E2
Te-132	2.611E2
Te-133m	1.871E2
Te-133	1.121E2
Te-134	2.918E2
Cs-137	2.245E1
Cs-138	3.174E2
Cs-139	3.123E2
Cs-140	2.688E2
Cs-142	9.984E1
Ba-137m	2.06E1
Ba-139	3.13E2
Ba-140	2.92E2
Ba-141	3.05E2
Ba-142	2.44E2
La-140	3.12E2
La-141	3.05E2
La-142	2.69E2
La-143	2.56E2
Ce-141	3.05E2
Ce-143	2.62E2
Ce-144	2.19E2
Ce-145	1.69E2
Ce-146	1.31E2

* At reactor level of 2436 Mwt.

TABLE II (cont'd)

Isotope	Activity uCi/cc
Pr-143	2.56E2
Pr-144	2.19E2
Pr-145	1.69E2
Pr-146	1.37E2
Nd-147	1.06E2
Nd-149	6.87E1
Nd-151	3.31E1
Pm-147	4.18E1
Pm-149	8.74E0
Pm-151	3.56E1
Sm-153	5.13E1

* At reactor power level of 2436 Mwt

TABLE III

Containment Atmosphere Source

	Isotope	Activity uCi/cc
Noble Gases	Kr-83m	9.65E2
	Kr-85m	2.08E3
	Kr-85	9.65E1
	Kr-87	3.85E3
	Kr-88	5.46E3
	Kr-89	6.43E3
	Xe-131m	5.78E1
	Xe-133m	6.43E1
	Xe-133	1.79E4
	Xe-135m	5.46E3
	Xe-135	3.15E3
	Xe-137	1.50E4
	Xe-138	1.41E4
Halogens	Br-83	2.41E2
	Br-84	3.77E2
	Br-85	5.23E2
	Br-87	8.84E2
	I-131	1.33E3
	I-132	3.38E3
	I-133	3.86E3
	I-134	4.98E3
	I-135	3.94E3
	I-136	1.61E3

TABLE IV

Pressurized Liquid Source

TOTAL SPECIFIC ACTIVITY (Mev/cc-sec) AT ENERGY (Mev)

TIME	.40	.80	1.30	1.70	2.20	2.50	3.50
.5hr	3.423E8	2.266E9	1.065E9	9.622E8	5.821E8	5.936E8	1.140E8
1.0hr	3.350E8	1.792E9	9.042E8	7.942E8	3.501E8	4.829E8	9.502E7
2.0hr	3.262E8	1.240E9	6.776E8	5.306E8	2.306E8	3.099E8	5.230E7
8.0hr	2.960E8	4.642E8	2.663E8	1.961E8	6.254E7	4.764E7	7.874E6
24.0hr	2.380E8	2.256E8	5.015E7	5.083E7	6.148E8	1.928E6	1.469E5
4 days	1.450E8	6.350E7	2.631E6	1.617E7	4.264E5	9.840E5	0.0
14 days	5.120E7	2.898E7	8.375E5	9.072E6	5.014E4	5.820E5	0.0
30 days	1.118E7	1.853E7	3.927E5	3.830E6	1.649E3	2.447E5	0.0
90 days	3.119E5	9.826E6	1.257E5	2.118E5	0.0	9.486E3	0.0
180 days	9.465E4	4.850E6	9.430E4	6.332E4	0.0	7.239E1	0.0

TABLE V

Depressurized Liquid Source

TOTAL SPECIFIC ACTIVITY (Mev/cc-sec) AT ENERGY (Mev)

TIME	.40	.80	1.30	1.70	2.20	2.50	3.50
.5hr	1.235E8	1.956E9	1.053E9	8.806E8	1.060E8	1.061E8	1.132E8
1.0hr	1.192E8	1.577E9	8.932E8	7.260E8	9.380E7	7.840E7	9.502E7
2.0hr	1.161E8	1.093E9	6.690E8	4.788E8	7.570E7	2.670E7	5.230E7
8.0hr	1.112E8	4.173E8	2.644E8	1.844E8	2.994E7	1.250E6	7.874E6
24.0hr	1.037E8	2.164E8	5.011E7	5.061E7	5.542E6	1.119E6	1.469E5
4 days	7.862E7	6.348E7	2.631E6	1.617E7	4.264E5	9.840E5	0.0
14 days	3.431E7	2.898E7	8.875E5	9.072E6	5.014E4	5.820E5	0.0
30 days	8.922E6	1.853E7	3.927E5	3.830E6	1.649E3	2.447E5	0.0
90 days	3.156E5	9.826E6	1.257E5	2.188E5	0.0	9.486E3	0.0
180 days	9.463E4	4.850E4	9.430E4	6.332E4	0.0	7.239E1	0.0

TABLE VI

Containment Atmosphere Source

TOTAL SPECIFIC ACTIVITY (Mev/cc-sec) AT ENERGY (Mev)

TIME	.40	.80	1.30	1.70	2.20	2.50	3.50
.5hr	1.382E8	6.408E8	2.945E8	3.227E8	2.712E8	3.005E8	5.218E7
1.0hr	1.355E8	4.897E8	2.540E8	2.618E8	1.557E8	2.453E8	4.536E7
2.0hr	1.303E8	3.258E8	1.872E8	1.595E8	9.907E7	1.578E8	2.575E7
8.0hr	1.103E8	1.068E8	5.941E7	4.826E7	2.425E7	2.390E7	4.029E6
24.0hr	8.509E7	4.570E7	1.163E7	8.301E6	1.532E6	4.165E5	7.564E4
4 days	5.187E7	7.984E6	6.588E3	8.564E4	6.935E2	7.114E3	0.0
14 days	1.695E7	2.022E6	0.0	2.732E1	0.0	0.0	0.0
30 days	3.148E6	5.113E5	0.0	0.0	0.0	0.0	0.0
90 days	1.318E4	2.957E3	0.0	0.0	0.0	0.0	0.0
180 days	1.057E1	1.300E0	0.0	0.0	0 0	0 0	0 0

<u>TABLE VII</u>

<u>Reactor Building Dose Rates (R/hr.)</u>

Detector No. Dose Pt.	Time	1/2 hr	1hr	2hr	8hr	24hr	4 days	14 days	30 days	90 days	180 days
1		9.48	6.2	4.0	1.0	.15	0.0	0.0	0.0	0.0	0.0
2		1.06	.64	.40	.083	.007	0.0	0.0	0.0	0.0	0.0
3		.97	.64	.40	.083	.007	0.0	0.0	0.0	0.0	0.0
4		14.3	9.45	5.98	1.30	.18	.04	.02	0.0	0.0	0.0
5		75.11	49.4	28.3	6.82	.87	.16	.07	0.0	0.0	0.0
6		132.6	87.0	55.0	11.8	1.35	.19	.09	0.0	0.0	0.0
7		54.8	38.0	25.8	9.06	3.14	1.02	.47	0.0	0.0	0.0
8		907.8	617.5	417.7	142.5	119.9	37.5	17.1	--	--	--
9		31.6	22.3	15.5	6.0	2.19	.73	.34	.16	.06	.026
10		.15	.103	.071	.027	.010	.003	.002	.001	.0002	.0001
11		21.2	15.0	10.37	4.06	1.55	.51	.233	.110	.039	.019
12		198.0	139.7	97.1	37.9	14.5	4.8	2.31	1.05	.368	.169
13		105.5	74.4	51.7	20.2	7.7	2.2	1.15	.54	.19	.09
14		1.50	1.06	.75	.293	.112	.038	.018	.008	.003	.001
15		1072.5	755.8	525.2	205.1	77.6	26.0	11.7	5.5	2.0	.94
16		5.97	4.0	2.6	.82	.26	.08	.02	.004	0.0	0.0
17		1.95	1.28	.80	.17	.057	0.0	0.0	0.0	0.0	0.0
18		1.44	.962	.637	.2	.06	.02	.006	.001	0.0	0.0
19		18.4	12.9	9.0	.35	1.4	.47	.22	.10	.04	.02
20		4.56	3.2	2.3	.87	.35	.12	.06	.04	.01	.004

TABLE VIII

ACCIDENT AIR DOSE RATES IN

THE CONTROL ROOM

Time After Accident (hr)	Dose Rate (mrem/hr)	
	Gamma	Beta
0.5	.001	.013
1	.003	.037
2	.008	.095
8	.022	.40
24	.021	.60
96	.006	.30
336	.002	.08
720	.0002	.01

6-Months Continuous Occupancy Integrated Dose (rem)		
Tyroid	Gamma	Beta
2.6	.002	.11

TABLE IX

ACCIDENT AIR DOSE RATES IN TECHNICAL SUPPORT CENTER (TSC) AREA

Time After Accident (hr)	Dose Rate (mrem/hr)	
	Gamma	Beta
0.5	.0035	.07
1	.0075	.15
2	.013	.27
8	.022	.62
24	.017	.69
96	.004	.32
336	.001	.09

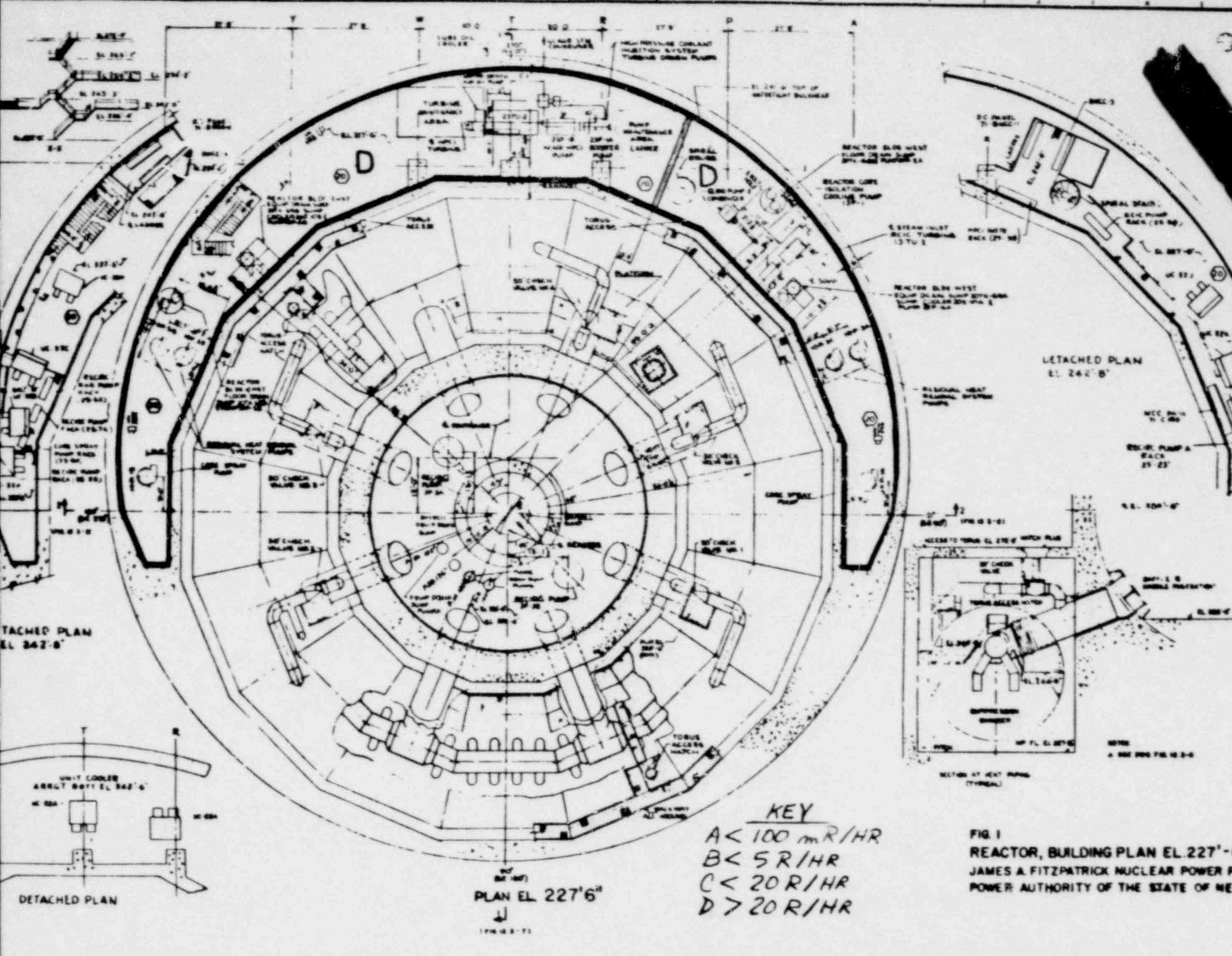
6-Months Continuous Occupancy Integrated Dose (rem)		
Tyroid	Gamma	Beta
28	.002	.12

TABLE X

ACCIDENT AIR DOSE RATES IN THE REACTOR BUILDING

Time After Accident (hr)	Dose Rate (rem/hr) Gamma	Beta
0.5	5.30 E+01	1.60 E+02
1	8.20 E+01	2.50 E+02
2	1.10 E+02	3.70 E+02
8	1.30 E+02	6.20 E+02
24	7.90 E+01	5.90 E+02
96	2.60 E+01	2.70 E+02
336	8.00 E+00	7.70 E+01
720	1.50 E+00	1.40 E+01
2160	8.70 E-03	1.90 E+00
4320	2.00 E-03	1.20 E+00

POOR ORIGINAL



[redacted]

FIG. 1
REACTOR, BUILDING PLAN EL.227'-6"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

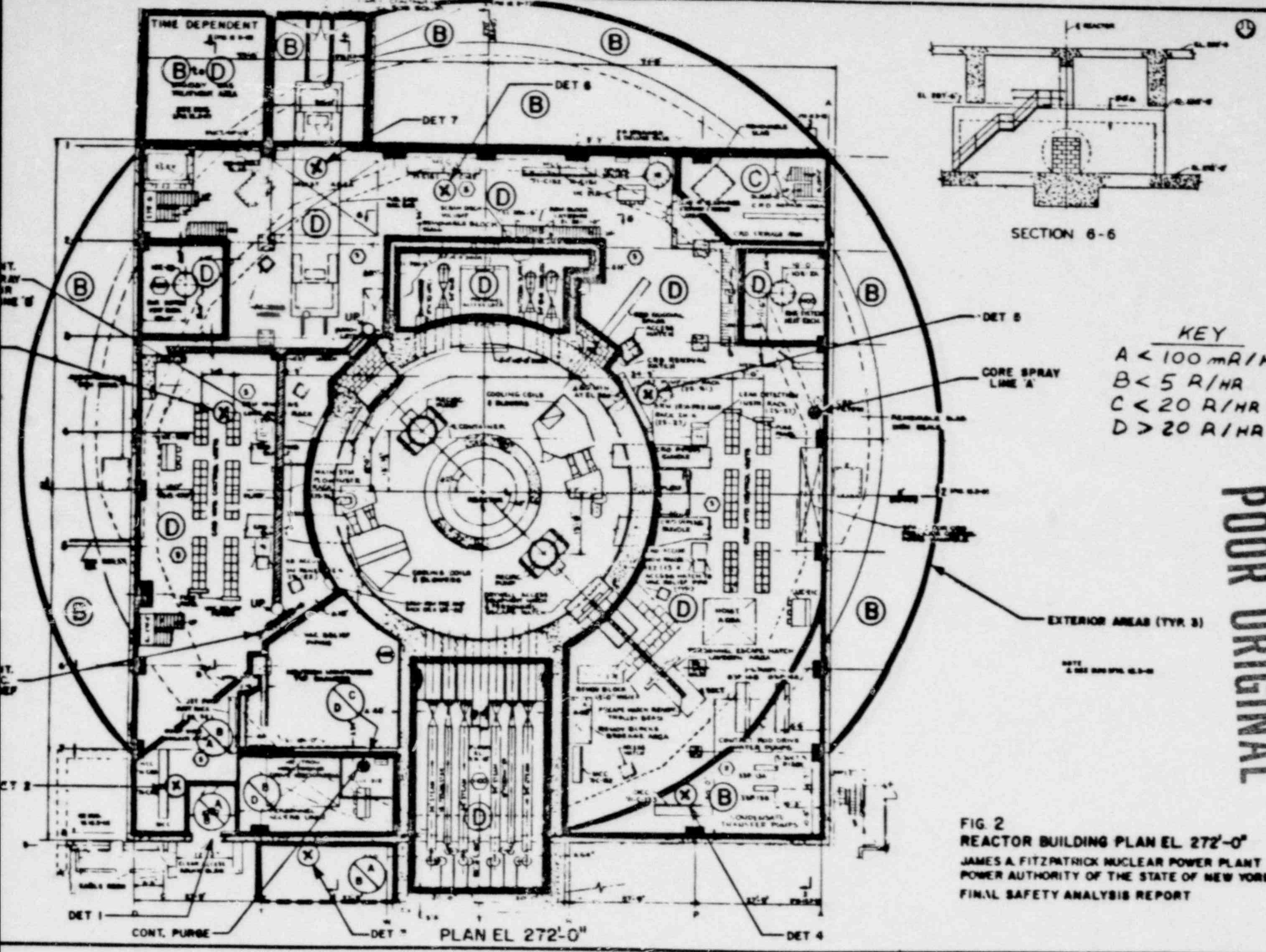
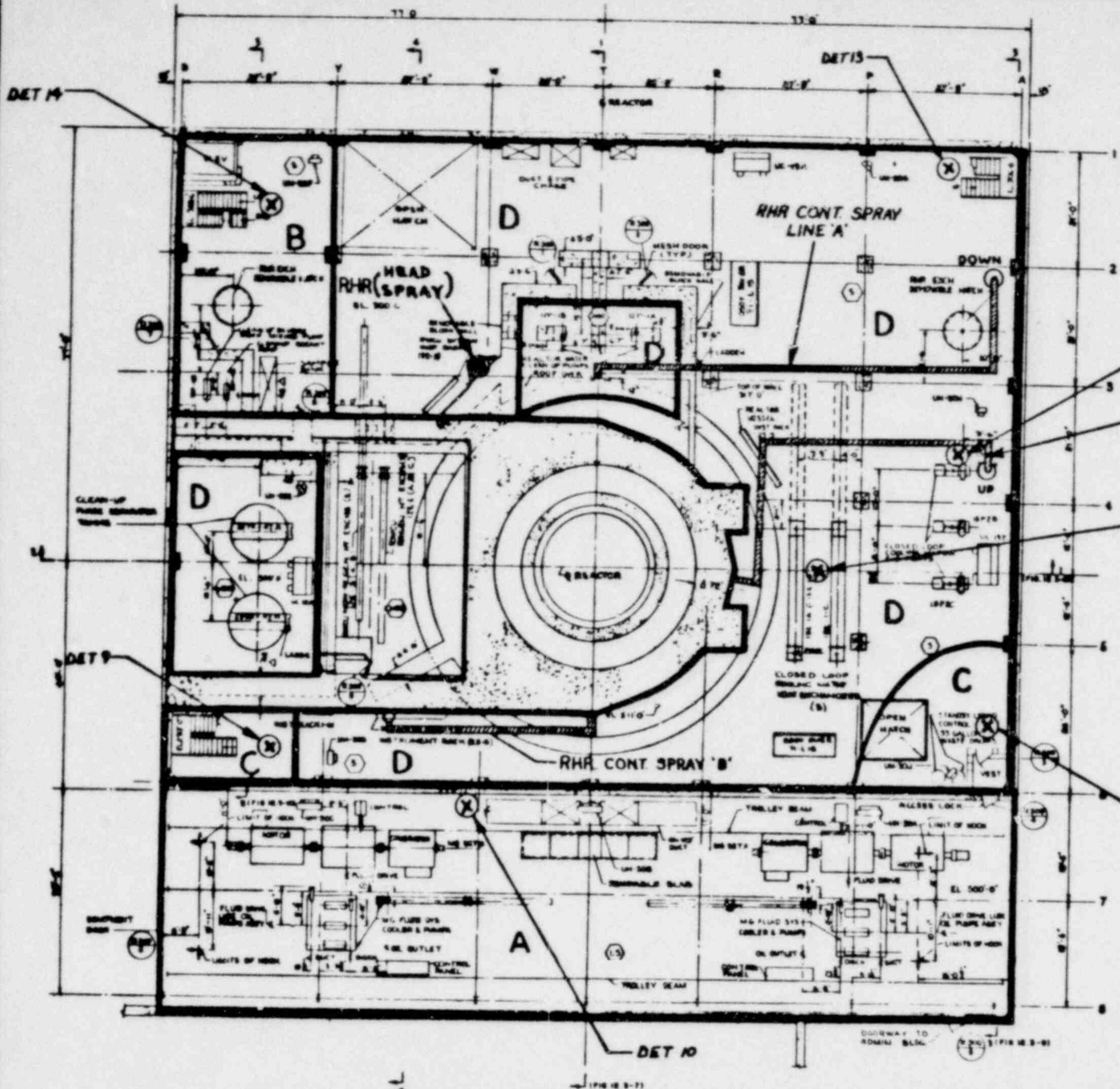


FIG. 2
REACTOR BUILDING PLAN EL. 272'-0"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK
FINAL SAFETY ANALYSIS REPORT



PLAN EL. 300' 0".

KEY

A < 100 mR/HR
B < 5 R/HR
C < 20 R/HR
D > 20 R/HR

FIG. 3
REACTOR BUILDING PLAN-EL. 300'-0"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

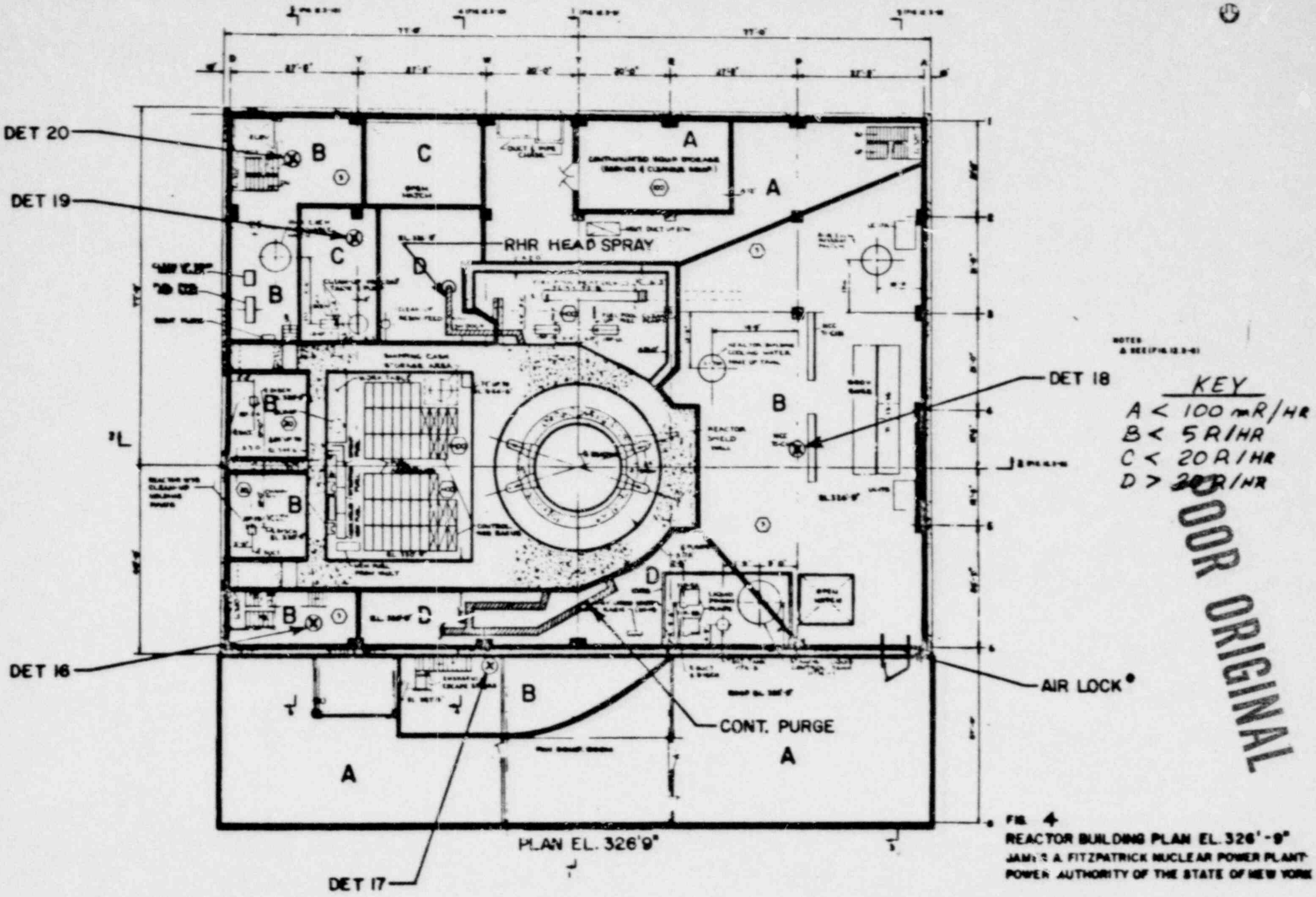
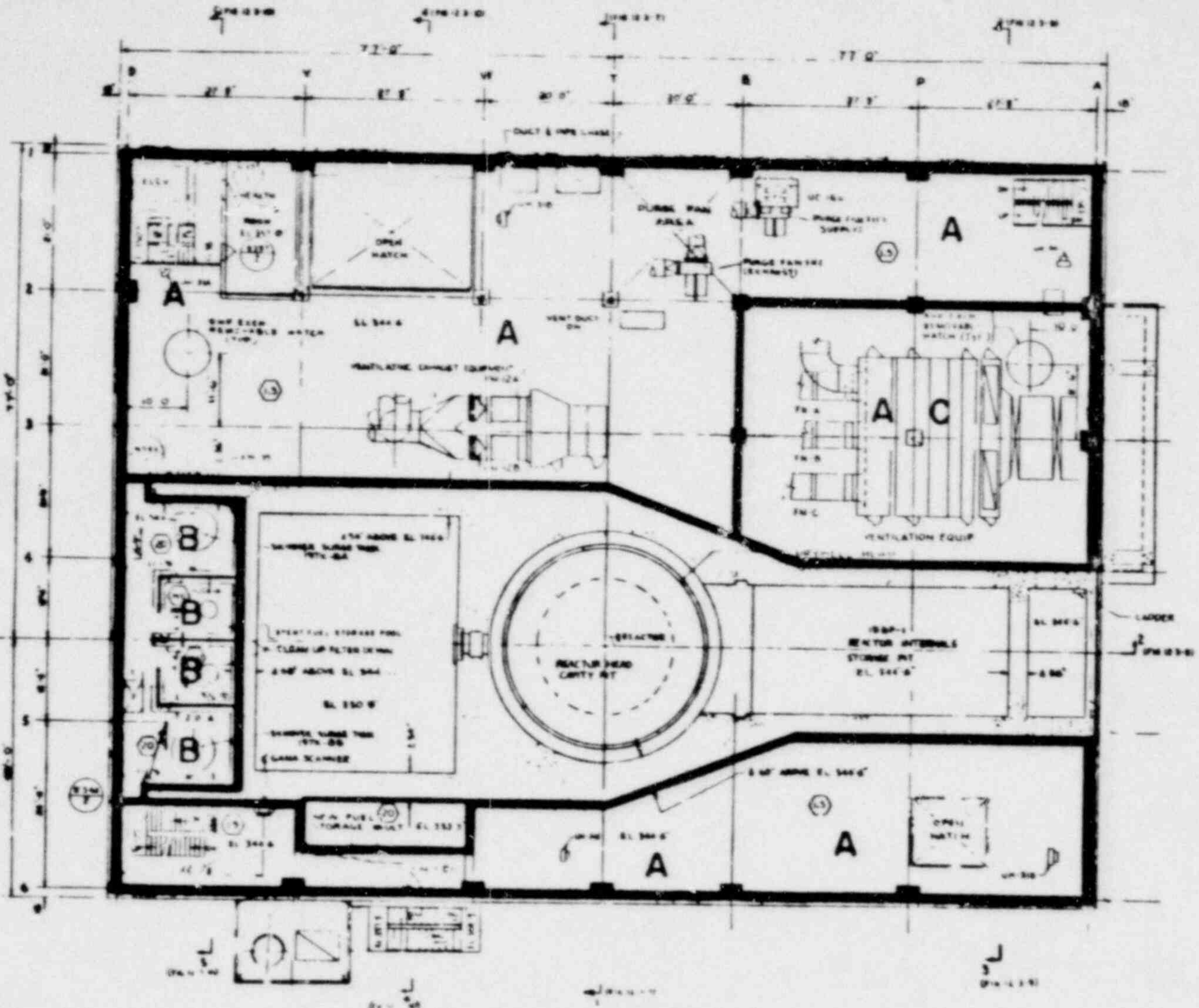


FIG. 4
REACTOR BUILDING PLAN EL. 326'-9"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK



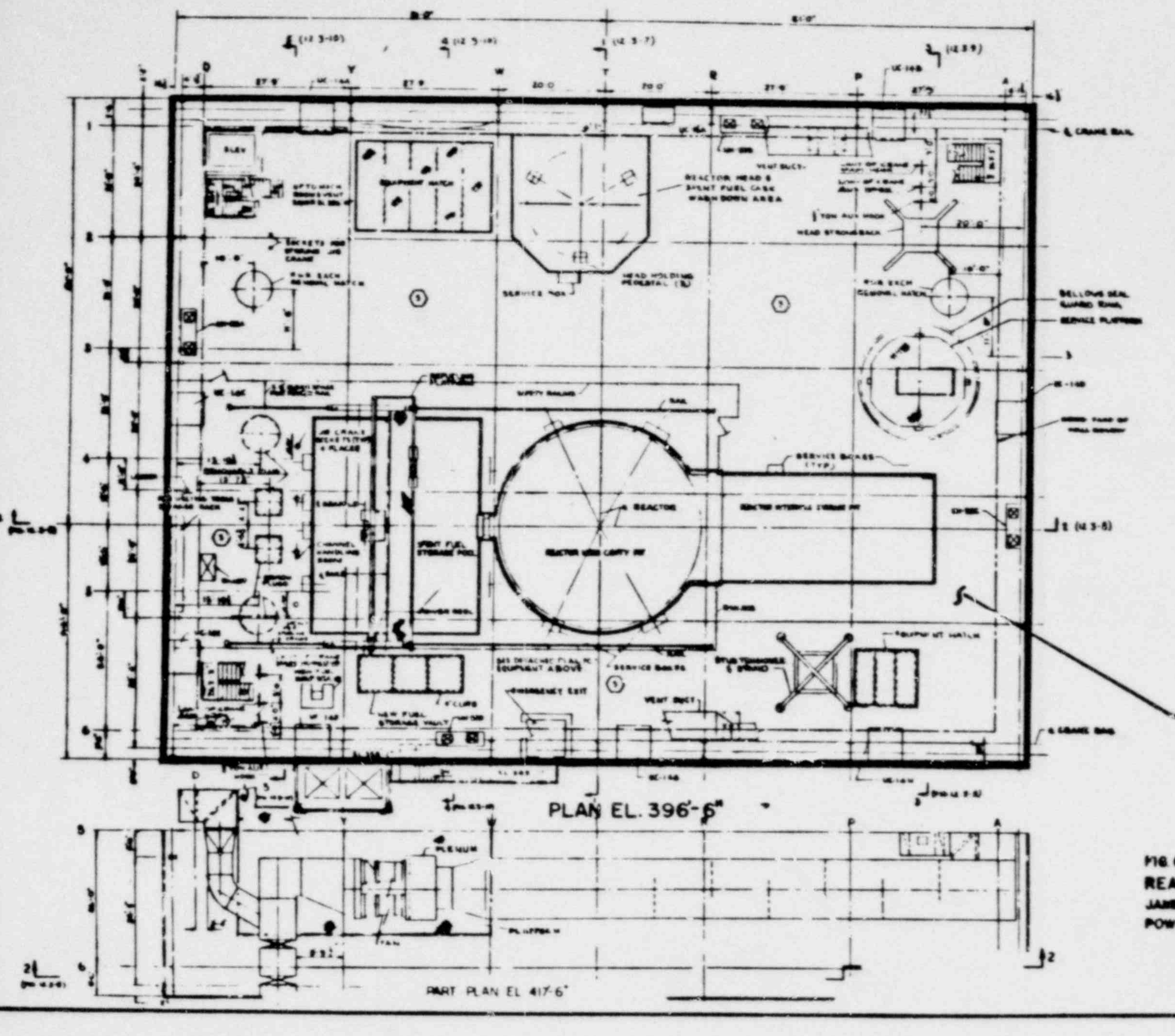
PLAN EL. 344'-6"

KEY

A < 100 mR/HR
B < 5 R/HR
C < 20 R/HR
D > 20 R/HR

POOR ORIGINAL

FIG 5
REACTOR BUILDING PLAN EL. 344'-6"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK



—TOTAL ELEVATION 'A'

KEY
A < 100 mR/HR
B < 5 R/HR
C < 20 R/HR
D > 20 R/HR

POOR ORIGINAL

FIG. 6
REACTOR BLDG PLAN EL. 369'-6"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

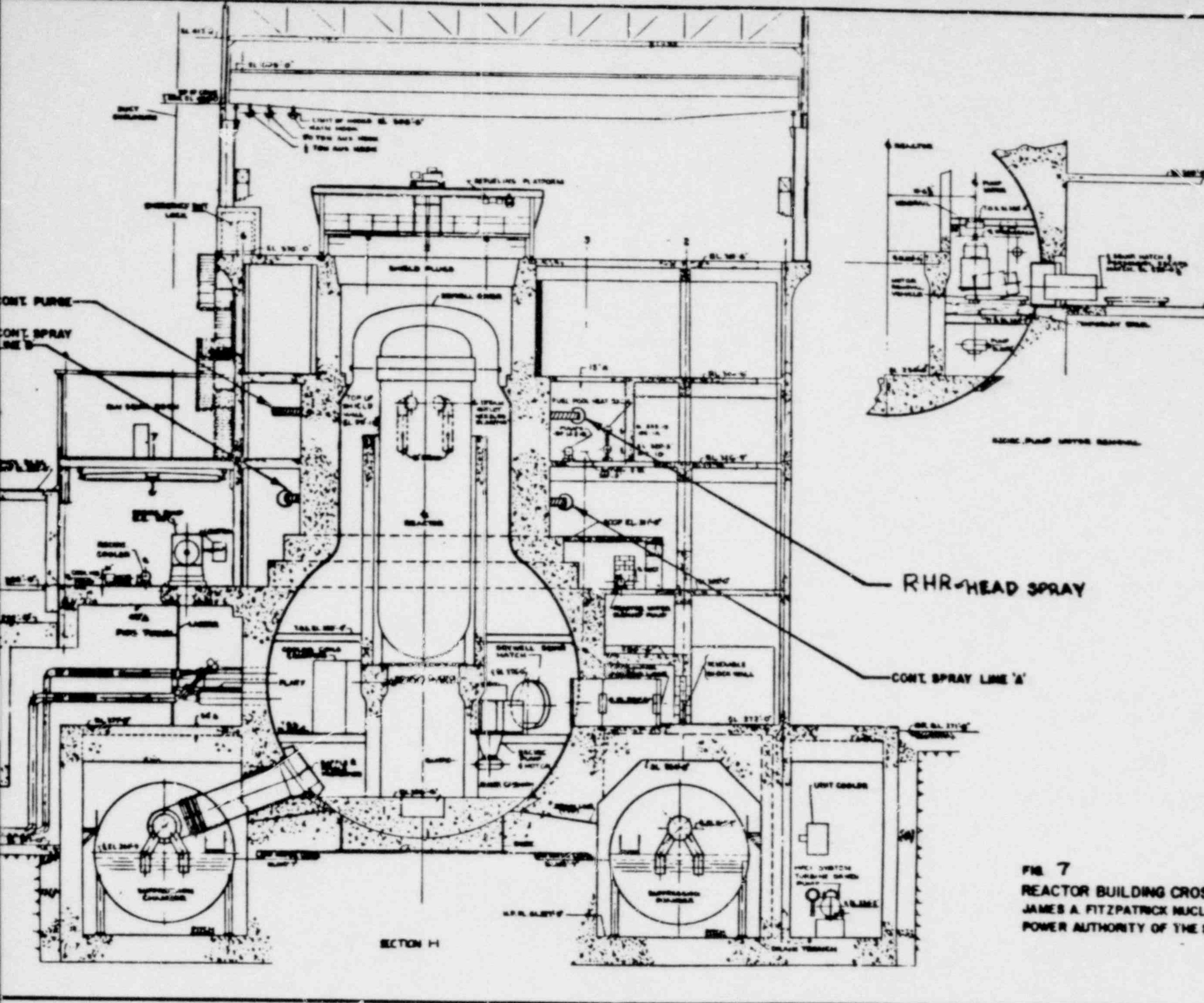
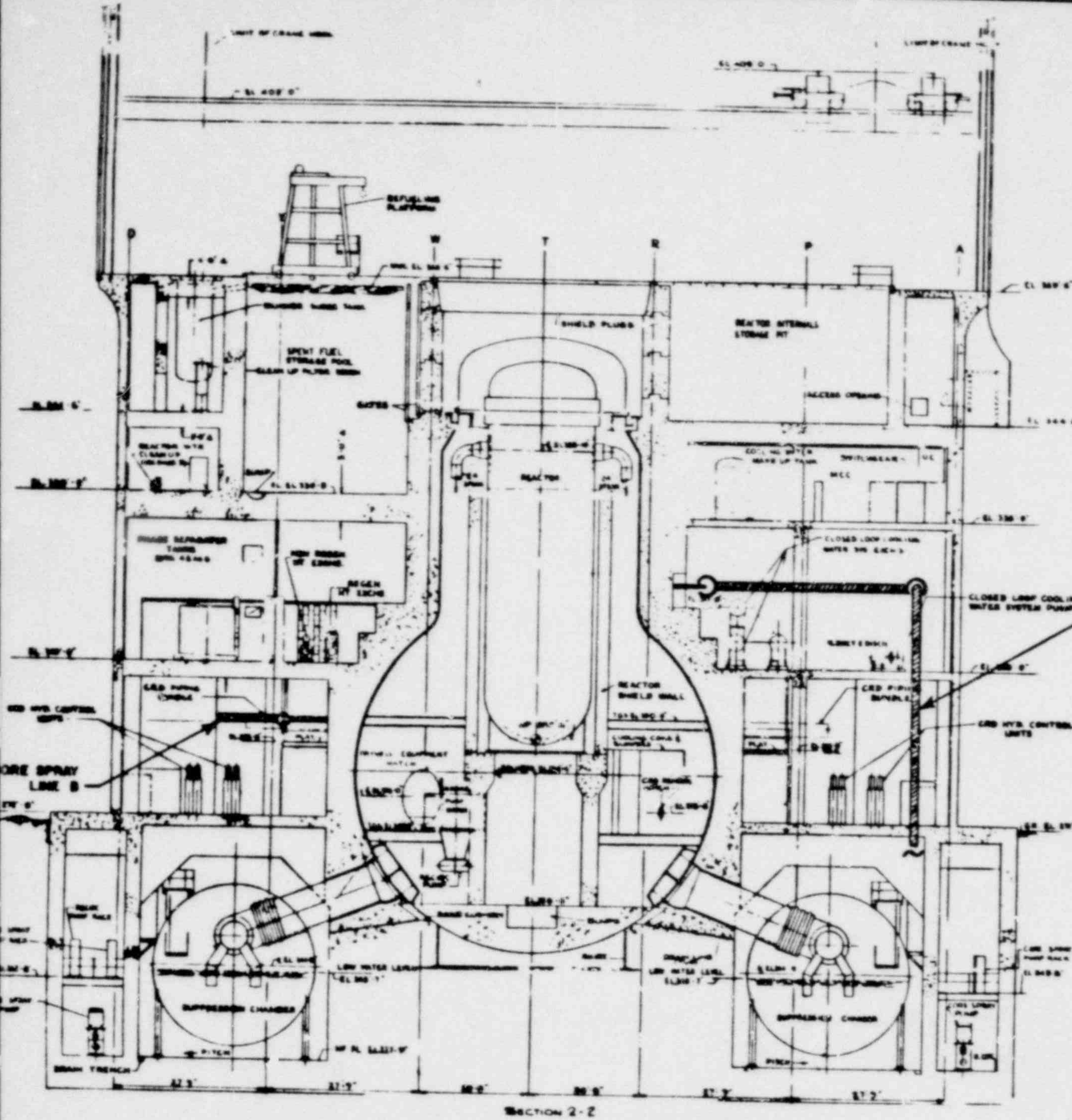


FIG. 7
REACTOR BUILDING CROSS SECT. I-I
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK



POOR ORIGINAL

FIG 8
REACTOR BUILDING CROSS SECT. 2-2
JAMES A FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

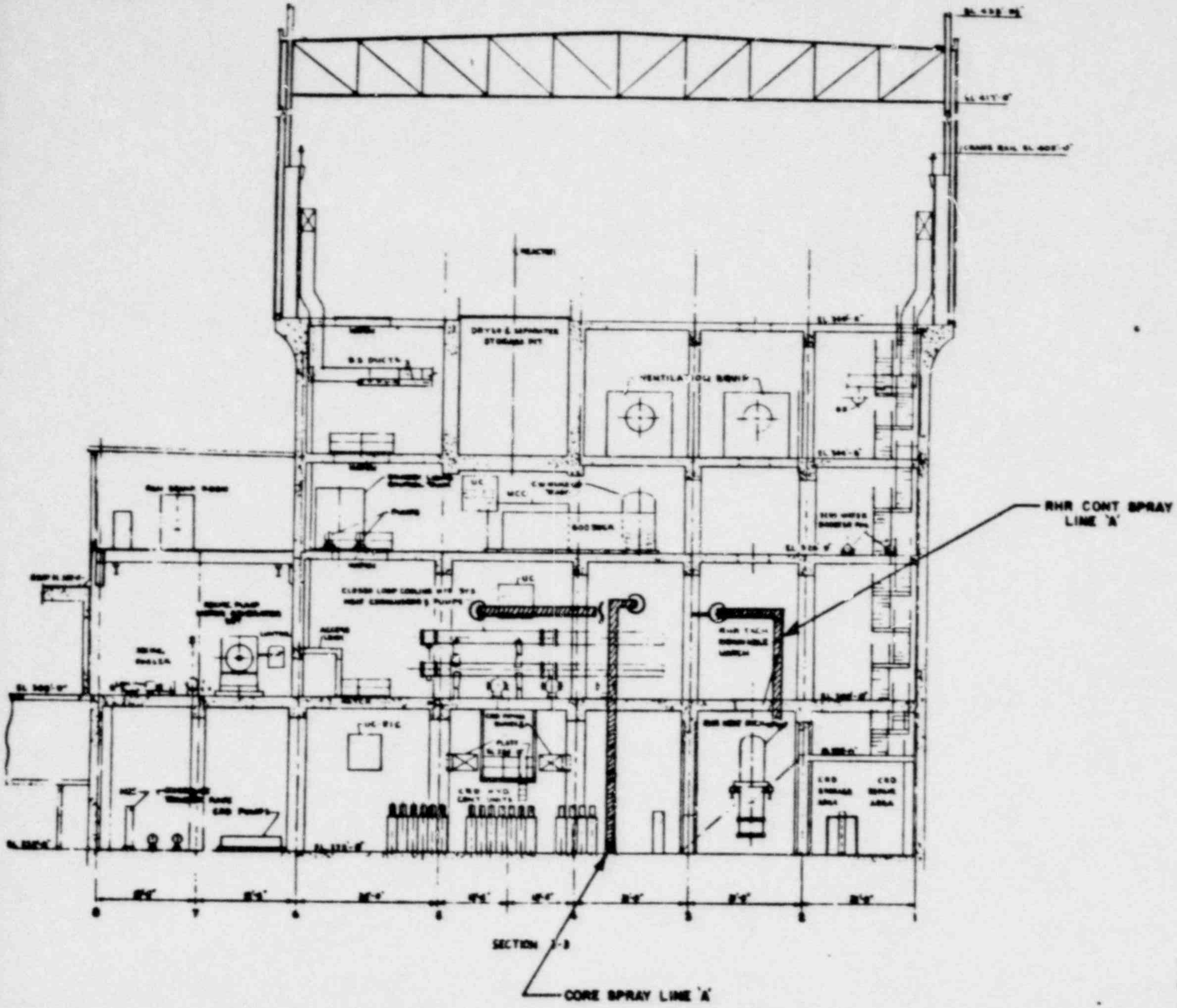


FIG. 9
REACTOR BUILDING CROSS SECTION 3-3
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

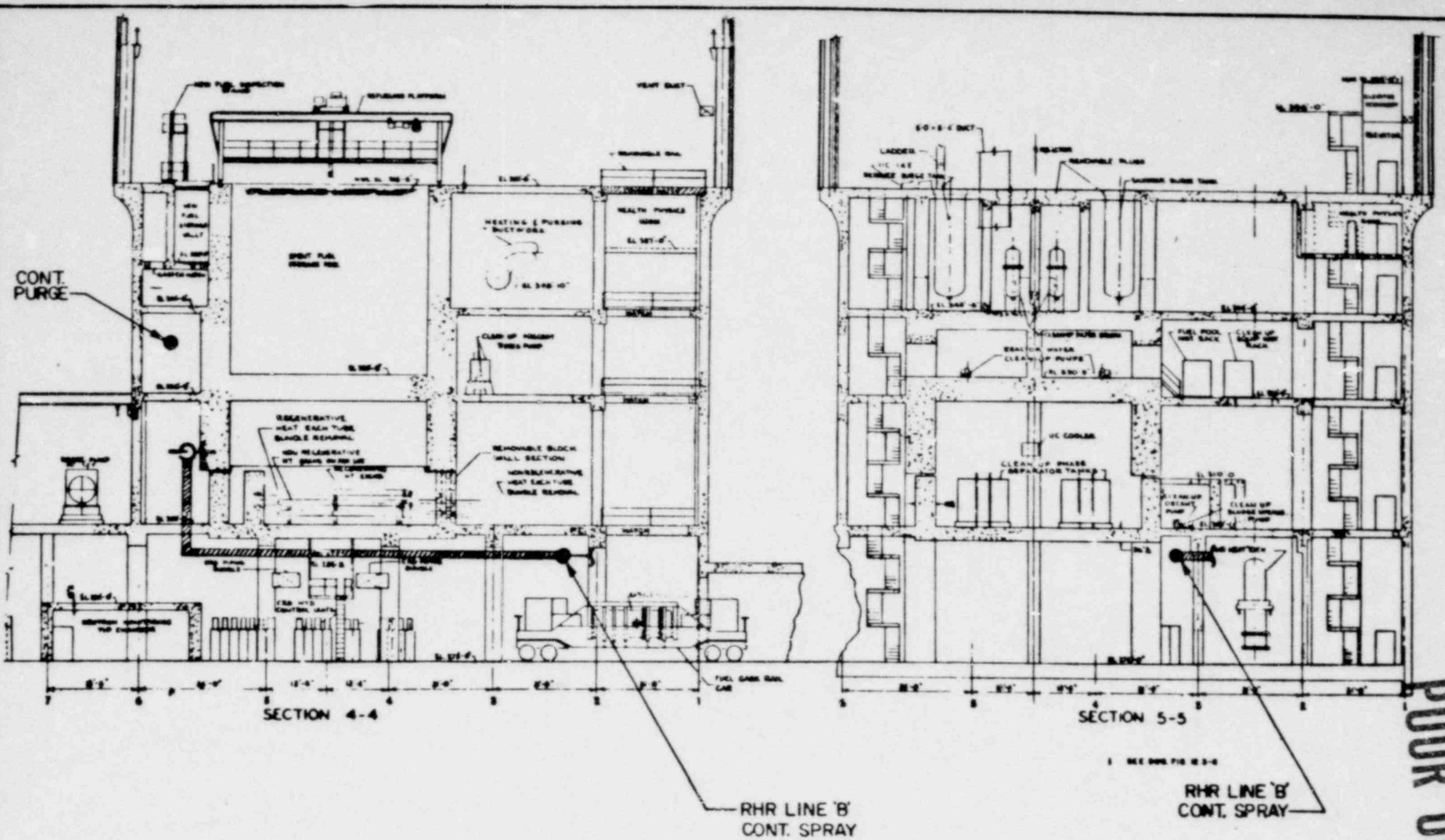
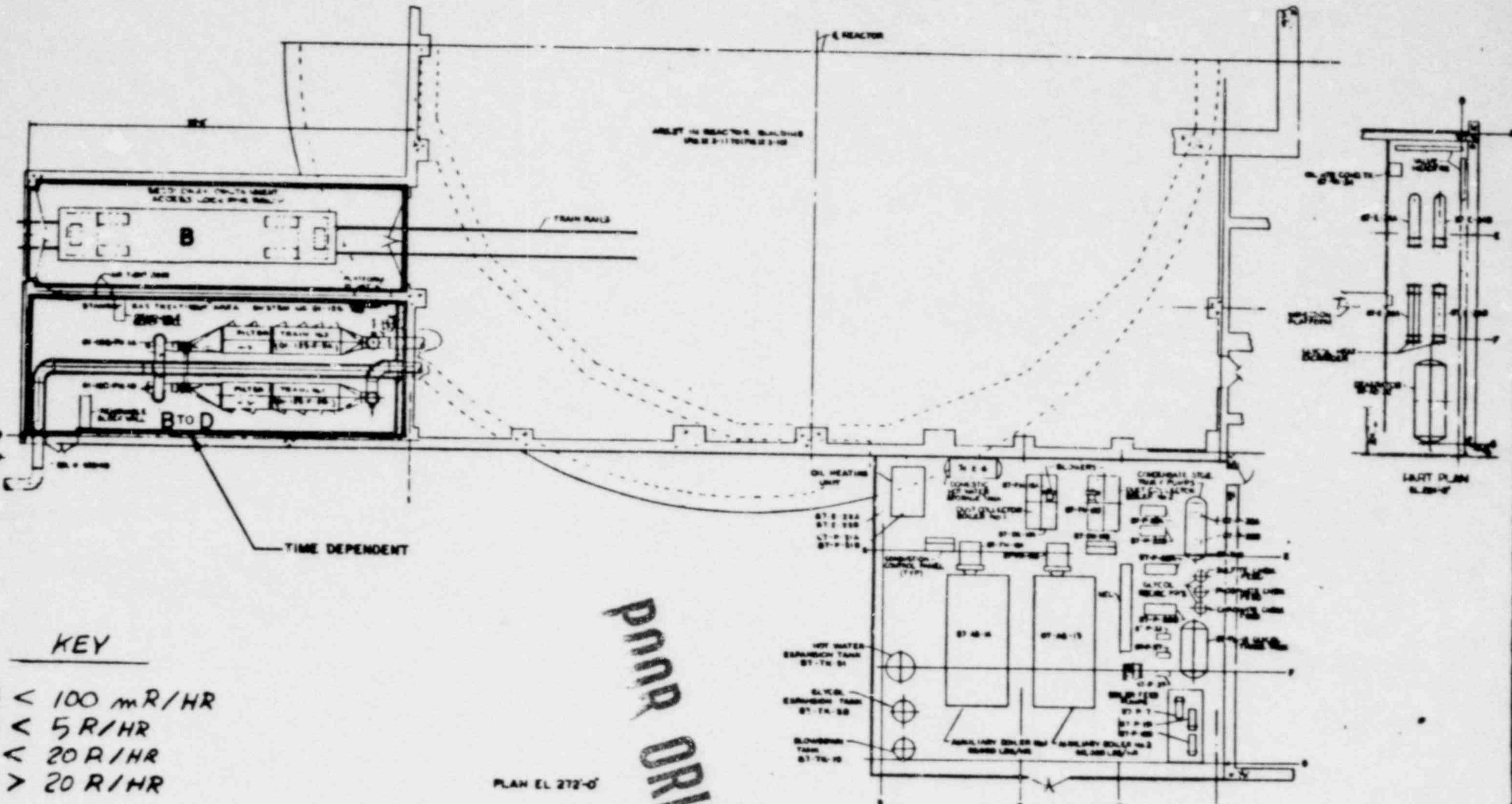


FIG 10
REACTOR BUILDING
CROSS SECTIONS 4-4 & 5-5
JAMES A FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

POOR ORIGINAL

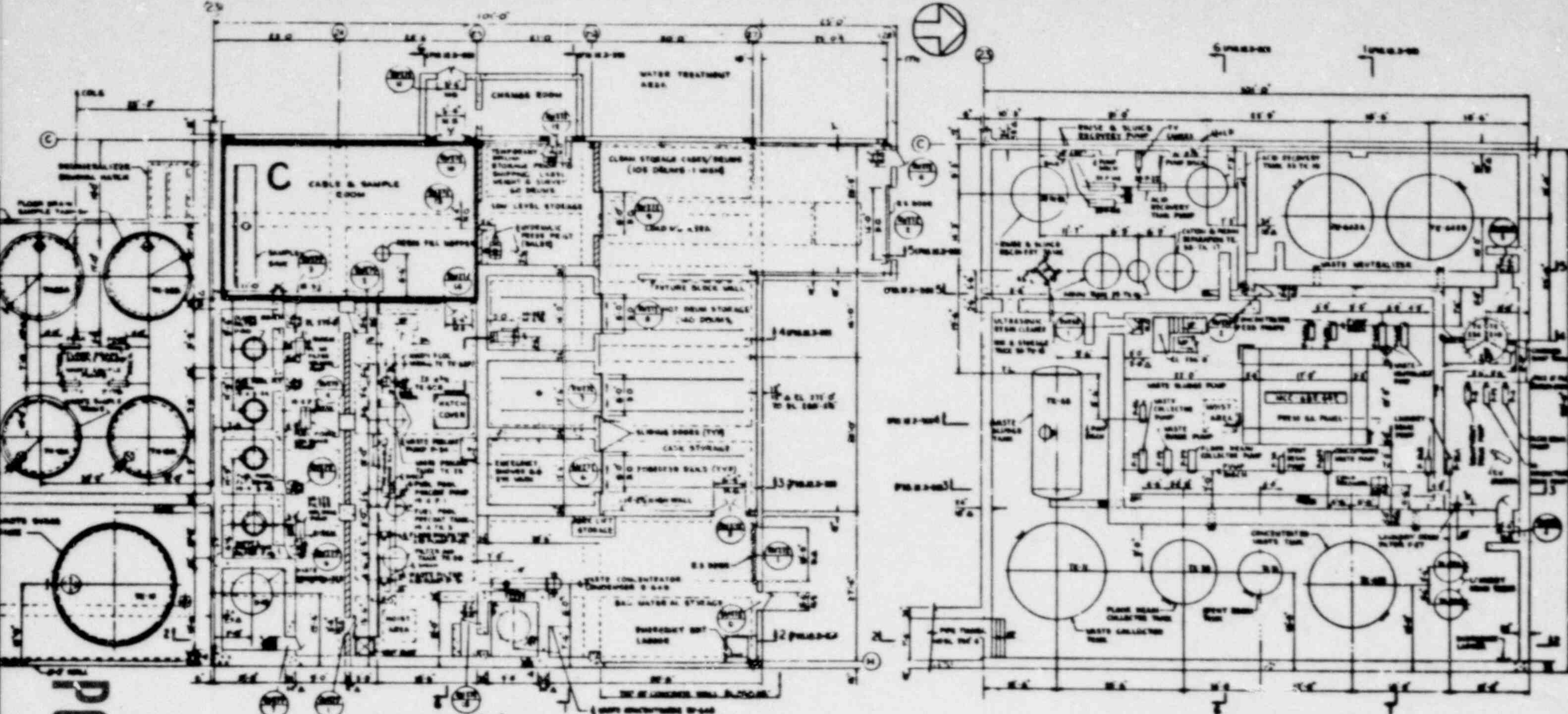


KEY

A < 100 mR/HR
B < 5 R/HR
C < 20 R/HR
D > 20 R/HR

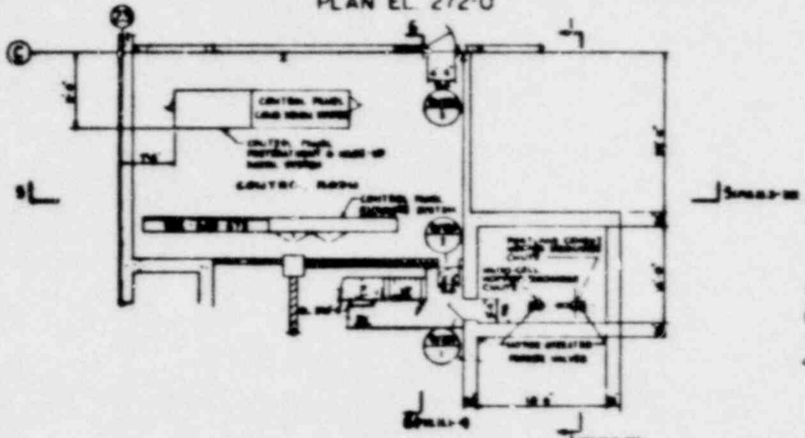
FIG. 11
AUXILIARY BOILER BUILDING
GAS TREATMENT BLDG. PLAN

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK



PLAN EL. 272-0"

PI.AN EL. 250'-0"



PART PLAN EL 284-0"

POOR ORIGINAL

KEY

A < 100 mR/HR
B < 5 R/HR
C < 20 R/HR
D > 20 R/HR

FIG. 12
RADIOACTIVE WASTE BUILDING, PLANS
EL. 250'-0", EL. 272'-0", EL. 284'-0"
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
POWER AUTHORITY OF THE STATE OF NEW YORK

REFERENCES

1) FSAR, J.A. FitzPatrick Nuclear Power Plant.

2) NUREG-0578, USNRC, dated 7/79.

3) NUREG-0737, USNRC, dated 11/80.

4) G.E. Specification No. 22A2703T, "Radiation Sources for BWR Requirements", Revision 1, dated 3/1/73.

5) QAD-QC, "3-Dimensionel Point Kernal Gamma Shielding Code", RP-100, by Richard E. Deem, dated 7/77. RSIC code Package ccc-401/QAD-QC.

6) ACTGEN, "Radioactivity Source Generation Code", RP-106, by Richard E. Deem, dated 8/77.

7) G.E. "Post Accident Sample Station Activity Source Terms", by H.R. Helmholz, dated 9/80.

8) SWEC Radiation Protection Code DRAGON3, "Dose and Radioactivity from Nuclear Facility Gascous Outflows", NU-115, Version 00, Level 01, dated 6/4/76.

9) "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations", SWEC, 5/80.

APPENDIX A

POWER AUTHORITY OF THE STATE OF NEW YORK

PROGRAM SUMMARY

TITLE	AUTHOR(S) / SPONSOR(S)	REF. No.	
ACTGEN - Radioactivity Source Generation Code	Richard D. Deem	RP-106	
PROGRAM STATUS	**USER MANUAL STATUS**	**ISSUE DATE**	**REV. DATE**
Complete	Complete	8-77	

PROGRAM PURPOSE, SCOPE, METHOD, INPUT DATA, OUTPUT RESULT

PURPOSE: This code provides for converting an activity concentration (i.e. μCi/cc) for radionuclides to a multigroup energy spectrum (i.e. MeV/sec-cc). This energy spectrum can then be used as a source term for shielding calculations or other dose rate/heating rate calculations.

SCOPE: This program is limited only by those constraints described in the users manual.

METHOD: Given initial source activities in a control volume at some time T_o, this program calculates the residual activities at the end of one, or several, decay periods according to the appropriate decay equation (first, second, and third order isotopic decay are considered). The effect of a continuous recirculation-filter on the system can also be determined. A library of more than 100 isotopes is provided with the ability, when desired, to modify any data within the library or add other isotopes of interest. Also calculated is the gamma ray energy released, due to radioactive decay, in seven energy groups.

INPUT: Decay times initial concentrations of the source isotopes, and any alterations/additions to the existing library information.

OUTPUT: Library data used, decay time, purification code, initial inventory, alterations/additions to the library data, gamma energy released in each of the seven energy groups, and various sums of gamma energy released for each isotope group and for all contributors.

PROGRAM NAME	SOURCE ORGAN	ORGAN. REF. No.
IBM 370 - ACTGEN, CDC6600 - ACTGEN	PASNY	

APPENDIX B

POWER AUTHORITY OF THE STATE OF NEW YORK

PROGRAM SUMMARY

TITLE QADQC - 3 Dimensional Point Kernal Gamma Shielding Code	AUTHOR(S) / SPONSOR(S) Richard E. Deem	REF. No. RP - 100	
PROGRAM STATUS Complete	USER MANUAL STATUS Complete	ISSUE DATE 7/77	REV. DATE

PROGRAM PURPOSE, SCOPE, METHOD, INPUT DATA, OUTPUT RESULT

PURPOSE: This program calculates the direct beam gamma dose rates at points in 3-dimensional space from point, volumetric, and cosine intensity function sources. The source and dose points can be described in either cartesian, cylindrical, or spherical coordinates, while the geometry description is limited to cartesian only. *

SCOPE: This program is limited only by the contraints specified in the users manual.

METHOD: Numerical integration of point sources in 3-dimensional geometry, Volumetric sources are divided up into a point source distribution then integrated numerically. The cosine intensity function sources are numerically integrated directly. The line-of-sight distance from each point source to the dose point is then calculated. Based on the distance traveled through each material region specified and the shielding characteristics of each material, the uncollided gamma ray flux at the dose point is calculated. A buildup factor is then applied to this dose to take into account the scattering component of the dose rate.

INPUT: Specific sources, geometric model, materials data, buildup factors, and flux to dose conversion factors.

OUTPUT: Dose rates with and without buildup in any units desired by the user.

*QADQC is similar in all respects to the QAD-P5 nuclear code RSIC Document (CCC-048/QAD) except that all neutron moment and heating calculations have been eliminated, the number of regions and boundaries has been reduced, and various changes to the output formats have been changed. This code was developed for quick and fairly inexpensive direct beam gamma dose calculations.

PROGRAM NAME IBM 370 - QADQC ; CDC6600 - QADF	SOURCE ORGAN. PASNY	ORGAN. REF. No.

START DATE	FINISH DATE	COMPUTER	LANGUAGE	MACHINE REQUIREMENTS