~~PDR~~ 70-2937

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TWO NORTH NINTH STREET, ALLENTOWN, PA. 18101 PHONE (215) 821-5151

December 23, 1980

Director
Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
SPECIAL NUCLEAR MATERIALS LICENSE APPLICATION
NEW FUEL STORAGE AND RECEIPT
ER 100450
PLA-594

FILE 841-06

Docket #50-387
#70-29-37

This application is filed pursuant to Title 10 Code of Federal Regulations, Part 70 for authorization to receive, possess, store, inspect, and package for transport nuclear fuel bundle/assemblies.* This application superceeds that which was filed on October 16, 1980 and has been revised to address concerns expressed by your Norm Ketzlach. It is requested that the Special Nuclear Materials License remain in effect until receipt of the Operating License.

This application is excempt from licensing fees per 10CFR 170.11(3).

Any questions concerning this application should be forwarded to Mr. Thomas E. Gangloff, (215)-821-5543 of the Nuclear Licensing Group.

The following information is submitted in support of the application.

1.0 APPLICANT

Pennsylvania Power & Light Company
P.O. Box 1870
Allentown, PA. 18105

2.0 ADDRESS OF STORAGE SITE

Unit 1 of the Susquehanna Steam Electric Station (SSES) is located in Salem Township, Luzerne County, in east central Pennsylvania, about five miles northeast of Berwick, Pennsylvania.

*bundle - that which is received from the fuel manufacturer
assembly - fuel bundle with channel affixed.

3.0 CORPORATE INFORMATION

The information set forth in the application for Construction Permit and Operating License, Docket No. 50-387 dated July 20, 1978 for the Susquehanna Steam Electric Station Unit 1, is hereby incorporated by reference. Information concerning control and ownership of the applicant is also set forth in the application for Construction Permit and Operating License.

4.0 RADIOACTIVE MATERIAL

Initial Core Fuel Assemblies

Maximum square dimension of fuel bundle/assembly	5.47 in.		
Active fuel length	150 in.		
Overall fuel bundle/assembly length	176.16 in.		
Maximum square dimension of fuel channel	5.45 in.		
Overall channel length	166.91 in.		
Channel wall thickness	.080 in.		
Rod array type	8 x 8 = 64 rods 62 fuel rods 2 water rods		
Fuel rod Pitch	0.640 in.		
Fuel rod clad thickness	0.032 in.		
Rod clad material	Zircaloy		
Fuel rod outside diameter	.483 in.		
Fuel pellet diameter	.483 in.		
Fuel pellet material	UO_2		
Nominal fuel stack density	10.32 gm/cm^3		
Total number of fuel assemblies	764 (3 types)		
Maximum U-235 enrichment (weight %)	3.00		
Average U-235 enrichment (weight %)	1.88		
Average U-235 enrichment (weight %) per assembly type	.711	1.76	2.19
Number of assemblies per type	92	240	432

UO_2 weight per assembly (kg)	207.58	207.31	207.00
Gd 0 weight per assembly (kg)	0.0	0.20	0.44
Average UO_2 weight per fuel rod (kg)		3.34	
U weight per assembly (kg)	182.96	182.73	182.45
(lb)	403.36	402.85	402.23

The composition and location of each type fuel rod in the assembly having the highest enrichment can be found in NEDC-21307, February 1977, entitled "BWR5 Standard Plant Fuel Design Summary" (pgs 15, 17 & 18; figs 6 & 8). These figures are included as proprietary figures 4.3-3 and 4.3-5 of the Susquehanna SES FSAR. Other fuel bundle types are described in NEDE 20944P.

The maximum amount of special nuclear material, as kilograms of U-235, which may be possessed at any one time is 3,043 kilograms. This represents the initial core and fifty additional assemblies per enrichment type.

5.0 TRANSPORTATION

All initial core fuel bundles are to be delivered to the site in accordance with shipping procedures and arrangements of the General Electric Company. The fuel bundles will be shipped by the fuel manufacturer as authorized under NRC Certificate of compliance number USA/4986/B(F) Revision 8. Fuel will normally be shipped to the site by truck, with a maximum of 16 containers per truck and two bundles per container.

6.0 STORAGE CONDITIONS

The fuel bundle/assemblies will be stored on the refueling floor of the Reactor Building (see SSES Final Safety Analysis Report, Section 9.1) in either the new fuel storage vault dry, in the spent fuel pool, or in their metal shipping containers on the refueling floor. When the fuel bundle/assemblies are stored in the spent fuel pool, the pool may be dry, partially flooded, and/or completely flooded during residence in the storage rack. The refueling floor is surrounded by structural steel and metal siding and, therefore, it is considered unlikely that a fire will occur. However, the use of flammable materials on the refueling floor will be carefully controlled as an added margin of safety. In addition, the movement of the new fuel vault cover will be administratively controlled by approved plant procedures. The refueling floor is shown on SSES Final Safety Analysis Report (FSAR) Figures 1.2-23, 1.2-24, 1.2-25, and 9.1-16B.

Preoperational testing, to the extent necessary to prove that the fuel handling equipment is functioning properly and is safe to handle the fuel over the fuel receipt, inspection and storage areas, will be performed prior to using this equipment to handle fuel. Likewise, the necessary support systems will be tested to the extent necessary to prove that they are capable of performing their specific functions for fuel receipt and inspection activities prior to the arrival of the first fuel shipment. Functional checks of the fuel handling equipment will also be done in accordance with approved plant operating and surveillance procedures.

Some construction and preoperational testing activities may be occurring on the refueling floor (e.g. reactor well work, communication cable pulling, conduit installation) while fuel is being received, inspected, and/or stored. However, due to the security provisions that will be in effect, which include the separation of the two types of activities, there will be no impact on the safe receipt, inspection or storage of the fuel. In addition, the responsible supervisor on shift during fuel receipt and inspection activities will have the authority to terminate either or both of fuel receipt and inspection activities or the construction and preoperational testing activities.

6.1 Inspection and Handling

As soon as practicable after the fuel carrier vehicle arrives at the plant gate, it will be taken to a predetermined off load area where the shipping containers will be unloaded from the carrier vehicle. The wooden containers will then be taken to the unloading bay on the ground floor of the Reactor Building (EL672). The metal inner shipping containers will be removed from the wooden outer shipping containers. The metal containers, holding the fuel bundles, will be lifted to the refueling floor of the Reactor Building, where they can be stored for an interim period of time. The metal containers will be placed in a near vertical position, and the bundles are then removed from the containers. The fuel bundles may be placed in the new fuel inspection stand to be checked for integrity and numbering, or placed in an appropriate storage location for future inspection.

Upon removal of the fuel bundle from the shipping containers, the plastic bag will be removed prior to inserting the fuel bundle into the storage rack. There is no way water can be entrapped within the open lattice structure of the fuel bundle. No credible mechanism exists with which to block the 3-inch diameter nozzle on the bottom tie plate of the fuel assembly.

All fuel receipt and inspection activities will be controlled by the applicable procedures which are reviewed and approved by the Susquehanna SES Plant Operations Review Committee (See Section 13.4 of the Susquehanna SES FSAR), and approved by the Superintendent of the plant. The ultimate responsibility for administrative controls over fuel receipt and inspection activities lies with the Superintendent of Plant.

In the event of damaged or unacceptable fuel which must be returned to the manufacturer, packaging will be in accordance with DOT and 10CFR71 requirements and shipments will be made via a licensed carrier.

6.2 Safety Evaluation

The new fuel bundle/assemblies can be stored in the new fuel storage racks in the new fuel storage vault, in high density spent fuel storage racks in the spent fuel pool and/or in their metal shipping containers. No more than three (3) fuel bundle/assemblies will be outside the shipping containers or storage racks at one time. The metal shipping containers are described, and their storage without a criticality warning system authorized in Amendment #5 to GE's Special Nuclear Materials License #SNM-1097 dated June 6, 1978. When fuel bundles are stored in their metal shipping containers, the containers will not be stacked more than four (4) high. The cross-sectional size of the arrays will only be limited by space available on the refueling floor.

Metal shipping containers containing fuel will not be stored closer than five (5) feet to any fuel handling equipment. No interaction will occur between fuel stored in the shipping containers and that stored in the new fuel storage vault as the vault is embeded in a concrete floor with a minimum of one (1) ft. of concrete between storage arrays.

Susquehanna SES Plant Staff Reactor Engineering personnel have the overall responsibility for nuclear criticality safety. These people are degreed physicists or nuclear engineers and have had training in BWR systems and fuel. In addition they will be certified as fuel inspectors for Susquehanna SES and will receive all the classroom training given to fuel handling or inspection personnel.

The criticality control and fuel rack design criteria for the new fuel storage vault and the spent fuel storage pool is outlined as follows.

6.2.1 Criticality Control - New Fuel Storage Vault Rack

The new fuel storage vault is designed to maintain Keff $\leq .95$ (see FSAR Subsection 9.1.1.1.1.2) in dry conditions and when flooded with water between 0.7 and 1.0 g/cc. The new fuel vault is covered by leak proof, metal, removable covers which are controlled by administrative procedures which prevent optimum moderation conditions no more than two (2) vault covers will be off at a time.

The new fuel vault covers (FSAR Figure 9.1-1a-Attachment 1) overlap the storage vault curb and have a protective lip that prevents direct impingment upon the sealing point. The modified I-beams that span the vault also are overlapped by the covers. The beams act as drains to direct run-off from the covers to the refueling floor. Sealing is provided by the covers resting upon the channel and beam upper surfaces. The mass of the covers, and the close tolerances of construction, make gaps negligible.

The calculations of keff are based upon the geometrical arrangments of the fuel array and subcriticality does not depend upon the presence of neutron absorbing materials. To meet the requirements of General Design Criterion 62, geometrically-safe configurations of fuel stored in the new fuel array are employed to assure that keff will not exceed 0.95 if fuel is stored in the dry condition or if the abnormal condition of flooding (water with a density of 1 g/cc) occurs. In the dry condition, Keff is maintained <0.95 due to under-moderation. In the flooding and flooded condition, the geometry of the fuel storage array assures the Keff will remain <0.95 due to over-moderation.

No limitation is placed on the size of the new fuel storage vault rack array from a criticality standpoint since all calculations are performed on the basis of an infinite array of assemblies. The New Fuel Storage Vault will, therefore, accommodate fuel from a multi-unit facility with no safety implications. All handling conditions remain the same and there is no compromise of safety considerations.

6.2.2 New Fuel Storage Vault Rack Design

a) The New Fuel Storage Vault contains 23 sets of castings which may contain up to 10 fuel bundle/assemblies; therefore a maximum of 230 fuel bundle/assemblies may be stored in the fuel vault.

b) There are three tiers of castings which are positioned to fixed box beams. This holds the fuel bundle/assemblies in a vertical position and supports the fuel at the lower and upper tie plate with additional lateral support near the center of gravity of the fuel assembly.

c) The lower casting supports the weight of the fuel bundle/assembly and restricts the lateral movement; the center and top casting restricts lateral movement only of the fuel bundle/assembly.

d) The New Fuel Storage Vault Racks are made from aluminum. Materials used for construction are specified in accordance with ASTM-1978 specifications. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the susceptibility of metals to galvanic corrosion, aluminum and stainless steel are relatively close together insofar as their coupled potential is concerned. The use of stainless steel fasteners in aluminum to avoid detrimental galvanic corrosion is a recommended practice and has been used successfully for many years by the aluminum industry.

e) The minimum center-to-center spacing for the fuel bundle/assembly between rows is 11.875 inches. The minimum center-to-center spacing within the rows is 6.535 inches. Fuel bundle/assembly placement between rows is not possible.

f) Lead-in and lead-out of the casting provides guidance of the fuel bundle/assembly during insertion or withdrawal.

g) The rack is designed to withstand the impact force of 4000 ftlbs while maintaining the safe design basis. This impact force could be generated by the vertical free fall of a fuel assembly from the height of 5.3 feet.

h) The rack is designed to withstand the pull-up force of 4000 lbs. and a horizontal force of 1000 lbs. The racks are designed with lead outs to prevent sticking. However, in the event of a stuck fuel bundle/assembly, the lifting bail will yield at a pull up force less than 1500 lbs.

i) The rack is designed to withstand horizontal combined loads up to 222,000 lbs, well in excess of expected loads.

j) The maximum stress in the fully loaded rack in a faulted condition is 25.9 Kip. This is significantly lower than the allowable stress.

k) The rack is designed to handle nonirradiated, low emission radioactive fuel assemblies.

6.2.3 Criticality Control - Storage of New Fuel in High Density Spent Fuel Racks

The design of the spent fuel pool storage racks assure that Keff will remain <0.95 for both normal and abnormal storage conditions. Normal conditions exist when the fuel is dry, flooding and flooded. An abnormal condition may result from accidental dropping of equipment or damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

To ensure that the General Design Criteria 62 is met, the following normal and abnormal spent fuel storage conditions were analyzed.

a) normal positioning in the spent fuel storage array,

b) fuel stored in control rod racks,

c) Pool water temperature increases to 212°F,

d) normal storage array of ruptured fuel,

e) abnormal positioning in the spent fuel storage array,

f) moving fuel bundle between work rack and storage area,

g) dropped fuel bundle adjacent rack

The criticality safety analysis was performed by Nuclear Associates International (NAI 78-75 dated 5-15-79). This analysis shows that under all postulated normal and abnormal conditions, Keff will remain ≤ 0.95.

A discussion of the analysis methodology and results follows:

6.2.3.1 The Principal Analytical Model

The criticality analysis of the Susquehanna spent fuel storage rack design used the CHEETAH-B/CORC-BLADE/PDQ-7 calculational model as the basic engineering tool. CHEETAH-B is the BWR lattice version of NAI's CHEETAH code which is a modified version of the original LEOPARD code and which uses a modified ENDF/B-II cross section library. CORC-BLADE generates the equivalent diffusion theory cross-sections for the control blade. The PDQ-7 program is the well known few-group spatial diffusion theory code. The transport to diffusion theory correction is generated using the Monte Carlo model described in Section 6.2.3.2.

The reference case storage rack cell is shown in Attachment 2. A zero current boundary condition is applied to the four sides of this cell to produce an infinite array effect. The two dimensional PDQ-7 reference case calculations are made for four neutron energy groups, two mesh intervals per fuel pin, a flat U-235 enrichment description and a zero axial buckling to simulate infinite fuel length.

6.2.3.2 <u>The Verification Model</u>

The verification calculation employs the KENO-IV/AMPX model. The basic neutron cross section data comes from the master library of AMPX - a 123 group GAM-THERMOS neutron library prepared from ENDF/B version II data.. The NITAWL module of the AMPX program is used to perform a Nordheim integral treatment of the U-238 resonances accounting for the self-shielding effect. The working library produced by the NITAWL/AMPX module retains the 123 group energy structure and is used directly by KENO-IV.

In the KENO-IV calculation, each fuel and water rod cell is represented discretely, as a box type. The array ption of KENO-IV is applied to arrange the box types into a matrix representing the fuel assembly. Then a water reflection region is added to the outside of this matrix followed by the aluminum canister, void and Boral slab regions to complete the reference case storage rack cell. To simulate the arrangement of a large number of storage rack units, and for a non-leakage condition in the axial direction, a specular reflective condition is applied to all six sides of the reference case storage rack cell (See Attachment 2).

The KENO-IV/AMPX model has been benchmarked against earlier ORNL critical experiments measured by Dr. E. Johnson, a La Crosse BWR cold critical, and the more recent experiment conducted by S.R. Bierman et. al. at the Battelle Pacific Northwest Laboratory (PNL-2438). The latter results show the average calculated Keff to be 1.001$\pm$0.006.

6.2.3.3 Basic Assumptions

The following assumptions were made in the analysis and ensure that the study follows a conservative approach.

1. A uniform 3.25 w/o distribution in an 8 x 8 bundle (which is more reactive than a distributed enrichment with the same overall average weight percent).

2. Fresh fuel, no burnable poison.

3. Minor structural members replaced by water.

4. Fresh water at 68°F.

5. Fuel assemblies are channeled and centered within the storage cavity.

6.2.3.4 Effect on Reactivity of Uncertainties in Material Properties and Rack Geometry and Fuel Placement

(a) Temperature Effect

Using the nominal storage rack cell geometry, the temperature of the fuel and pool water was varied. Intermediate steps at 32°F, 95°F, 120°F, 150°F, 180°F, 212°F were studied. The results of the analysis shows that reactivity decreases continuously as temperature increases from 32°F.

(b) Void Effect

The effect of boiling (assuming equal voids inside and outside of the rack) is studied by varying the voids from 0% to 20% at a temperature of 212°F with the nominal geometry. The results indicate a continuous decrease in reactivity for this range of voids.

(c) Boral Width Reduction

The nominal width of the Boral slab is 5.250". The change in reactivity due to the -0.03 inch minimum tolerance on this width is $\Delta K=0.001$.

(d) Channel Effect

The rack must accommodate both channeled and unchanneled fuel. Studies reveal that the channeled fuel in the rack is more reactive than the unchanneled fuel. Taking the conservative approach, the study here involves channeled fuel except in the accident condition where unchanneled fuel is dropped. The decrease in Δk from channeled to unchanneled fuel is 0.002 in the base case rack (See Attachment 2).

(e) Boral Acceptance Criterion

The criticality calculations described in this report are based on the assumption that the core of the Boral slabs contains a homogeneous mixture of fine B_4C and aluminum powder. An adjustment must be made to take into account directly the self-shielding effect due to the random distribution of B_4C grains of finite sizes in the mixture. Based on a homogeneous mixture, the neutron attenuation factor for a Boral slab of 80 mil thickness with 0.0232 g/cm^2 B-10 loading was calculated to be 0.969. The vendor Brooks and Perkins has proposed the value 0.963 as the acceptance criterion for their measured neutron attenuation factor for the above Boral slabs.

Based on NAI's analysis, the difference between an attenuation factor of 0.969 and 0.963 amounts to about 0.3% Δk for the Susquehanna Spent Fuel Pool. The reactivity reserve of the new pool is more than 1% Δk below the 0.950 limit permitted by NRC; hence Boral slabs meeting the 0.963 criteria will have no adverse effect on the safety of the pool.

(f) Change in Pitch (pitch sensitivity))

The reactivity effect of mechanical tolerance changes caused by structural, fabrication, installation, and seismic conditions can be conservatively determined by the pitch sensitivity study. The calculations were carried out on a 0.125 inch increment. The pitch reactivity coefficient was determined to be about +0.7% Δk/0.125 inch pitch reduction in the range from 6.875 inch pitch to 6.375 inch pitch.

(g) Off-Center Positioning

The reactivity effect of off-center positioning due to improper loading or some natural phenomenon was investigated. The calculations show that the reactivity decreases so no adverse effect occurs.

6.2.3.5 Results of the Criticality Analysis

(a) Diffusion Theory Results

The results of the criticality analysis of the proposed storage rack for Susquehanna Spent Fuel are summarized below:

PDQ Results:

Keff Nominal Case	0.893
Dimensional and Positional Tolerance, ΔK	0.014
Boron Width Effect, ΔK	0.001
Temperature Effect, ΔK	0.004
Variation on Boral Accept.	0.003

Void Effect, ΔK	Negative
Adjusted Keff	0.915

(b) Monte Carlo Results and the Calculational Bias

The KENO-IV verification calculation for the nominal case yields a Keff value of 0.912+0.004 with 95% confidence interval ranging from 0.904 to 0.920. The KENO model has a slightly negative bias (ΔK= -0.001) deduced from NAI's benchmarking calculations. Comparing the upper bound 95% confidence interval result of the KENO nominal case and the PDQ runs, and based on the KENO benchmarking results, a calculational bias of 0.026 in ΔK is applied to the adjusted PDQ Keff to provide the transport to diffusion theory correction.

(c) Summary of Results

Keff (PDQ) adjusted (from (a))	0.915
Transport to diffusion theory correction (from (b))	0.026
Final Keff	0.941
Design Limit, Keff	0.950
Calculational Margin ΔK	0.009

The final Keff value (0.941) includes all the design specification tolerances, model bias, and the 95% confidence interval from the KENO calculations.

6.2.3.6 Abnormal Conditions

(a) Assembly Drop Accident

No adverse reactivity effect is expected from dropping a fuel assembly on top of a fully loaded storage rack during fuel handling because of the large water thickness (∿10 inches) existing between the top of the assemblies already inside the cavities and the dropped assembly resting on top of the rack. Moreover, the basic calculational model assumes an infinite fuel length in the axial direction.

The consequences of dropping an assembly outside the rack and parallel to an assembly located in the outermost row of the storage array were analyzed. The assembly dropped during handling is assumed to lodge parallel to an assembly in an outer cavity with no Boral slab separating the two assemblies. The dimensions of the rack units and the spacing is the same as the refernce case, although the dropped fuel is unchanneled to permit the closest contact with the rack. Due to the reflective boundary conditions used, the array is repeated in three directions. The fourth boundary uses a zero flux condition. The analysis produced an increase in reactivity less than +0.5 ΔK in a 4 x 4 array with 8 inches of pool water in the outer cavity.

(b) Interspersed Moderation

Water densities from .15 to .75 gm/cm^3 were analyzed. The resultant Keff's are:

-4 group PDQ -7 model

fraction of full water density	Keff
.15	.5691
.25	.6276
.35	.6845
.50	.7556
.75	.8392
1.00	.8931

- KENO IV Model

.15	.5301$\pm$.0023
.25	.5958$\pm$.0029
1.00	.9124$\pm$.0043

These results indicate that the limiting moderation condition exists when water density is 1.0 gm/cm^3. Therefore misting is not a significant problem.

6.2.4 High Density Spent Fuel Rack Design

Spent fuel storage racks provide a place in the fuel pool for storing new and spent fuel. The high density spent fuel racks are of a bolted anodized aluminum construction containing a neutron-absorbing medium of natural boron carbide (B4C) in an aluminum matrix core clad with 1100 series aluminum. This neutron absorber is marketed under the trade name of Boral. The core has a thickness of 80 mils and a minimum B-10 density of 0.0232g/cm . The Boral is sealed within two concentric square aluminum tubes hereinafter called "poison cans".

Each rack consists of six basic components:

1) top grid casting

2) bottom grid casting

3) poison can assemblies

4) side plates

5) corner angle clips

6) adjustable foot assemblies

The material dimensional characteristics of the rack are:

Boral slab		(.125 x 5.25 x 152)in
a)	sheath thickness	0.0225 in
b)	sheath material	Aluminum
c)	Core material	
	B_4C	35%
	aluminum	65%
d)	B_4C a real density	0.185 g/cm^2
e)	W/o B in B_4C	70-77%
f)	a/o B-10 in B	19.75 $\pm$ 0.30%
g)	B-10 a real density	0.0232 gm/cm^2

Inner can

a)	each side	6.156 in. inside, square
b)	thickness	0.125 in
c)	Material	Aluminum

Outer can		
a)	each side	7.093 in. outside, square
b)	thickness	0.125 in.
c)	material	Aluminum

-fuel assembly pitch 6 625 in. $\pm$ .25

The construction of the rack assures that all adjacent storage cavities are separated by a Boral slab.

Programmed and Remote Systems Corporation (PAR), the rack manufacturer, assures that correct Boral locations and quantities were present in accordance with the design and procurement documents through a rigorous quality assurance program evaluated and approved by Bechtel.

Bechtel's quality assurance program through auditing, surveillance, and complete shop inspection activities at PAR's facilities, verified implementation of PAR's QA program. Also, during fabrication PP&L performed additional independent verification of PAR's QA program. Documentation related to the high density spent fuel storage racks is located at the Susquehanna site.

Periodic verification of Boron effectiveness will be accomplished through an inservice inspection program utilizing test coupons. The effectiveness of the spent fuel storage rack is determined through periodic weighing of test specimens. The test specimens are contained in the test coupons.

Each component is anodized separately. The top and bottom grids are machined to maintain nominal fuel spacing of 6.625" inches center to center within the rack and a spacing of 9.375" inches between centers of cavities in adjacent racks. Poison cans nest in pockets which are cast in every other cavity opening of the grids. This arrangement ensures that no structural loads will be imposed on the poison cans. The poison cans consist of two concentric square tubes with four Boral plates located in the annular gap. The Boral is positioned so it overlaps the fuel pellet stack length in the fuel assemblies by 1 inch at the top and at the bottom. The outer can is formed into the inner can at each end and seal welded to isolate the Boral from the SFP water. Each can is pressure and vacuum leak tested. The grid structures are bolted and riveted together during fabrication by four corner angles and four side shear panels. Leveling screws are located at the rack corners to allow adjustment for variations in pool floor level of up to $\pm$.75 inch. To maintain a flat, uniform contact area, the bearing pad at the bottom of each leveling screw is free to pivot. The bottoms of the racks are at least 7 inches above the floor to allow for coolant flow under the racks. Provisions are made for coolant flow in the corner cavities between the foot assembly and the bottom of the casting. These are top entry racks designed to maintain the spent fuel in boron-carbide/anodized aluminum poison compartments that preclude the possibility of criticality under normal and abnormal conditions. Lateral support is provided along the length of the storage cells in a manner that minimizes the application of lateral and bending loads to the fuel assembly during handling and storage. The weight of the fuel asembly is supported in a chamber hole by the bottom casting.

The location of the spent fuel storage facility within the station complex is shown in Section 1.2 of the SSES FSAR. The racks are connected to wall embedments on the pool walls and shown in FSAR Figure 9.1-2b. Each pool has 24 racks for a storage capacity of 2840 fuel bundle/assemblies plus 10 multipurpose cavities for storage of control rods, control rod guide tubes, and defective fuel containers.

The rack arrangement is designed to prevent accidental insertion of fuel bundle/assemblies between adjacent racks. The design of the rack modules and their construction requirements preclude the possibility that the racks may be put together so that adjacent fuel assemblies will not have a boral plate between them. The racks are designed such that alternating storage cells contain poison. The orientation of the modules when installed in the pool, is such that the "checkerboard" pattern will be maintained across module boundaries. The rack modules are braced to the spent fuel pit walls, such that movement are braced to the spent fuel pit walls, such that movement (i.e., slippage) along the module boundaries is prevented.

6.3 Radiation Monitoring

The training and experience of the Radiation Protection Officer is presented in Attachment 3.

6.3.1 Personnel Monitoring

Personnel authorized to uncrate, survey, store, and inspect new fuel assemblies shall carry personnel monitoring devices. Records of exposure will be maintained on Form NRC-5 or equivalent.

6.3.2 New Fuel Assembly Monitoring

Each new fuel assembly will be monitored for removable alpha and beta contamination following unpackaging. In addition, a beta-gamma dose rate survey will be taken on each assembly. Records of results will be maintained in accordance with station procedures.

6.3.3 Area Monitoring

Fuel unpackaging and storage areas will be routinely monitored for removable contamination levels and general radiation levels as applicable. Survey frequency, posting, and record keeping will be performed in accordance with Health Physics procedures.

The criteria determining the minimum area survey frequency is based on the fuel handling activities occuring and personnel access to the area. Survey frequency will be at a maximum during periods of fuel receipt when personnel access is at a minimum.

When receipt and inspection work is in progress, area monitoring will be performed daily. Otherwise, monitoring will be done on a weekly basis.

6.3.4 Instrumentation

The following instrumentation, or equivalent, will be available to support the receipt and inspection of new fuel bundle/assemblies:

Type	Model	Range	Use
Gas Flow Counter	Eberline FC-2	NA	Removable contamination survey
Ionization	Victoreen 470-A	0 mR/hr-1000 R/hr	Radiation level survey

These instruments will be calibrated on a quarterly basis.

6.3.5 Emergency Actions

Precautionary measures will be included in applicable procedures which will state actions to be followed in the event a fuel bundle/assembly is dropped and the fuel racks damaged.

The SSES Operations Shift Supervisor or designated alternate is responsible for initiation, direction, and termination of any emergency response.

6.3.6 Request for Exemption

Applicant requests exemption from the monitoring and emergency procedures requirements of 10CFR70.24. This exemption is requested because the nature of the special nuclear material, storage arrangements, and procedural controls which Applicant proposes to employ preclude any possibility of accidental criticality during receipt, unloading, inspection and storage of new fuel assemblies.

7.0 FIRE PROTECTION

Construction of the fire protection systems pertaining to fuel receipt, inspection and storage will be completed prior to receipt of fuel. Construction activities on the refueling floor involving a potential fire hazard or using flammable materials will be completed or controlled so as not to pose a threat to fuel being received, inspected or stored.

For a complete discussion of the Susquehanna SES fire protection program, see the Susquehanna Fire Protection Review Report (FPRR) previously submitted to the NRC.

Fire suppression capabilities important to new fuel receipt and storage are identified below.

7.1 Railroad Airlock and Hoistway (EL 670'-0"-818')

The railroad airlock and hoistway have been designated as fire zone 1-2C (See FPRR Subsection 4.3.13). Fire suppression capability is supplied by automatic dry pipe sprinklers and manually operated fire fighting equipment consisting of fire hose stations and portable extinguishers.

7.2 Refueling Floor (EL 818')

The refueling floor has been designated as fire zone 0-8A (See FPRR Subsection 4.3-32). Fire suppression capability is supplied by manually operated hose stations equipped with straight stream nozzles (See Attachment 4).

8.0 SECURITY

See the Susquehanna Physical Security plan which was submitted under separate cover to Mr. George McKorkle of the Physical Security Licensing Branch.

9.0 IDEMNITY

Application has been made by Pennsylvania Power & Light Company to American Nuclear Insurers for nuclear liability insurance in the amount of $1,000,000 for the period from first shipment of fuel assemblies from General Electrics manufacturing facilities in Wilmington, North Carolina, until the first fuel assembly is loaded into the reactor. Proof of financial protection is being furnished to Ira P. Dinitz, NRC Indemnity Specialist.

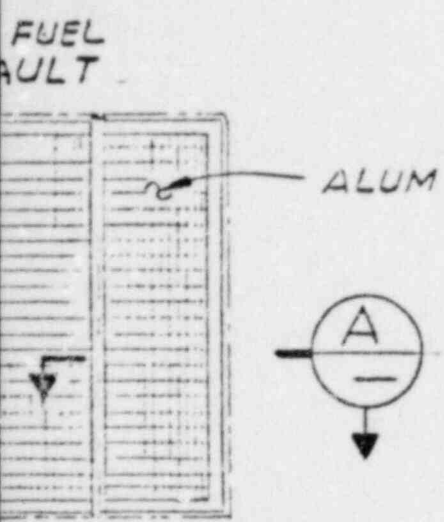
PENNSYLVANIA POWER & LIGHT COMPANY

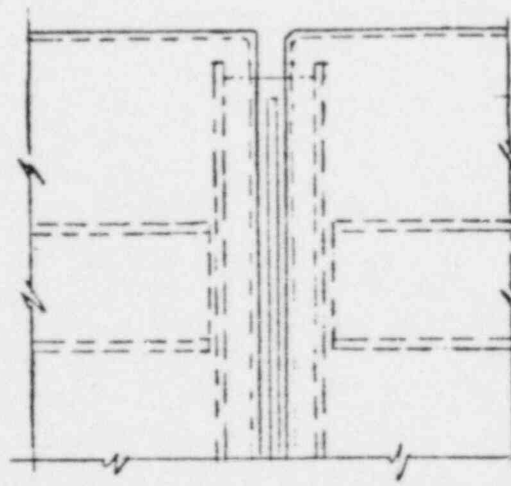
By: N W Curtis

Norman W. Curtis
Vice President
Engineering and Construction-Nuclear

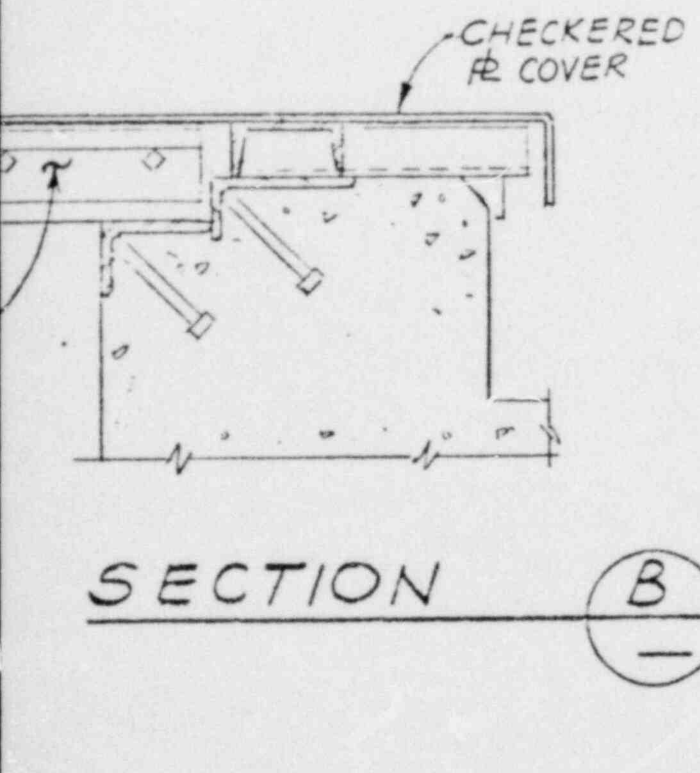
TEG#]49:5

POOR ORIGINAL





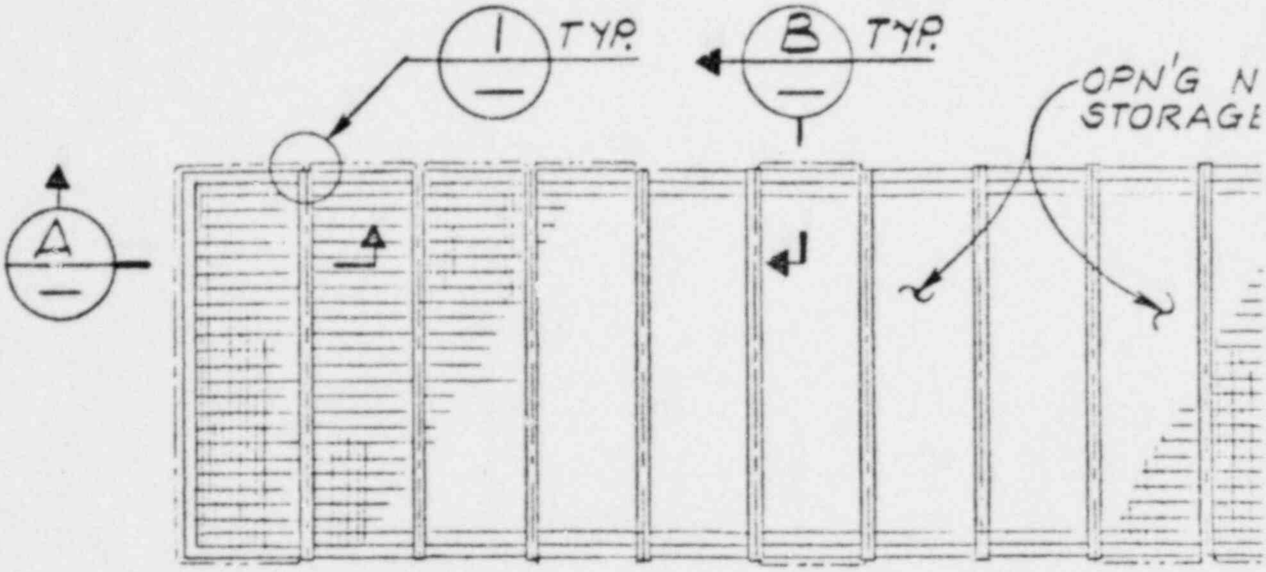
DETAIL 1



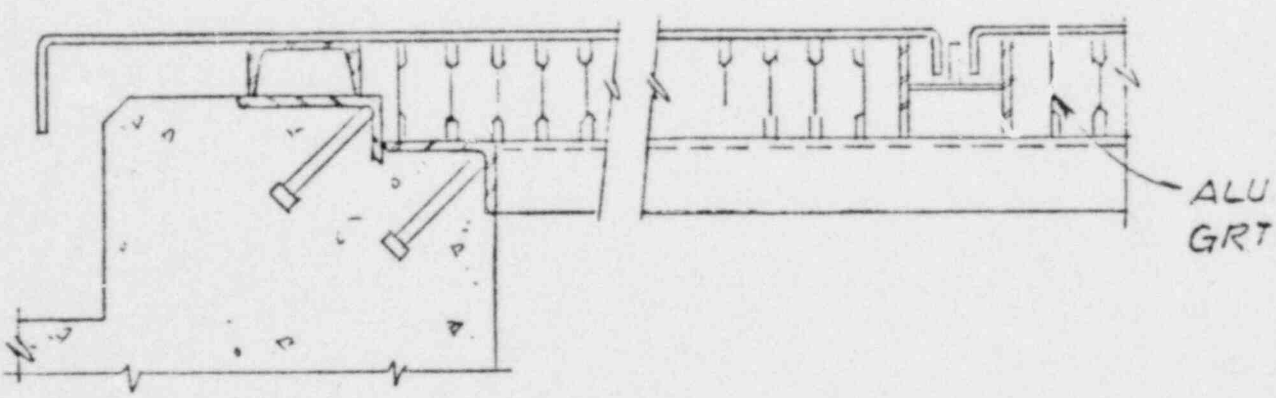
SECTION B

POOR ORIGINAL

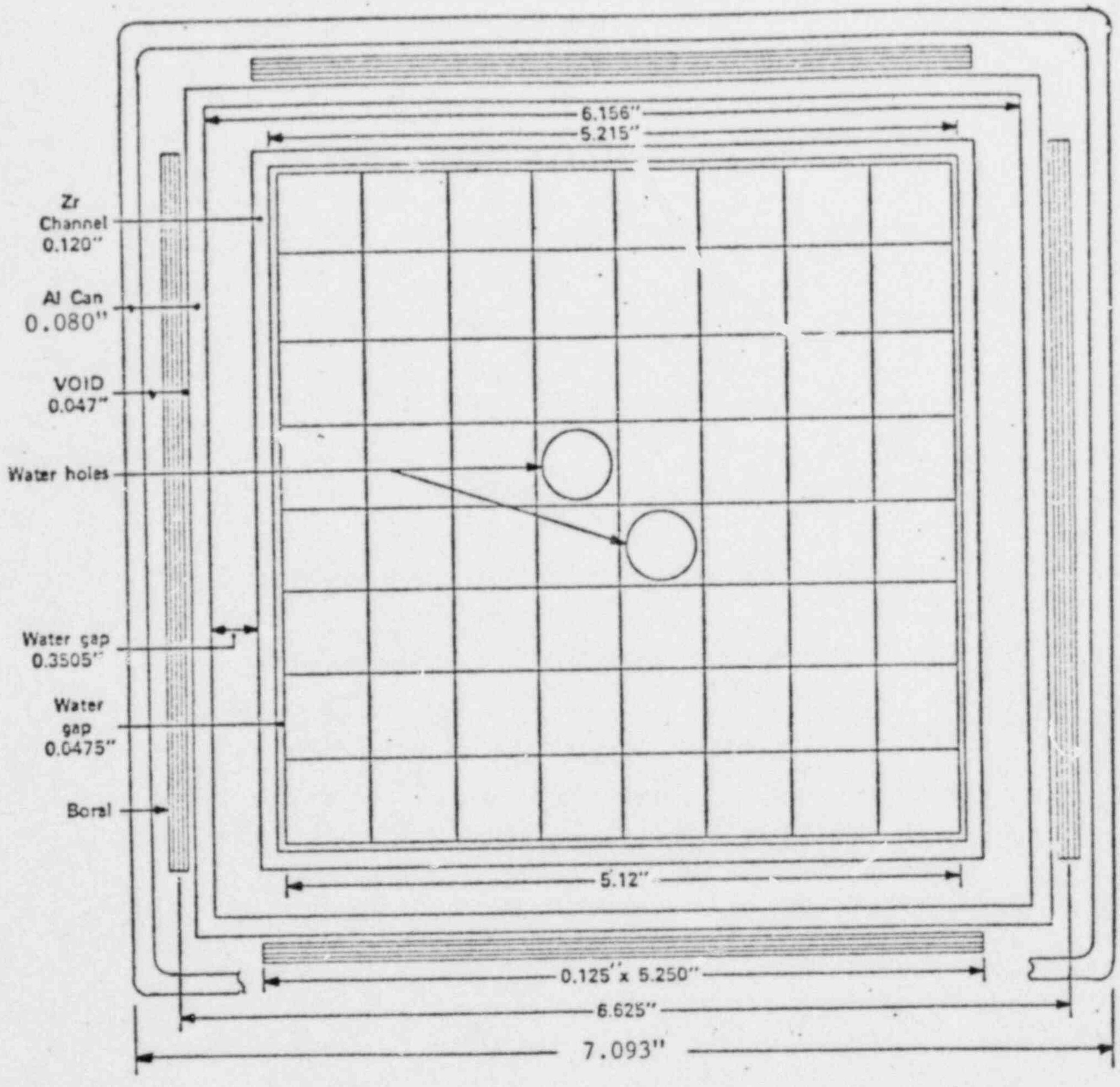
SUSQUEHANNA STEAM ELECTRIC STATIO UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
NEW FUEL VAULT COVER DETAILS FIGURE 9.1-1a



PLAN VIEW — WATERTIGHT COVER
FOR NEW FUEL STORAGE VAULT



SECTION A



SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

REFERENCE CASE FUEL STORAGE POISON CAN

FIGURE 9.1-4b

RADIATION PROTECTION OFFICER - TRAINING AND EXPERIENCE

NAME: Dennis J. Trout

WORK EXPERIENCE:

NUCLEAR WORK:

Position: Acting Health Physics Supervisor
Pennsylvania Power & Light Company
Susquehanna SES

Dates: July 31, 1979 to Present

Location: Susquehanna Steam Electric Station - Berwick, PA

Duties: Supervise Health Physics Section activities. Direct development of Health Physics Programs and Procedures to assure implementation of FSAR commitments and regulatory requirements. Selection of qualified Health Physics personnel and implementation of required training programs. Develop equipment and manpower budgets to meet program requirements. Assist in Emergency Plan and implementing procedure development. Serve as member of Corporate ALARA Review Committee, and Radiation Monitoring Review Committee.

Position: Health Physics Engineer
Pennsylvania Power & Light Company
Susquehanna SES

Dates: March 22, 1976 - July 31, 1979

Location: July 12, 1977 to July 31, 1979
Susquehanna Steam Electric Station - Berwick, PA

March 22, 1979 to July 12, 1977
Susquehanna Steam Electric Station Plant Staff
Two North Ninth Street, Allentown, PA

Duties: Developed Nuclear Station "as low as reasonable achievable" (ALARA) Program. Assisted Health Physics Supervisor in the evaluation and/or direction of Health Physics Program and acted for Health Physics Supervisor in his absence. Participated in the preparation of FSAR Section 12, developed Health Physics procedures, organized the external and internal radiation exposure program, evaluated in selected Station Health Physics equipment, participated in various Health Physics Training Programs for Station personnel. Served as Chairman of the Station ALARA Review Committee. Directed emergency on-site and off-site monitoring teams in response to March 28, 1979 accident at Three Mile Island.

POOR ORIGINAL

Position: Assistant Engineer - Nuclear - Three Mile Island (TMI) Nuclear Station - Metropolitan Edison Company

Dates: March 1, 1974 - March 12, 1976

Location: Three Mile Island Nuclear Station - Middletown, PA

Duties: Developed Health Physics procedures and implemented reporting requirements. Performed release rate and dose calculations. Implemented training programs for Station personnel and Rad/Chem. Technicians. Organized radiation exposure program and directed in-house TLD operations. Implemented the Station Health Physics Program through first line supervision of Rad/Chem. Technicians. Supervised technicians and assisted with Health Physics evaluations through initial fueling, startup, and numerous maintenance estimates for various aspects of the first refueling outage of Three Mile Island Unit I.

Promoted In January, 1976 to the position of Technical Analyst III.

Position: Public Information Coordinator

Dates: June 11, 1973 - March 1, 1974

Location: Three Mile Island Nuclear Station - Middletown, PA

Duties: Lectured to public interest groups concerning various aspects of TMINS with emphasis on environmental and Health Physics concerns.

EDUCATION AND TRAINING

1969-1973 Albright College, B.S. Physics

1975	Health Physics Certification Training/Refresher
1975	Penn. State Reactor Operations Course
1976	Respiratory Protection Programs and OSHA
1976	SSES Radioactive Waste Treatment System
1977	GE Boiling Water Reactor Technology
1977	UNI Radiological Training for Maintenance Managers and Engineers
1978	GE Radiological Engineering Course
1979	Susquehanna SES Plant Systems Course
1979	Packaging and Transportation of Radioactive Waste

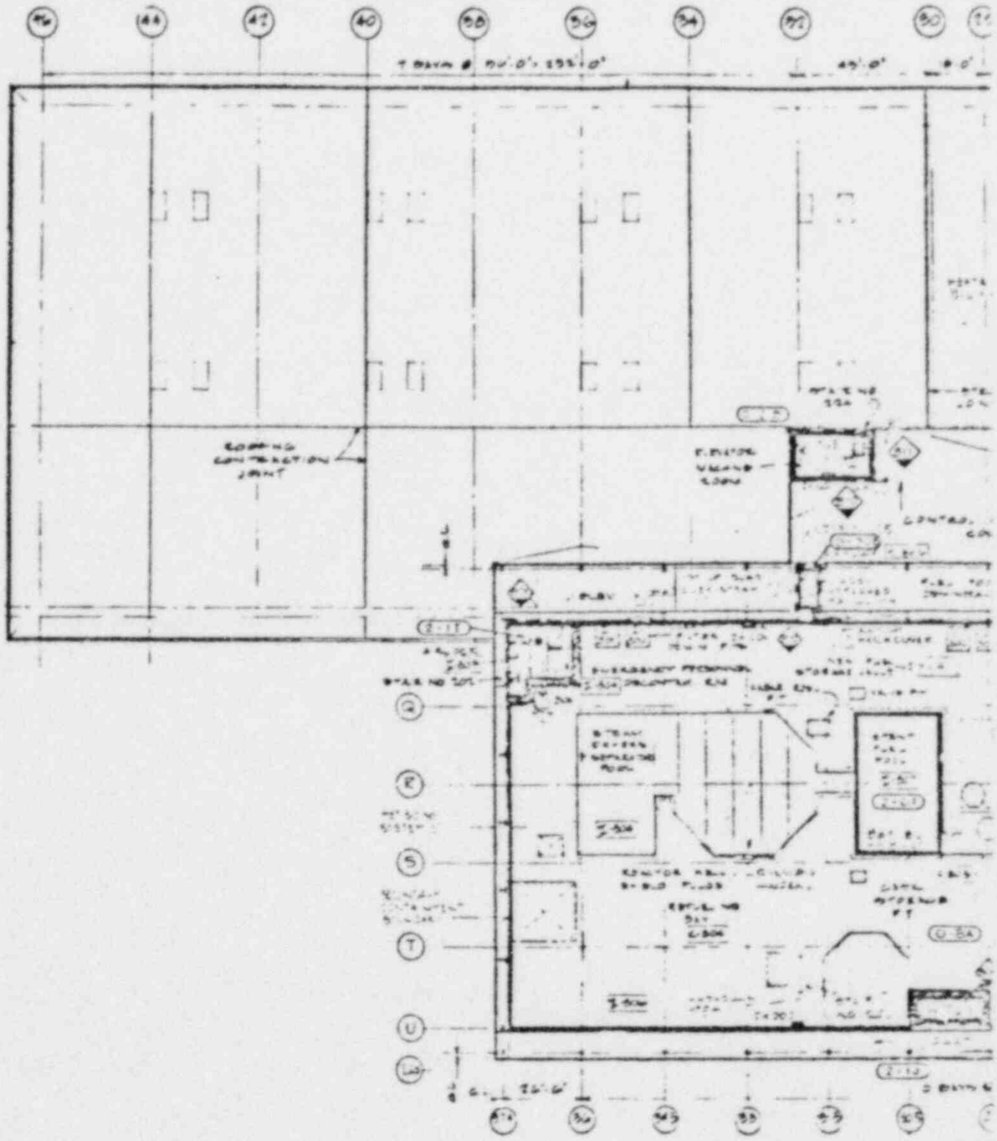
PROFESSIONAL SOCIETIES:

Health Physics Society
Delaware Valley Society for Radiation Safety

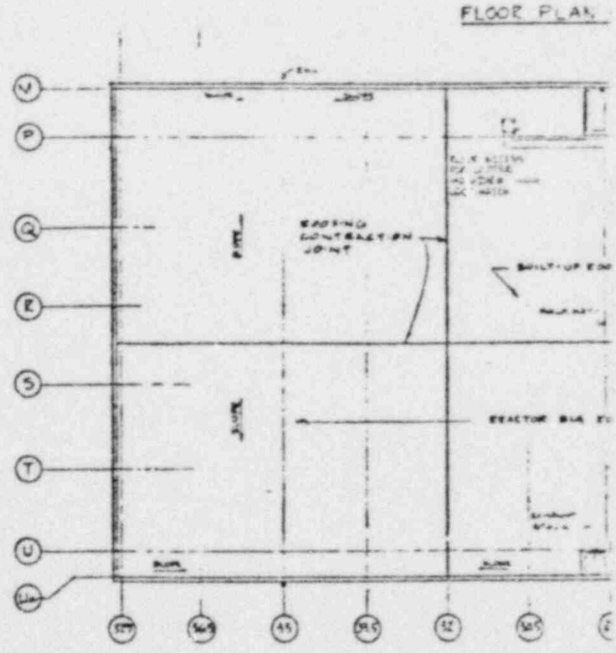
EXPERIENCE WITH RADIATION

Isotope	Amount	Location	Duration of Use	Type of Use
Mixed Fision, Activation, & Corrosion Products Byproduct, source & special nuclear Material	Trace-KiloCurie	Three Mile Island Nuclear Station	2 years	Reactor Health Physics Program

POOR ORIGINAL



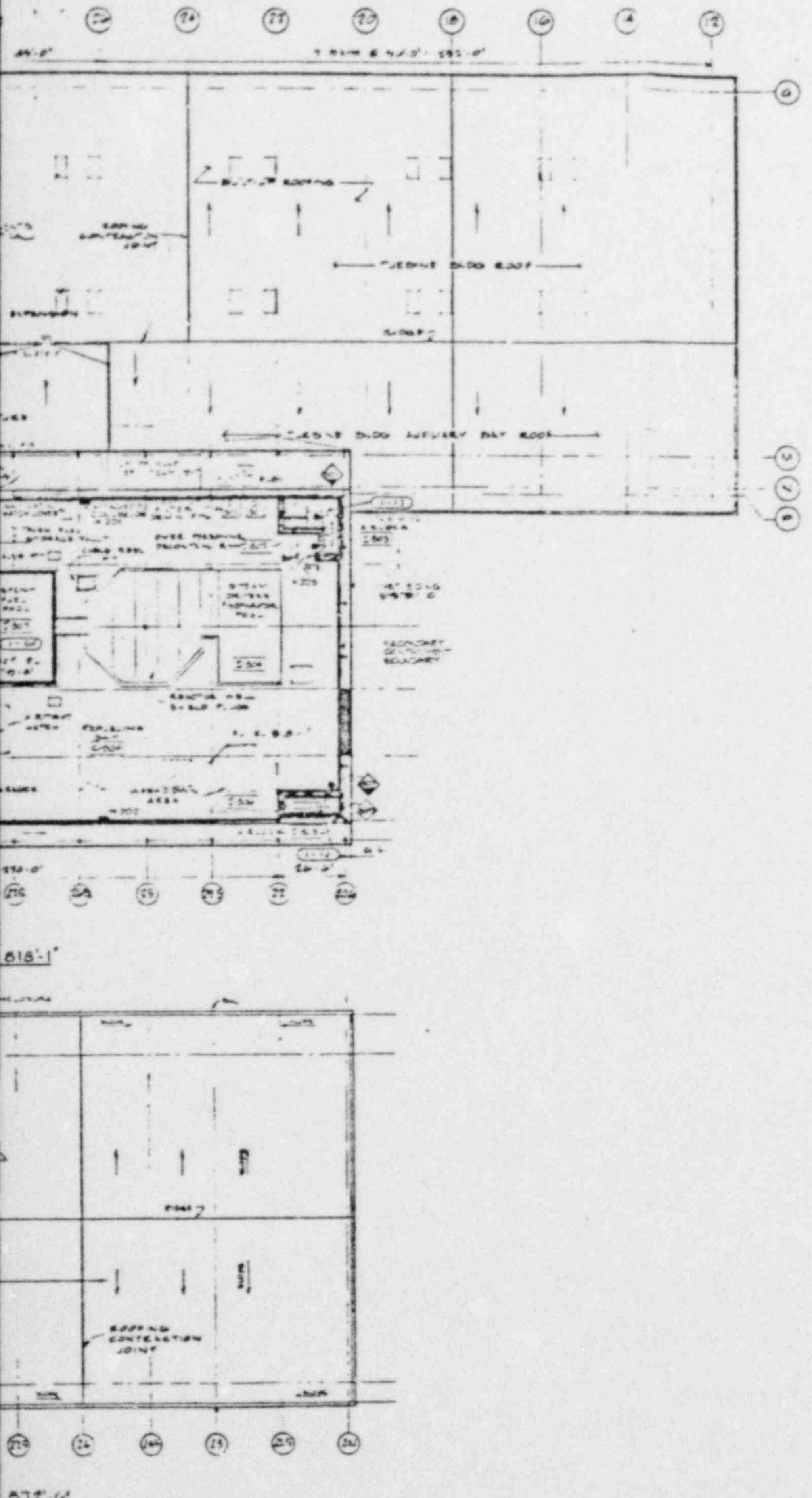
FLOOR PLAN

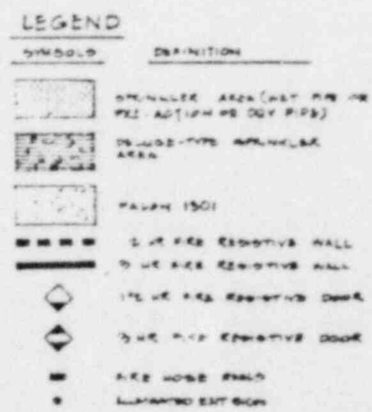


ROOF PLAN

POOR ORIGINAL

ATTACHMENT 4 TO PLA 594





PENNSYLVANIA POWER & LIGHT COMPANY

FIRE PROTECTION REVIEW REPORT
GENERAL FLOOR PLAN
EL. 818'-0" & 872'-0"

8856 FIG. 5-14

POOR ORIGINAL

1017