MARTIN COMPANY

NUCLEAR
DIVISION
Baltimore,
Maryland
21203

Mail No. 845
February 12, 1965

Refer to:
ACC-376

U. S. Atomic Energy Commission
Division of Material Licensing
Washington, D. C.

Attention: Mr. Robert Brinkman

Subject: Application for Byproduct Material License

Enclosure:
(1) Application for Byproduct Material License Form AEC-313
(2) Application for License for Radioisotope-Phosphor-Photovoltaic-Generator (RPPG) - (3 copies)

Gentlemen:

The Martin Marietta Corporation, Middle River, Maryland hereby makes application for a byproduct material license to possess 100 curies promethium-147 for utilization in research and development effort in connection with Radioisotope-Phosphor-Photovoltaic-Generators (RPPG).

Attached is Form AEC-313 "Application for Byproduct Material License," and additional information supporting our application.

Very truly yours,

MARTIN MARIETTA CORPORATION
MARTIN COMPANY - Nuclear Division

C. W. Keller
Nuclear Accountability
& Licensing Representative

CWK:mal

B/34

A DIVISION OF
MARTIN MARIETTA

Form AEC-313 (5-58)	ATOMIC ENERGY COMMISSION **APPLICATION FOR BYPRODUCT MATERIAL LICENSE**	Form approved. Budget Bureau No. 38-R027.4.

INSTRUCTIONS.—Complete Items 1 through 16 if this is an initial application. If application is for renewal of a license, complete only Items 1 through 7 and indicate new information or changes in the program as requested in Items 8 through 15. Use supplemental sheets where necessary. Item 16 must be completed on all applications. Mail three copies to: U. S. Atomic Energy Commission, Washington 25, D. C. Attention: Isotopes Branch, Division of Licensing and Regulation. Upon approval of this application, the applicant will receive an AEC Byproduct Material License. An AEC Byproduct Material License is issued in accordance with the general requirements contained in Title 10, Code of Federal Regulations, Part 30 and the Licensee is subject to Title 10, Code of Federal Regulations, Part 20.

1. (a) NAME AND STREET ADDRESS OF APPLICANT. *(Institution, firm, hospital, person, etc.)* Martin Marietta Corporation Baltimore, Maryland 21203 Internal Mail No. 845	(b) STREET ADDRESS(ES) AT WHICH BYPRODUCT MATERIAL WILL BE USED. *(If different from 1 (a).)* Martin Marietta Corporation Middle River, Maryland
2. DEPARTMENT TO USE BYPRODUCT MATERIAL Nuclear Division	3. PREVIOUS LICENSE NUMBER(S). *(If this is an application for renewal of a license, please indicate and give number.)* New Application.
4. INDIVIDUAL USER(S). *(Name and title of individual(s) who will use or directly supervise use of byproduct material. Give training and experience in Items 8 and 9.)* F. Huffman, Staff Engineer J. Neace, Senior Engineer	5. RADIATION PROTECTION OFFICER *(Name of person designated as radiation protection officer if other than individual user. Attach resume of his training and experience as in Items 8 and 9.)* R. J. Brisson, Supervisor - Health Physics Section. Resume submitted with March 29, 1963 application for byproduct license 19-1398-9.
6. (a) BYPRODUCT MATERIAL. *(Elements and mass number of each.)* Promethium-147	(b) CHEMICAL AND/OR PHYSICAL FORM AND MAXIMUM NUMBER OF MILLICURIES OF EACH CHEMICAL AND/OR PHYSICAL FORM THAT YOU WILL POSSESS AT ANY ONE TIME. *(If sealed source(s), also state name of manufacturer, model number, number of sources and maximum activity per source.)* 100 curies as $PmCl_3$ in solution or as Pm_2O_3 as powder or compact.

7. DESCRIBE PURPOSE FOR WHICH BYPRODUCT MATERIAL WILL BE USED. *(If byproduct material is for "human use," supplement A (Form AEC-313a) must be completed in lieu of this item. If byproduct material is in the form of a sealed source, include the make and model number of the storage container and/or device in which the source will be stored and/or used.)*

Please see attached application for license.

APPLICATION FOR LICENSE

for

RADIOISOTOPE-PHOSPHOR-PHOTOVOLTAIC-GENERATOR (RPPG)

CONTENTS

REFERENCES:

1. P. Miller, W. F. Senett and A. Bradley, Proceedings of the Eleventh Annual Research and Development Conference, "The Elgin-Kidde Nuclear Battery," p. 102 (1957).

2. NYO 9434, "Stability of Phosphors to Beta Radiation" (1960).

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SECTION A

RADIOISOTOPE-PHOSPHOR-PHOTOVOLTAIC-GENERATOR (RPPG) FABRICATION PROCEDURE

I. INTRODUCTION

The application of the photovoltaic effect in the conversion of radioactive energy to electrical energy consists of two steps. First, the beta energy of a radioisotope is converted to photons by a scintillating phosphor. Second, the photovoltaic (in practice, a solar cell) cell converts the photons into an electrical output by means of a P-N or N-P junction. To optimize power density and efficiency, light sources are required with brightness levels an order of magnitude higher than commercially available self-luminous materials. A conceptual design of a radioisotope-phosphor-photovoltaic-generator (RPPG) utilizing a typical promethium oxide/phosphor light source is given in Fig. 1.

The Martin development program to improve the power density and efficiency of the RPPG is expected to require a total inventory of 100 curies of Pm-147. To achieve the required uniform coating of phosphor by radioisotope, radioisotope loadings of the order of one curie of promethium-147 is required for each light source. A maximum of 20 curies of $PmCl_3$ in HCl solution will be open for processing at any one time. The completed light sources will incorporate Pm_2O_3 in intimate phosphor mixtures which are individually sealed in polystyrene capsules.

II. PHYSICAL LOCATION

The phosphor coating program will be located in the radioisotope laboratory (KJ Building) of the Martin facilities at Middle River, Maryland. This facility is located in a limited access area on a restricted sector of Martin property. Figure 2 shows the location of the facility. The floor plan of the radioisotope laboratory is given in Figure 3.

III. RADIOISOTOPE CHARACTERISTICS

The promethium-147 radioisotope will be obtained from Oak Ridge National Laboratory. The radioisotope will be aged and specially processed to reduce radiation and chemical impurities. The ratio of Pm-146/Pm-147 activity is expected to be approximately 8×10^{-7} and the europium content is anticipated to be roughly 10^{-11} curie per curie of Pm-147. Traces of Am-241 (ORNL reports about 10^{6} dis/(min-gm of Pm)) will also be present. The chemical form will be $PmCl_3$ in HCl solution. The concentration will be no less than 50 mc/ml and no greater than 2000 mc/ml. The radiochemical purity will be better than 99 percent. Promethium-147 has a half-life of 2.67 years with a maximum beta energy of 0.223 mev. The specific activity of pure Pm-147 is approximately 912 curies/gram.

A recent batch of promethium chloride solution had the following characteristics:

Pm-147 concentration --- 1787 mc/ml

Pm-146 " --- 10^{-3} mc/ml

Eu-154 " --- 5.5×10^{-4} mc/ml

Gross alpha less than 10^{4} counts/min/ml

IV. SAFETY CRITERIA

The likelihood of undue radiation exposure and contamination will be minimized by:

1. The radioisotope handling procedures from receipt of the promethium chloride solution to waste disposal will be under the surveillance of the Martin Company Health Physics Section.
2. The radioisotope laboratory is relatively isolated, has filtered ventilation, and continuous sampling of the stack effluent. The atmosphere in the work area is continuously sampled for airborne radioactivity through the use of a constant air monitor.
3. All radioisotope handling will take place inside a Kewaunee CBR Lab. 2C-304 glove box equipped with an air lock and absolute intake and exhaust filters. A negative pressure of at least 1/4 inch water guage will be maintained within the box. Glove port covers will afford protection when the glove box is not being utilized. Flexible tubing will connect the absolute exhaust filter to the filtered exhaust system of the KJ Building.
4. Cold runs of the wet chemistry operations will be made using samarium chloride as mock fuel in order to define the coating procedure and to establish safe radiochemical operations.

5. Initial radioisotope runs will be at a maximum activity of 10 millicuries in order to further refine the handling procedures.

6. Typical wet chemistry operations will involve batches of approximately one curie (ten curie maximum).

7. The maximum promethium chloride solution inventory inside the glove box will be restricted to 20 curies.

V. PHOSPHOR PREPARATION PROCEDURE

The general phosphor coating operation is outlined below. The specific loading procedures will be developed using mock fuel (samarium) coatings. Handling techniques will be refined during initial hot runs which are restricted to loading of less than 10 millicuries.

1. Concentration of the $PmCl_3$ solution is not expected to be necessary. In the event that concentration is required, a suitable quantity of $PmCl_3$ solution (corresponding, typically, to about one curie) will be transferred from the shipping container to a concentration beaker. The solution will be slowly concentrated to roughly 2000 mc/ml with either a hot plate or a heat lamp.

2. The concentrated solution is transferred to the precipitating container.

3. A slurry of the insoluble phosphor and $PmCl_3$ solution is maintained as NH_4OH is slowly added until precipitation takes place. Since rare earth hydroxides are insoluble

in near neutral solutions, promethium will be precipitated as $Pm(OH)_3$ which coagulates around the suspended phosphor particles. The supernatant is then transferred into an absorbing medium in a plastic waste bottle. The coated phosphor is dried over a hot plate or with a heat lamp. In drying the hydroxide is converted to a coating of Pm_2O_3. This configuration favors the efficient conversion of beta energy to light by minimizing the beta self absorption.

4. The Pm_2O_3 coated phosphor is next transferred to a polystyrene capsule used as the source holder. The source is sealed with polystyrene cement.

VI. RPPG FABRICATION

A sketch of the anticipated RPPG configuration is shown in Fig. 1. After the coated phosphor is sealed in a polystyrene capsule, the plastic button will be checked for contamination and cleaned as required prior to removal through the air lock for evaluation as a light source. The integrity of the polystyrene capsule will be checked by immersing the capsule in distilled water for a period of one day. The distilled water will then be evaporated and the residue counted. Any significant count above background will be evidence of a leaky light source which will be either disposed of or repaired.

Polystyrene encapsulation is chosen on the basis of its stability to radiation (Ref. 2). The preparation areas will be decontaminated as required to preclude the transfer of contamination out of the glove box.

All source removals (e.g., encapsulated light sources or planchets) will be through the air lock after being placed in a closed shielded container by suitable handling tools. When not in use the light sources will be stored in a closed and labelled shielded container in a filtered hood. Optical evaluation of the sealed light source will take place in the counting room of the critical facility (KC Building, refer to Fig. 2) which is adjacent to KJ Building.

Selected light sources will be combined with photovoltaic cells to form complete radioisotope-phosphor-photovoltaic-generators (RPPGs). Shields and handling tools will be used to minimize external radiation exposure during fabrication. The assembled phosphor-photocell sandwich will be placed in a shield and sealed in plastic for additional protection. The shield will reduce the RPPG dose rate to less than 10 mr/hr. Light sources not used in RPPG fabrication will be monitored for light degradation and leakage of radioactive materials.

VII. RADIOACTIVE WASTE DISPOSAL

Solid radioactive waste generated during source preparation will be removed from the glove box utilizing standard bag-out

techniques, sealed in a plastic bag and placed in a closed I.C.C. approved waste container for ultimate disposal at an approved site. Liquid wastes shall be poured into an absorbing medium in plastic bottles and handled as solids.

VIII. HAZARD EVALUATION

The external radiation hazard arises from the harder components of the soft bremsstrahlung produced by the Pm-147 beta. For light shielding, the Pm-146 contribution to the dose rate is very much less than that of the Pm-147 bremsstrahlung. The radioisotope is sufficiently aged and purified so that the Pm-148 impurity as well as europium and americium trace activities make negligible contribution to the dose rate.

Calculations show that typical surface bremsstrahlung dose rates on the chemical glassware surface will range around 10 R/hr for one curie of aged $PmCl_3$ in the configuration shown in Fig. 4. However, this low energy radiation is rapidly attenuated. Local low atomic number shielding and handling tools will be used to reduce personnel exposure. The dry box gloves will provide beta shielding and the design of the phosphor coating apparatus will minimize the bremsstrahlung dose rate to the hands and arms. Double gloves will be used to make hand contamination improbable. Phosphor coating operations will take place on an in-box tray and spills, if any, will be cleaned up with absorbent paper. A plastic lining will be used to facilitate decontamination of the glove box.

The dose rate at the surface of the encapsulated phosphor button is expected to be about 100 R/hr (Ref. 1) of predominately low penetrating radiation. Tongs, shielding, distance and minimum handling time will be used to reduce exposure from the sealed phosphor button. The surface dose rate from a completed RPPG generator will be less than 10 mr/hr.

The only inflammable materials which are required in the glove box are a small quantity of polystyrene cement and a few absorbent wipes, as required. In addition, electrical power will be supplied to the glove box only when responsible personnel are in attendance and the glove ports will be covered when the box is not in use. Thus, any fire hazard or danger from explosion inside the glove box would appear to be remote.

Sail switches are installed in the exhaust duct of the main system to provide an alarm to indicate exhaust system malfunction and automatically switch to emergency power. Filters on both the intake and exhaust will prevent radioisotope leakage from the glove box in the low probability case of ventilation loss from both normal and emergency system.

The most **credible** accident would be a spill within the glove box during the phosphor coating procedure. As in the case of exhaust fan failure, the activity will be contained.

On the basis of the foregoing considerations, it appears that RPPG fabrication does not present undue hazard to facility personnel.

Pages 15 through 16 redacted for the following reasons:
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(b)(4)

Fig. 3

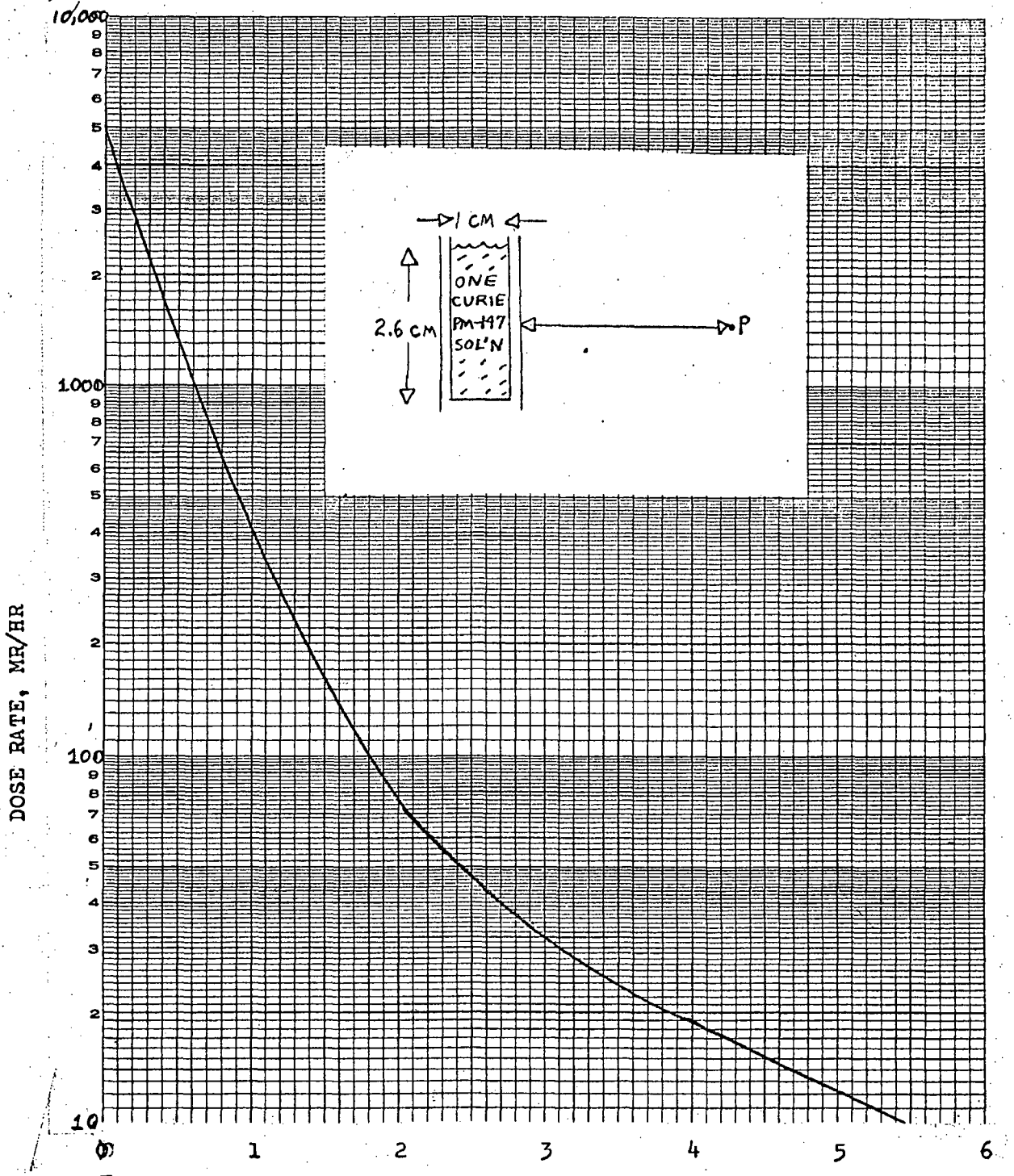

Fig. 4

Dose Rate vs. Distance from Surface of Cylindrical Source of One Curie of Aged Pm-147 in Aqueous Solution (Dose rate is given at point, P, along the midplane of the source).

SECTION B

radiation safety is achieved involve the analysis of postulated hazards resulting from both routine and accidental occurrences within the facility and evaluation of the effectiveness of the control and surveillance mechanisms instituted to cope with any unique radiation problem which might arise.

C. Scope of the Radiological Safety Program

The Martin Company Health Physics Section provides direct support in all programs concerned with radiation and radioactive materials. These include nuclear fuel element research, development and fabrication, gamma and X radiography, criticality, testing of reactor cores and radiochemical processing and handling of multi-kilocurie amounts of Strontium 90.

The major elements of this radiological safety program are:

1. Nuclear facilities design consultation
2. Program plan review
3. Radiological indoctrination and training
4. Personnel exposure surveillance and documentation including bioassay
5. Facilities, operations and environmental monitoring
6. Radioactive material hazard control
7. Radioactive waste disposal surveillance
8. Emergency planning and capability
9. Liaison with regulatory agencies

D. Health Physics Experience

The individuals comprising the Health Physics Section have had wide experience in organizing and conducting Radiological Safety Programs supporting the following nuclear activities:

1. Research, power and production reactors and nuclear accelerators.

2. Radiochemical processing plants, hot cell operations, and high level gamma and x-radiation facilities.

3. Large scale environmental monitoring and waste disposal programs, radioactive material transportation problems, personnel monitoring and instrumentation development, evaluation and calibration.

E. Radiological Safety Organization

The Health Physics Section provides complete capability for radiation protection management on a level-of-effort basis. The Supervisor, Health Physics Section reports to the Chief, Security Department who in turn reports to the Director, Support Division. In addition he reports on a "dotted line" to the Vice President, Nuclear Division and serves as the radiation protection officer.

The Health Physics Staff consists of health physicists who are primarily responsible for the radiation protection program in their own assigned areas. However, each staff member participates in the total program and any one can function in any location.

Surveys are usually performed by a group of technicians under the supervision of a health physicist.

II. Radiation - Contamination Control Measures

A. Introduction

Survey and control procedures are utilized to evaluate radiological hazards and establish effective control measures to limit personnel exposure. These surveys are conducted to determine radiation levels, the presence or absence of surface contamination

and/or airborne radioactivity in work or occupancy areas. Instrumentation currently used in the Radiological Safety Program is described in Table 1.

As a result of these surveys, suitable control measures are imposed to insure adequate personnel protection. The frequency and extent of the surveys are governed by the work performed or by existing conditions. Documentation of all surveys are recorded in operational logs.

B. External Dose Rate Measurements

The measurement of external dose rates is conducted using portable monitoring instruments. Selection of a particular instrument is based upon the type, energy and intensity of the radiation to be monitored. All portable monitoring instruments are calibrated at least once every 3 months and after each repair.

C. Surface Contamination Measurement

Each area where radioactive materials are utilized is surveyed for the presence of fixed and loose contamination. The frequency of these surveys is governed by the nature and type of operations performed. An accelerated survey schedule is employed during the initial phases of any new work or change in work utilizing radioactive materials to determine any unusual radiological hazard. Thereafter, a general daily survey is conducted in the work area at locations which are most likely to indicate changes in contamination levels. If this general survey indicates an increase in contamination level, a detailed survey is initiated to determine the cause and institute measures to reduce or possibly eliminate the source of contamination.

Loose contamination is evaluated employing the wipe technique which consists of wiping an absorbent paper over an area of

approximately 100 cm^2 and counting the wipe in a shielded detector-scaler assembly.

Fixed contamination levels are measured employing portable monitoring instruments.

D. Contamination Criteria

The contamination criteria currently in use is:

Unrestricted Area	Removable	Fixed
Gross Beta-Gamma	1000 dpm/100 cm^2	0.2 mrad/hr @ approx. 1 in.
Gross Alpha (Transuranic Elements)	100 dpm/100 cm^2	1000 dpm/60 cm^2
Restricted Area		
Gross Beta-Gamma	10,000 dpm/100 cm^2	1.0 mrad/hr @ approx. 1 in.
Gross Alpha (Transuranic Elements)	1000 dpm/100 cm^2	10,000 dpm/60 cm^2

E. Surface Contamination Control

Surface contamination is controlled in the laboratory area by employing one or a combination of the following techniques:

1. Performing regular surveys to determine contamination levels.
2. Providing filtered ventilation at the source of dispersable contaminants.
3. Establishing isolated areas when dispersable contaminants are generated.
4. Utilizing absorbent paper, polyethylene and coatings on equipment and exchanging those during the course of the work.
5. Requiring the use of anti-contamination clothing.

The spread of dispersable contaminants to unrestricted areas is minimized by:

1. Establishing buffer zones between the areas and providing frequent contamination surveys.

2. Monitoring personnel at the point of egress from the work area.

3. Monitoring equipment and utilizing an equipment check tag system.

4. Providing receptables for contaminated wastes and used anti-contamination clothing.

5. Controlling the wearing of anti-contamination clothing outside of a restricted area.

Note: (Anti-contamination clothing is not allowed in Company Cafeterias.)

F. Personnel Contamination Control

Anti-contamination clothing requirements are specified by Health Physics prior to starting any operation where potential contaminants may be encountered. These requirements are based upon operational experience and/or survey results.

Anti-contamination clothing is laundered after use in a Martin Company laundry specifically established for contaminated clothing. The laundry wastes are handled and disposed of in accordance with 10CFR20. Laundered items are monitored on a random basis since operating experience has indicated effective decontamination of the clothing. Furthermore, highly contaminated items are segregated and laundered separately or discarded as solid radioactive waste.

G. Description of Decontamination Techniques

Floor contamination is normally removed by scrubbing with mops and detergent solutions. Persistent contamination is removed by employing a mechanical scrubber and steel wool pads. The scrub solution is removed with a wet type vacuum cleaner and finally

damp mopped. Waste solution is discharged to the radioactive waste hold system.

Equipment contamination is removed by scrubbing with detergents and water or wiping with damp cloths. Large metal machinery and hand tools are decontaminated by wiping with solvent to minimize corrosion. Contaminated scrubbing equipment, mops and wipes are discarded as solid radioactive waste. Equipment exhibiting low level contamination is designated for use in contaminated areas.

N. Airborne Radioactivity Measurement

The laboratory is equipped with a Continuous Air Monitor utilizing a gas proportional alpha-beta gamma detector. The intake of the air sampler is positioned to collect samples of the atmosphere adjacent to the work area. The monitor is adjusted to detect, measure and record any hazardous release of airborne radioactivity.

Furthermore, the monitoring unit is equipped with an audible and visual alarm which is activated if the airborne radioactivity exceeds the preset level.

This unit is capable of detecting airborne radioactivity concentration as low as 10^{-12} uc/ml.

Additional 24 hour air samples are collected from the ventilation system exhaust stacks and also analyzed daily during the work week. The samples are collected from the exhaust train prior to exhausting to the environs. Exhaust stack sampling equipment consists of a vacuum system drawing approximately 1 cfm per filter assembly. Air samples are collected using in-line filter holders containing filter paper which is greater than 95% efficient for collecting air particulates greater than 0.3 microns in diameter.

Air sample filters are counted and analyzed employing gas flow proportional detector-scaler assemblies. This procedure can

adequately determine concentrations as low as 1×10^{-13} uc/ml.

I. Wipe Testing Encapsulated Sources

Each sealed light source will be surveyed to verify source integrity prior to removal from the glove box. Thereafter, the sources will be inspected frequently to assure continued capsule integrity. Once capsule integrity has been established, future tests will be made on at least 6 month intervals.

If the test reveals the presence of 0.005 microcurie or more of removable contamination upon a 100 cm^2 (equivalent) wipe test, the source will be repaired and/or decontaminated, and retested. If repair of a source is impractical, the source will be discarded as radioactive waste.

J. Source Receipt Inspection

After receipt at Martin-Nuclear, the shipping container containing the promethium solution will be surveyed for external radiation -contamination levels and removed from the delivery vehicle. The delivery vehicle will be surveyed prior to release from Martin property.

The shipping container will be transferred to a hood ventilated through a high efficiency filter. The outer container will be opened in the hood to expose the inner, primary container. This container will be visually inspected for damage and then carefully removed from the outer container. Extended handling tools will be employed depending on survey data. The primary container will be surveyed for external radiation and contamination, and decontaminated when necessary. The container will then be transferred and stored behind the shield in the glove box. If the promethium is not immediately required, it will be stored in a shielded area within a filtered, ventilated enclosure.

III. Personnel Monitoring

A. Introduction

Personnel exposure surveillance is performed to measure the amount of external and internal radiation exposure received by individuals whose work involves exposure to ionizing radiation.

B. Film Badges

All personnel permanently assigned to work in areas where exposure to ionizing radiation is likely are required to wear film badges. Additional monitoring devices are issued as dictated by the type of work performed. Film badges are currently supplied and processed by Health Physics Services, Inc., Baltimore, Maryland.

Beta-gamma film is used in all cases when personnel monitoring is required, whereas, neutron film is used only when personnel are exposed to neutron radiation. Film packets are processed and evaluated bi-weekly, quarterly, after any suspect or unusual exposure as indicated by supplementary monitoring devices. Visitors are monitored in accordance with 10CFR20.202.

C. Bioassay

All personnel whose work involves the handling of radioactive materials are periodically scheduled to submit bioassay samples for analysis to aid in determining exposure to concentrations of radioactive materials. In the case of suspected body intake of radioactive materials, special samples are collected from the personnel involved and analyzed.

D. Records

Results of all personnel exposures are posted in the individual's exposure file where a cumulative exposure record is maintained.

IV. Description of Instrument Calibration Procedures

A. Introduction

All radiation detection instruments are calibrated employing radiation standards in order to assure proper operation and measurement.

B. Portable Radiation Detection Instruments

All portable survey meters are calibrated at least once every three months or after each servicing.

Beta-gamma detection instruments are calibrated utilizing a one curie Co^{60} source. The source exposure rate at varying distances is determined employing a Roentgen "R" meter calibrated by NBS. The instrument can be remotely placed at a given distance from the source and adjusted to read the radiation intensity at that distance. Where possible, each instrument is checked at two points on each scale or range setting. Smaller sources are used to calibrate low range (0-50 mr/hr) instruments. These include a 1 mg Ra^{226} and a 5 mc Co^{60} source.

Neutron survey instruments are calibrated using Ra^{226} Be (25 mc) or Pu^{239} Be (10c) sources of known flux. Calibration is performed out of doors to reduce scattering from building, walls, etc. The instrument is adjusted to read the calculated neutron dose rate after adjusting the bias against gamma radiation.

Alpha survey meters are calibrated employing reference sources of uranium and plutonium and adjusted to read the known value in counts per minute based upon the manufacturers recommendations concerning counter efficiency.

C. Area Radiation Monitoring System

The area radiation monitoring detectors are functionally checked at least once every 3 months or after each servicing. This check

includes utilization of either a 1 mg Ra source placed at varying distances from the detector or a vendor supplied calibrator, such as the Tracerlab Model AX-2, in order to check the response of the detector. If the meter reading varies beyond 20% from the true reading, the equipment is adjusted and/or repaired.

D. Detector/Scaler Counting Assemblies

All counting room instrumentation is checked daily for background and statistical accuracy. Calibrated alpha and beta-gamma reference sources are placed in each detector and counted for a preset time. The background and total count is recorded in a log. If the instrument readings vary greater than the calculated statistical error, it is recalibrated by plotting source counts versus high voltage setting to establish the proper counting voltage. The counter efficiency is then determined by comparing the counter count rate with that of the standard source.

E. Instrument Maintenance and Records

Minor repair and instrument calibration are performed by Health Physics personnel. Major repairs are performed by the facility instrument technician. Health Physics maintains permanent records of all instrument calibrations and repair records are maintained by the instrument technican. Each instrument is tagged indicating the calibration and recalibration date.

RADIATION DETECTION INSTRUMENTS AND RELATED EQUIPMENT

TYPE OF INSTRUMENT MAKE AND MODEL NO.	NUMBER AVAILABLE	RADIATION DETECTED	SENSITIVITY RANGE (MR/HR)	WINDOW THICKNESS (MG/CM^2)	USE
Victoreen 592 Ion Chamber	4	Gamma	0-1000	-	Monitoring
Eberline E-500B G.M.	5	Beta-Gamma	0-2000	30 mg/cm^2	Surveying, measuring
Victoreen CD 700 G.M.	6	Beta-Gamma	0-50	30 mg/cm^2	Surveying, measuring
Victoreen MD 740 Ion Chamber	2	Beta-Gamma	0-10,000	7 mg/cm^2 (End window)	Monitoring
Eberline PAC-1SA Scintillation	4	Alpha	$0-2 \times 10^6$ cpm	Less than 1/4 mil mylar	Surveying, measuring
Victoreen MD GDV 715 Ion Chamber	6	Gamma	10-500,000	-	Emergency Monitoring
Eberline PAC-3G Gas Proportional	2	Alpha	$0-1 \times 10^5$ cpm	Less than 1/4 mil mylar	Surveying, measuring
Eberline FN-1A Scintillator	1	Fast Neutron	0-5000 $n/cm^2/sec$	-	Monitoring
Eberline PNC-1 BF_3-Tube	3	Fast and Thermal Neutron	$0-5 \times 10^5$ cpm	-	Monitoring
NMC Mod. PC-1 Gas Proportional Sealing Assemblies	2	Alpha Beta-Gamma	1 to 10^6 dpm	Less than 1/4 mil mylar	Measuring
Eberline PC4-4 Gas Prop. Detector with MD. PC-6 Sealing Assembly	1	Alpha Beta-Gamma	1 to 10^6 dpm	Less than 1/4 mil mylar	Measuring

TYPE OF INSTRUMENT MAKE AND MODEL NO.	NUMBER AVAILABLE	RADIATION DETECTED	SENSITIVITY RANGE (MR/HR)	WINDOW THICKNESS (MG/CM^2)	USE
Eberline Md SAC-2 Scintillation detector with TMC, Eberline or Tracerlab Sealer	1	Alpha	10 to 10^6 dpm	Less than 1/4 mil mylar	Measuring
End Window Geiger Tubes with Tracerlab MD 1000 and TMC S62A Scalers	2	Beta-Gamma	10 to 10^6 dpm	Less than 1.4 mg/cm	Measuring
Tracerlab Area Radiation Monitoring System	3 Channels	Gamma	0.01 to 10^4 mr/hr	-	Area Monitoring KJ Bldg.
NMC Model AM-2, AM-3 Constant Air Monitors	4	Beta-Gamma	10^{-12} uc/ml	1.4 mg/cm^2	Air Particulate Monitoring
NMC Model AM-22R CAM	1	Alpha-Beta Gamma	10^{-12} uc/ml	Less than 1 mg/cm^2	Air Particulate Monitoring KJ Bldg.

Gast Air Sampling Vacuum Pumps and Associated 2" Filter Holders	Air sample filters counted for alpha and beta-gamma activity	Collection of air samp : from Nuclear work area atmosphere and stack discharge to the environs.
Film Badges. (Service currently supplied by Health Physics, Inc.)	Upper limit of film response: Gamma - 600 Rem Beta - 400 Rem Neutron - 10 Rem	Film badges are worn in accordance with applicable sections of 10CFR20 as part of the Personnel Monitoring Program.

SECTION C

RESUMES OF USERS

NAME: FRED NORMAN HUFFMAN

POSITION: Staff Engineer, Engineering Department

EDUCATION: Ph. D. in Physics, Johns Hopkins University (b)(6) Ex6

U.S.A.E.C. Advanced Health Physics Fellow 1959-64. Courses in Biophysics, Modern Physics and Advanced Laboratory gave formal training in the principles and practices of radiation protection, biological effects of radiation, radioactivity measurements and radioactivity calculations.

ORSORT Graduate (b)(6) Formal course work and laboratory experiments involving principles of radiation protection, biological effects of radiation, radioactivity measurements and calculations. Ex6

M.S. in Physics, Vanderbilt University (b)(6) Ex6

U.S.A.E.C. Radiological Physics Fellow 1954-56. Formal course in Radiological Physics. Several months experience in ORNL health physics offices during summer of 1955. Masters thesis research performed in ORNL Health Physics Division, 1955-56.

B.S. in Chemistry, Lenoir Rhyne College (b)(6) Ex6

Resume of Fred Norman Huffman (cont'd.)

EXPERIENCE WITH RADIATION:

Johns Hopkins University. Experience with a variety of X-ray machines and Mossbauer sources as a Research Assistant in JHU X-ray Laboratory (1961-64).

The Martin Company. Supervised loading of thermoelectric generators (approximately 1500 curies of Po-210, sealed sources) as Assistant Project Engineer in charge of SNAP III (1958-59). Performed shield design calculations for Aircraft Nuclear Propulsion studies (1957-58).

Oak Ridge National Laboratory. Performed ORSORT laboratory experiments (1956-57). Thesis work done in ORNL Health Physics Division [(1953-56).] Ex6 Toured ORNL health physics offices during summer of 1955 as part of training program for U.S.A.E.C. Radiological Physics Fellows. Encountered wide range of production and research radiation problems.

PROFESSIONAL SOCIETIES:

Health Physics Society

American Physical Society

NAME: JAMES C. NEACE

POSITION: Senior Engineer, Development Section, Nuclear Division

EDUCATION: B.S. Chemistry - Morehead State College - Magna Cuma Laude - (b)(6) Graduate Courses in Radiochemistry, University of Maryland - 1960-61 Ex 6

EXPERIENCE: February 1961 to present: Martin Marietta Corporation, Nuclear Division, Senior Engineer, Materials Development Section. Development of purification methods and radiochemical research with operating nuclear reactors anddevelopment of analyses in support of nuclear reactor program. Research on preparation and purification of chemical compounts utilized in reactor programs.

1956 to 1960: The Martin Company, Nuclear Division, Materials Section, Supervisor, Analytical Chemistry Group. Responsible for setting up a laboratory and training personnel for the chemical analysis of nuclear reactor components. Developed new methods and techniques for these analyses.

1952 to 1956: General Electric Company, Aircraft Nuclear Propulsion Department, Materials Section, Research Chemist. Developed analytical techniques for the analysis of nuclear reactor components.

Did research work on methods of producing spherical particles of UO_2. Studied methods of salvaging uranium from scrap fuel elements. Studied production and purification of rare earth compounds.

PUBLICATIONS: "Production of Spherical Particles of UO_2" USAEC (1954)

"The Analysis of Boron by Mercury Cathode Electrolysis," Chemist Analyst (1959)

"A Spot Test for the Identification of Molybdenum Metal," Chemist Analyst (1960)

AFFILIATIONS: American Chemical Society

PROGRAM RESPONSIBILITIES: Development of analytical procedures for the Sr-90 Fuel Development Program; supervision of routine process control analyses on non-radioactive, simulated fuel materials.

Estimated Time: 100%