

APPENDIX A

Significant Appraisal Findings

Power Authority of the State of New York  
James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333  
License No. DPR-59

Based on the results of the NRC Health Physics Appraisal conducted November 10-21, 1980, it appears that several significant weaknesses exist in your health physics program as indicated below. Details regarding these weaknesses are found in the referenced sections of the appraisal report.

A. Radiation Protection Organization

1. Formally document and issue assignments of supervisors and technicians to the major functions of the department. (Section 1.3)
2. Assign additional qualified personnel at the professional level to the Radiological and Environmental Services Group. (Section 1.3)

B. Training

Implement a formal training and retraining program including lesson plans, acceptance criteria and formal examinations to maintain technician competence at a prescribed level. (Section 2.2)

C. External Exposure Control

1. Select and implement a method for personnel neutron exposure monitoring and provide training in this method. (Section 3.1.3)
2. Establish a procedure to require and specify investigations of personnel external exposures in excess of established limits. (Section 3.1.4)
3. Include in the external dosimetry program provisions and procedures for quality assurance of extremity dose and neutron dose equipment and evaluations. Include a provision for irradiation of whole body and extremity equipment by known amounts of low energy beta radiation. Include procedural requirements for review and sign-off of all quality assurance measures by responsible supervision. (Section 3.1.6)

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D. Internal Exposure Control - Respiratory Protection

1. Establish a formal program to identify, evaluate and implement corrective actions for personnel airborne radioactivity exposures in excess of "40-MPC hours". (Sections 3.2.4 and 3.2.5.3)
2. Establish an approved, documented program to relate direct and indirect bioassays to the effectiveness of the respiratory protection program. (Section 3.2.5.3)
3. Establish a quantitative respiratory equipment fit testing and training program and procedures. (Section 3.2.5.4 and 3.2.5.5)
4. Establish adequate respiratory protective equipment testing, storage, control and issuance program and procedures. (Section 3.2.5.4, and Section 3.2.5.6)
5. Establish an internal dosimetry quality assurance program and procedures consistent with ANSI N343 and Chapter 10 of NUREG-0041. (Section 3.2.5.4)
6. Establish means to ensure respiratory protective equipment is not routinely used in airborne radioactivity concentrations whose MPC fraction exceeds the protection factor of the equipment. (Section 3.2.5.7)
7. Establish a program and procedures to assure that process or other engineering controls are used to the extent practicable to limit the concentrations of airborne radioactive materials. (Section 3.2.5.7)
8. Establish a program and procedures for calibration and quality assurance checks of the whole body counter consistent with the recommendations of ANSI N343. (Section 3.2.6)

E. Surveillance Program

1. Establish and implement a formally documented and approved routine plant radiation and contamination Surveillance Program. (Section 3.3.2)
2. Establish formal procedures on the type of radiation surveys required prior to issuance of radiation work permits. (Section 3.3.2)
3. Obtain and utilize appropriate airborne radioactivity sampling equipment and media with known sampling, collection and retention efficiencies for iodine. (Section 3.3.3.2)
4. Enforce personnel contamination self survey requirements upon departure from contaminated areas and furnish an adequate number of portal monitors at appropriate locations. (Section 3.3.3.3)

F. Radioactive Waste Management Program

1. Formally assign radioactive waste shipping responsibilities. (Section 4.2)
2. Promptly repair and utilize the Off Gas Treatment System. (Section 4.3.2.1)
3. Review all radioactive waste storage areas including temporary storage areas which had not been previously reviewed to ensure a documented 10 CFR 50.59 evaluation is on file. (Section 4.4.1)
4. Establish and implement radioactive waste shipping cask loading and closure procedures to meet the requirements of 10 CFR 71.54. (Section 4.4.2.2)
5. Establish and implement means to maintain and update all documents required to be on-hand prior to shipment of radioactive waste. (Section 4.4.2.2)
6. Establish and implement a radioactive waste shipping records program which meets the requirements of 10 CFR 71.62. (Section 4.4.2.2)
7. Establish and implement a quality assurance program sufficient to assure radioactive waste is packaged, transported and transferred in accordance with applicable regulatory requirements. (Section 4.4.2.2)

G. ALARA

1. Establish, document, and implement a formal corporate and plant ALARA program that conforms to the guidance in Section C of Regulatory Guide 8.8, and to Regulatory Guide 8.10. (Section 5.2)
2. Provide full-time professional level manning plus the necessary supporting personnel to operate the plant ALARA program and provide the necessary corporate level manpower. (Section 5.2)
3. Provide procedural action levels in radiation work permit review, planning and job review, consistent with good ALARA principles. (Section 5.2)

H. Facilities and Equipment

1. Locate clothing change areas and personnel access control points consistent with ALARA principles. (Section 6.1.4)
2. Reestablish quantitative fit testing program using appropriate equipment for quantitative fitting of respirators and for retesting of repaired equipment. (Section 6.1.8)

3. Provide additional personnel contamination "frisking" stations at appropriate locations to create conditions under which the procedure for personnel contamination self-surveys can be conscientiously followed. (Section 6.1.5)



U.S. NUCLEAR REGULATORY COMMISSION  
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-333/80-20

Docket No. 50-333

License No. DPR-59 Priority -- Category C

Licensee: Power Authority of the State of New York  
10 Columbus Circle  
New York, New York 10019

Facility Name: James A. FitzPatrick Nuclear Power Plant

Inspection at: Scriba, New York

Inspection conducted: November 10-21, 1980

Team Members:

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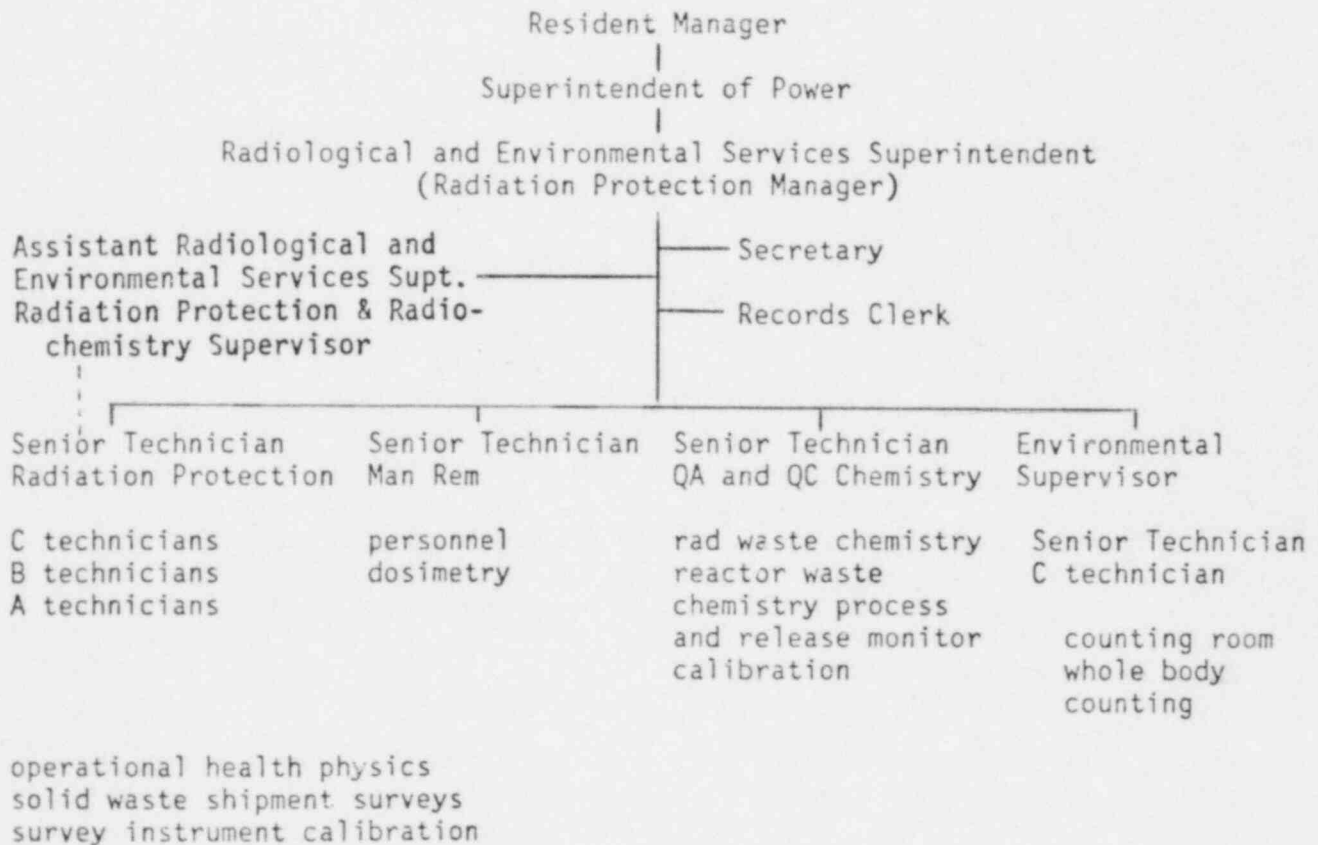
## 1.0 Radiation Protection Organization

### 1.1 Documents Reviewed

- a. Administrative Procedure No. 1.1, "Composition and Responsibility of Plant Organization", Revision 0, April 1977.
- b. Administrative Procedure No. 6.1, "Plant Chemistry Radiation Protection and Environmental Control", Revision, April 1980.
- c. Radiological and Environmental Services Department Standing Order No. 3, "Organization of the Radiological and Environmental Services Department", Revision 1, October 1980.

### 1.2 Organization Description

The organization of the RES Department as specified in RES Department Standing Order No. 3, Revision 1, is shown below.



### 1.3 Responsibilities

The responsibilities of the RES Superintendent are described in Section 7.7 of Administrative Procedure (AP) No. 1.1, Section 5.1 of RES Standing Order No. 3 which contains wording identical to that in AP No. 1.1, and in Section 5.1 of AP No. 6.1

According to AP No. 1.1, he is responsible for compliance with approved, procedures for the radiological control and protection of personnel and the general public from radiological hazards. He has overall responsibility for the radiochemistry and radiation protection areas. He has the responsibility for custodianship of byproduct material and responsibility for nuclear shipments leaving the plant. If in his opinion radiological conditions threaten a radiation hazard to plant personnel or the general public he may recommend cessation of work or that the plant be shut down.

Although Section 7.7 of AP No. 1.1 and Section 5.1 of RES Standing Order No. 3 states, "He monitors the environmental program and other functions having to do with the radiological and ecological effects of the plant", Section 5.1.1 of AP No. 6.1 states, "The RESS has the responsibility of the direction of the radiation protection, chemistry and environmental control program at the JAFNPP."

RES Standing Order No. 3 defines the responsibilities of the RES Superintendent, the Assistant RES Superintendent, the Radiation Protection and Radiochemistry Supervisor, the Environmental Supervisor and others.

However, the responsibilities of the Assistant RES Superintendent and the Radiation Protection and Radiochemistry Supervisor are so generalized that it is not possible to establish from the standing order what specific duties have been assigned as can be seen from the following direct quotes:

"The Assistant to the Radiological and Environmental Services Superintendent (Asst. RESS) shall be responsible to the RESS and assigned duties so as to function in support of the operation of the department. The Asst. RESS may be assigned to function as the RESS, RPRS or ES in their absence."

"The Radiation Protection and Radiochemistry Supervisor (RPRS) is responsible to the RESS to assist the RESS implement the radiation protection, chemistry/radiochemistry program at JAFNPP."

Although the Assistant RES Superintendent and the Radiation Protection and Radiochemistry Supervisor are separate positions at the time of the appraisal, a single individual was serving in both positions.

At the time of the appraisal the licensee had two professional level employees, twelve Power Authority of the State of New York (PASNY) technicians working in the RES Department.

The RES Superintendent, one of the two professional employees was qualified as Radiation Protection Manager under the Criteria of Regulatory Guide 1.8. The Assistant RES Superintendent - Radiation Protection and Radiochemistry Supervisor holds a B.S. in Chemical Engineering but has only limited (short course) training in radiation protection.

To establish which of the two professional level members of the RES Department were responsible for each of the major aspects of the radiation protection program and which technicians were assigned it was necessary to question the RES Superintendent. There is no documented assignment of these responsibilities.

Further, the RES Superintendent reported that he was directly managing almost every major radiation protection function of the RES operation. He stated that the Assistant RES Superintendent was managing the whole body counting operation. As can be seen from the previously presented organization chart supervision below this level is exercised by senior technicians.

At the time of the appraisal the licensee had the following technicians working in the RES organization:

Table 1.1 - Technicians Assigned to Radiological and Environmental Services Department

PASNY EMPLOYEES

<u>Classification</u>	<u>Number</u>
Senior *	4
C *	3
B	4
A	1

(Qualification requirements for PASNY technicians are listed in Table 1.2)

Contractor Employees

<u>Classification</u>	<u>Number</u>
Sr.* (not equivalent to PASNY Senior)	8
Jr.	5

\*Qualify as technician in responsible position under ANSI 18.1

The contractor technicians listed in Table 1.1 had been at the site for six months on the average. The maximum time on site was twelve months (three technicians) and the minimum was one month (three technicians).

The qualification requirements for Radiological and Environmental Technicians are formally documented and are summarized in the following table.

Table 1.2

Qualifications

<u>Technician Level</u>	<u>Education</u>	<u>Experience</u>	<u>Duties</u>
A	2 yrs. physical science including chemistry	None	under direct supervision work of noncomplex nature assists technicians of higher grade
B	Same as A	1 yr. as Technician A are equivalent	under direct supervision technical work within established procedures works with and assists technicians of a higher grade
C	Same as A	2 yrs. as Technician B or equivalent	under general supervision work of a complex nature special complex tests and studies assists in training technicians of a lower grade.
Senior	Same as A	2 yrs. as Technician C or equivalent	under general supervision works with and directs work of a group of technicians. Performs a substantial amount of the higher types conducts special complex tests and studies. Assists in training technicians of a lower grade.

Review of records indicated that all PASNY technicians had met all qualification requirements for the positions they were assigned to.

Based on the above findings, improvement in the following area is required to achieve an acceptable program.

- Assignments of supervisors and technicians to the major functions of the RES Department should be formally documented (50-333/80-20-01).
- Additional personnel qualified in radiation protection at the professional level should be added to the RES Department. The positions of Assistant RES Superintendent and Radiation Protection and Radiochemistry Supervisor should be filled by separate persons and responsibilities for major functions of the Department should be divided and carried by the appropriate individuals to avoid overloading the RES Superintendent. (50-333/80-20-02)

Based on the above findings, the following matter should be considered for improvement of the program:

- Revision of the appropriate Administrative Procedures or Standing Order consistently specify whether the RES Superintendent monitors or directs the environmental control program.

## 2.0 Training

### 2.1 Documents Reviewed

Plant Standing Order No. 9, "Request for Training Services", Revision 1, May 11, 1979.

Plant Standing Order No. 23, "Employee/Visitor Indoctrination", Revision 0, July 7, 1980.

Indoctrination and Training Procedure No. 3, "General Employee Training", Revision 0, October 28, 1978

Indoctrination and Training Procedure No. 7, "Training for Radiological and Environmental Technicians", Revision 0, May 26, 1978.



Indoctrination and Training Procedure No. 11, "Training for Professional and Supervisory Personnel", Revision 0, August 28, 1978.

Indoctrination and Training Procedure No. 14, "Respiratory Protection Training", A draft procedure under review.

## 2.2 Training Program

Training is provided to all personnel at the plant who work in areas where radiation exposure is possible. Courses are given to qualify workers to three levels. The first of these courses, designated as a course in radiation protection training, consists of four hours of instruction plus a written examination. Anyone who is to be allowed unescorted access to work in or frequent portions of restricted areas on site where radiation exposure is possible must complete this course. A second course consisting of two hours of instruction plus an examination qualifies a person as a self monitor. This means that he can work under the provisions of an extended radiation work permit.

The third course, consisting of two more hours of instruction plus another written examination is a requirement for an individual to serve as lead man. A lead man's functions relative to radiation work permits, include: initiating requests for permits, obtaining proper approvals, ensuring that proper protective equipment is on hand, bearing responsibility that those who work under the permit obey its instructions, assuring control of high radiation areas, completing appropriate sections of the permit and exercising key control.

Review of training records and contacts with several licensee representatives indicated that the appropriate levels of training were being provided. Training records were being maintained.

Visitor "indoctrination" is covered by Plant Standing Order No. 23. This consists of requiring anyone who must enter the protected area under a "visitor, escort required" security badge, to read a single paragraph regarding radioactivity and radioactive material. A page bearing this paragraph and a colored reproduction of the radiation symbol is given to each individual to sign signifying that he cannot enter areas identified by the symbol unless he has been trained in radiation safety or is accompanied by a trained escort.

Training RES technicians is covered in Indoctrination and Training Procedure No. 7. This procedure requires the RES Superintendent to assure that personnel under him are qualified by training and experience to carry out the duties assigned to them. He is also required to assure that completed forms documenting the training provided on specific topics are submitted to the Training Coordinator at the completion of any training session.



The Training Coordinator is required by this procedure to prepare, "information packages" for RES Technicians. The RES Superintendent must assure that the information in these "packages" are brought to the attention of the technicians through reading assignments or training sessions.

The "information packages" are to be initiated for reportable occurrences with general or specific application to the activities of the RES technicians and for technical specification amendments which alter the characteristics of radiation and environmental monitoring systems.

Each newly hired RES technician must receive, "plant design orientation" covering major plant systems, technical specifications, general reporting requirements and duties and responsibilities of the plant staff, within six months of the date of hire. The procedure requires the Training Coordinator to maintain completed training records.

Administrative Procedure 4.3, "Test and Inspection System" requires certification of personnel who perform tests and inspections on nuclear and environmental safety related systems and tests performed as a requirement of the Technical Specifications. Some RES technicians are certified under this procedure to conduct surveillance tests in radiation protection and chemical analysis and to perform radiation work permit surveys.

Review of training records revealed that the technicians had received the training specified. With regard to the "information packages" it appeared that training sessions had been given and attended as specified in the table below.

Table 2.1

Training Sessions Provided to RES Technicians

Year	No. of Technicians	No. of Sessions	Total Hours	Mean (hours)	No. of Sessions attended
1978	11	6	9	4	(4)
1979	12	5	6	(4)	(4.5)
1980 (thru Nov)	11	8	13.75	(4)	(6.6)

The appraiser observed from the topics of the training sessions that they were given on an ad hoc basis to cover topics of interest and that they generally met the provisions of Indoctrination and Training Procedure No. 7. He further noted that there was no indication that all RES technicians were required to receive the training provided in these sessions.

Review of the records indicated that there are no formal procedures, lesson plans or acceptance criteria for technical training of RES technicians in their specialty and there are no provisions for retraining to maintain technical proficiency.

Based on the above findings, improvement in the following area is required to achieve an acceptable program.

A formal training and retraining program including lesson plans, acceptance criteria and formal examinations designed to maintain technician competence at a prescribed level should be implemented. Further, all technicians must be required to receive the training specified by the Training Coordinator.  
(50-333/80-20-03)

### 3.0 Exposure Control

#### 3.1 External Exposure Control

##### 3.1.1 Documents Reviewed

- a. RTP-9, "TLD Operation and Calibration," Revision 0, May 22, 1978
- b. RTP-10, "Self Reading Dosimeter Calibration," Revision 1, October 19, 1979
- c. RTP-13, "Evaluation of Emergency Neutron Dosimeters," Revision 0, October 7, 1978
- d. RTP-14, "TLD Issuing and Collecting," Revision 1, March 26, 1980
- e. RTP-15, "Lost TLD Report Procedure," Revision 1, March 28, 1980
- f. RTP-21, "Condenser R-Meter Instructions," Revision 1, March 31, 1980
- g. RTP-24, "Model 1000B, Calibrator Operation," Revision 1, October, 1980
- h. RTP-26, "UD-7105 Automatic TLD System Operation," Revision 0. October 7, 1978
- i. Plant Standing Order No. 14, "Self Reading Dosimeter Control," October 19, 1979
- j. Radiological and Environmental Services Department Standing Order No. 2, "Maintenance of Radiation Protection Records," April 4, 1977
- k. Radiological and Environmental Services Department Standing Order No. 4, "Film Badge Issues for Persons with Previous Exposure," June 22, 1977
- l. Radiological and Environmental Services Department Radiation Protection Operating Procedures, Revision 4, December 26, 1979
- m. Report by School of Public Health, The University of Michigan, titled "Calibration of Panasonic Model UD-710A Automatic TLD Reader and UD-801A Personnel Dosimeters"

- n. Report by Matusushita Industrial Equipment Co., titled, "Characteristics of TL Badge Model UD-801A"
- o. "Plant Dosimetry Procedure," (under development), dated November 4, 1980
- p. Radiation Protection Operating Procedures, Revision 4, December 26, 1979, Section C, Personnel Monitoring

### 3.1.2 General

The responsibility for the dosimetry program at James A. FitzPatrick is assigned to the Radiological and Environmental Services (RES) Superintendent who has assigned a Senior RES technician to supervise day-to-day operation of the external dosimetry program. All external beta/gamma dosimetry is performed by the assigned individuals while neutron monitoring device readout is provided by a vendor.

The licensee normally badges approximately 350 individuals and uses a two week badging period. During outages, up to approximately 650 individuals are provided badges on a two week badging period cycle.

### 3.1.3 External Dosimetry Program

To read the TLD chips in the external dosimetry badges, the licensee maintains a Panasonic UD-710A automatic TLD reader. A model UD-702D manual reader is also maintained. The output, from these devices is routed to a Hewlett Packard Model 9830A computer for processing. The processed information is stored in a Hewlett Packard Model 9867B mass memory unit. A four chip (2 calcium sulfate, 2 lithium borate) Panasonic UD-801A card, which is carried in the badge, is used as the monitoring device. Using the four chips, the computer performs calculations which convert the readings to shallow and deep dose equivalents (mrem) at 7 and 1,000 mg/cm<sup>2</sup> of tissue respectively. The licensee maintains both hard copy and computer storage of personnel exposure records.

Dose equivalent correction factors have been determined using various types of radiation at the University of Michigan. These radiation types included X-rays with energies down to 20 keV, Sr-90 and natural uranium beta radiation and Co-60 radiation. The response of the UD-801A TLD card was determined to be uniform for gamma ray energies above 3 MeV.

Extremity monitoring is performed using a single lithium borate TLD chip. The chip is affixed to an extremity using tape. Extremity dose determination is made through readout of the chip with the above equipment. This single chip indicates total absorbed dose but does not provide penetrating/nonpenetrating dose discrimination.

Neutron personnel monitoring is performed using contractor (Landauer) film badges and by combining dose rate measurements and stay time data to calculate accumulated dose. The contractor badge holds a two film packet with a combination beta, gamma, x-ray film which when exposed through the appropriate badge filters, acts as the thermal neutron detector ( $E_n < 0.4\text{eV}$ ) and a second film (NTA) to provide for monitoring of fast neutrons ( $\sim 1\text{ MeV} < E_n < 10\text{ MeV}$ ). In addition, RES technicians are provided with a combination plastic monomer (Cr-39) and film device. The film provides for thermal neutron monitoring ( $E_n < 0.4\text{eV}$ ) while the plastic monomer provides for fast neutron monitoring down to neutron energies of approximately 144 MeV.

The contractor supplied film badges and combination plastic monitor and film device do not permit complete neutron personnel monitoring.

Recent NRC sponsored studies of neutron spectra at operating power reactors published as the Department of Energy's Environmental Measurement Laboratory (EML) Report EML-376 (July 1980) and Report EML-379 (September 1980) and a draft interim report (to be published) by Battelle Pacific Northwest Laboratory all indicate that the plastic monomer device (Cr-39) exhibits a lower limit of detection that is greater than the typical range of neutron doses received by workers at light water nuclear power reactors. In addition, there are gaps in the neutron energy range where the dosimeters are insensitive. The film badge will not monitor neutron energies between 0.4 eV and 1 MeV and the plastic monomer and film device is insensitive in the range of 0.4 eV to .144 MeV. Consequently, these devices are not suitable for use as the primary personnel neutron dose equivalent monitoring device.

The auditors found no formal management direction to the radiation protection staff regarding what method would be used to obtain the neutron dose equivalent. The auditors also found some radiation work permits which indicated neutron radiation levels in terms of fluence (neutrons per  $\text{cm}^2$ ) while others listed dose equivalent rates (mrem/hr). No documented instructions specifying the proper units to be used and no procedure for conversion of one set of

units to the other had been provided the technicians. The auditor noted that neutron radiation levels should be expressed in a uniform manner so that there would be no confusion among those who rely on this information for radiation protection purposes.

An exposure records staff is responsible for maintenance of exposure records. Review of selected personnel exposure records indicates that the records were being kept up-to-date. Contractor exposure records are reviewed and maintained by a Dosimetry Records Clerk at the contractor main access point. The RES Department secretary mails all worker termination and personnel monitoring reports.

The maintenance and review of the exposure records is performed in accordance with the approved Radiation Protection Operating Procedures and with a dosimetry procedure entitled, "Plant Dosimetry Procedure," dated November 4, 1980. This procedure was in a status designated as, "under development". The Radiological and Environmental Services Superintendent pointed out that this procedure was issued in this status about one month prior to the appraisal visit, was fully approved and was being used.

Auditor review of the contractor access point and record storage area revealed that a large volume of work was in progress, i.e., records review, updating, and termination of active status. The review indicated that the personnel reviewing and maintaining the personnel exposure records at the contractor access point to the controlled area did not have a copy of the Plant, Dosimetry Procedure, (under development status), for use. The auditor noted that the copy of the Radiation Protection Operating Procedures in use was an out of date revision. However, no irregularities were noted in the operation. The latest revision of the Radiation Protection Operating Procedures was issued in December 1979 approximately one year prior to the time of the appraisal. Other aspects of the external dosimetry program appeared to be conducted in accordance with approved procedures.

Based on the above findings, improvement in the following area is required to achieve an acceptable program:

- A determination of what neutron monitoring methods and equipment will be used to provide an accurate determination and record of personnel neutron dose equivalents over all neutron energies which make a significant contribution to personnel dose. The established method should be prescribed by a procedure and appropriate training in procedure implementation should be provided. (50-333/80-20-04)



Based on the findings in the above area, the following matter should be considered for improvement of the program:

- Controlled copies of the dosimetry procedures dealing with exposure review and records maintenance should be placed in the contractor control area access point records area and any other place of use.

#### 3.1.4 Exposure Review

The auditors examined the licensee's provisions for reviewing and evaluating external personnel monitoring data.

The licensee's RES Superintendent performs reviews of personnel exposure data. The auditors found that the Maintenance Superintendent also performs a review on approximately a weekly basis of the records of exposures to personnel in his organization and uses the review as part of an ALARA program which he has implemented in the maintenance department. The RES Superintendent performs his review periodically with increased frequency during outages. However, his review is for the purpose of detecting unusual exposures and to assist in preventing overexposures. There was no indication that his review was used as part of an ALARA program. With the exception of the work of the Maintenance Superintendent, there was no indication of such ALARA evaluations as the plotting of exposure trends or other forms of feedback in the exposure control program. No established frequency for these reviews was identified by the auditor. The auditors could not identify instances of reviews by higher levels of site or corporate management.

The review of this area indicated the licensee's Plant Dosimetry Procedure dated November 4, 1980, provided for a comparison of TLD and self-reader dose data to identify and evaluate dose discrepancies. This procedure also provided instructions for dealing with lost dosimetry situations. However, there was no procedure for investigating exposures in excess of established limits.

Based on the above findings improvement in the following area is required to achieve an acceptable program:  
(50-333/80-20-32)

- Establish a procedure to require and specify the investigation which must be carried out when exposures in excess of established limits occur.

- Establish a fully documented exposure review program. This program would contain the applicable exposure reviews by the appropriate levels of management recommended in Regulatory Guides 8.8 and 8.10 to provide significant input and data base for the ALARA program.
- A procedure setting forth the appropriate considerations and action for investigation of lost dosimeters, unexplained exposures and overexposures should be developed and implemented.

### 3.1.5 Exposure Limitation

Procedures were in effect which specified issuance and use of personnel monitoring equipment. Administrative limits on maximum permissible exposures below those specified by 10 CFR 20.101 and 10 CFR 20.102 are imposed through procedure and these limitations are adhered to. Procedures which provided for exposure limitation through posting, barricading and access control were also in effect. The procedures were available at the places where they were needed.

The auditors reviewed the documents listed in Section 3.1.1 of this report with respect to the topic of exposure limitation and routine issuance and use of personnel monitoring devices. The following items were noted:

- Procedure RTP-9 provides for use of supplementary dosimetry (2 TLD chips behind a 285 mg/cm<sup>2</sup> aluminum filter) for work in certain areas. Due to the thickness of the filter, monitoring of low energy beta radiation would be prevented. The procedure does not specifically indicate that the devices will only supplement the 4 TLD chip badge nor does the procedure prohibit the use of the supplementary dosimetry in-lieu of it.
- Radiological and Environmental Services Department Standing Order No. 4 provides for issuance of film badges if official or unofficial\* exposure results total less than 1500 mrem and a NRC Form #4 is on file but, this standing order does not take into account that 10 CFR 20.101 places a limitation of 1250 mrem per quarter when the information required by 10 CFR 20.102 has not been obtained.

\* Official results, as used here, means results which have been determined through evaluation of a dosimetry system, the use of whose components are controlled and whose operation is subject to quality assurance. Unofficial results are obtained from programs which do not have all of these controls. For example, a record based on pocket dosimeters read by the individual himself is "unofficial."



Further, the procedure specifies only film dosimetry but the licensee, in fact, uses TLD dosimetry for most situations. In addition, the procedures contain no requirement that unshielded beta dosimetry be used when exposed skin surfaces are involved.

The auditors noted that principal means of external exposure limitation included posting and barricading of radiation and high radiation areas and locking of high radiation areas with dose rates in excess of 1000 mrem/hr and the use of radiation work permits (RWPs). During tours throughout the plant, surveys were performed by the auditors to determine if posting, barricading and locking was in accordance with the facility Technical Specifications and 10 CFR 20 requirements. For the most part, high radiation areas were controlled adequately.

During a tour of the Turbine Deck (300 ft. elevation North Side) on November 10, 1980, the auditors identified an accessible radiation area, caused by radiation emanating from the turbine generator, which produced whole body dose rates up to 25 millirem/hr and the area was not posted. The auditors noted that 10 CFR 20.203(b) requires each radiation area to be conspicuously posted and that 10 CFR 20.202 to define a radiation area as any accessible area where personnel could receive 5 millirem in any one hour. Licensee representatives immediately posted the area upon notification.

Regarding use of RWPs for exposure control, Radiation Protection Operating Procedure Section II.A.1, states in part, "A Hi Rad Area is any area in which an individual could receive in one hour a whole body dose in excess of 100 millirem... Entrances to Hi Rad Areas shall be made under the radiation work permit procedures." Section II.B.3 of the radiation work permit procedure requires a radiation work permit for all work in a High Radiation Area.

During a tour of the controlled area on November 10, 1980, the auditor noted one individual performing non-destructive testing of a scram discharge line at the 272' elevation of the Reactor Building. The auditor review of the area where the individual was working and discussion with the individual indicated the line emanated contact radiation dose rates of 2000 millirem/hr which produced dose rates accessible to the whole body in excess of 100 millirem/hr. The auditor determined that the individual was working

without a RWP and had been testing this line for approximately one week (also without an RWP). Licensee representatives were notified of the above and took action to ensure all individuals performing the testing were cognizant of the RWP requirements.

This item was noted to indicate inadequate control of High Radiation Areas by use of the RWP for exposure control purposes.

Based on the above findings, improvements in the following areas are required to achieve an acceptable program:

- Improve surveillance requirements to ensure radiation areas are posted properly (50-333/80-20-05)
- Provide adequate control and oversight of on-going High Radiation Area Work for exposure control purposes. (50-333/80-20-06)

Based on the findings in the above area, the following matter should be considered for improvement of the program:

- Procedures should be reviewed and revised to assure that they relate to current conditions and that they are consistent with the current regulatory requirements. The procedures should reflect implementation of the ALARA concept.

### 3.1.6 Quality Assurance

The auditors reviewed the Quality Assurance (QA) program for the licensee's whole body TLD dosimetry system described in Section 3.1.3 of this report.

The licensee's Plant Dosimetry Procedure dated November 4, 1980, provides for QA of the TLD system in accordance with Draft American National Standard, ANSI N13.11, "Criteria for Testing Personnel Dosimetry Performance" and included semi-monthly verification of TLD system performance by TLD irradiations at the plant with subsequent readout. The exposures were in the protection range (.03 to 10 rem). The irradiations are performed in an Eberline Model 1000B calibrator (up to 130 curies Cs-137) in accordance with the calibrator operation procedure, RTP-24 and the guidance provided in the procedure discussed above. Also provided for in the QA portion of the procedure is the laboratory testing of the TLD badges in the radiation test categories deemed appropriate for the radiation environment of the facility.

It was also learned that the RES technicians are required to wear beta/gamma film badges which are processed by a vendor and compared to the TLD results. This was reported to be a QA measure. However, it was not made clear how this technique could be expected to provide quantitative information that would contribute to QA.

Auditor discussions with licensee radiation protection personnel indicated the TLD system in use had been tested by the University of Michigan. The tests performed included testing in the appropriate radiation categories presented in ANSI N13.11.

The review of this area indicated the following shortcomings:

- The licensee's whole body dosimetry QA program did not include irradiation of TLD badges with low-energy beta radiation and subsequent TLD system performance determination.
- There was no established QA program for extremity dosimetry.
- Procedure RTP-10 was not developed consistent with Regulatory Guide 8.4 in that those self-reading dosimeters issued to personnel as primary monitoring devices, e.g., visitors and personnel on facility tours, were not calibrated quarterly.
- There was no established QA program for neutron dosimetry.
- No provisions were evident in the TLD system QA procedures for formal review or sign-off by a responsible individual to assure that the QA test results had been evaluated and found acceptable.

Based on the above findings, improvements in the following areas are required to achieve an acceptable program (50-333/80-20-07):

- Inclusion of low energy beta radiation tests in the whole body dosimetry QA program.
- The establishment of a QA program for extremity dosimetry.
- The inclusion in the current dosimetry QA program of provisions for formal review or sign-off by a responsible individual of QA test results.

- The establishment of a neutron dosimetry QA program.

Based on the above findings, the following matter should be considered for improvement of the program:

- Revision of Procedure RTP-10 to provide for quarterly calibrations of self-reading dosimeters used as primary monitoring devices.

### 3.2 Internal Exposure Control

#### 3.2.1 Documents Reviewed

- a. Procedure No. CRI-6, "Whole Body Counter - Operation and Calibration," Revision 0, November 1980.
- b. RES Department Standing Order No. 5, "MPC Hour Data Collection," Revision 0, June 1980.
- c. "Radiation Protection Operating Procedures," Paragraph A, Revisions 0-4, June 1978 - December 1979.
- d. Whole body counting results for the period from June 1979 thru October 1980.

#### 3.2.2. General

The responsibility for the internal dosimetry program at the James A. FitzPatrick plant is assigned to the RES Superintendent who has assigned a Senior RES technician to supervise this area. All internal dosimetry, is limited to whole body and thyroid counting.

#### 3.2.3 Internal Dosimetry Program

Licensee representatives indicated other bioassay techniques, e.g., excreta analyses have not been used at the FitzPatrick plant on the basis that the radionuclides that are controlling in terms of internal exposure are readily detectable by gamma spectroscopy. The auditor discussed the need with the licensee to formally document the technical basis for not conducting a routine excreta analysis program and discussed the need to have a contingency bioassay program available to be utilized when a certain pre-determined triggering event occurred, e.g., the measurement of an organ burden > 10% of the permissible organ burden.

The whole body counter (WBC) utilized at this facility consisted of a vendor supplied, chair counter located in the plant near the health physics control point. The WBC was equipped with two large (4" x 4" and 4" x 5") NaI

crystals for trunk counting (GI tract and chest) and a smaller (1 1/2" x 2") NaI crystal for thyroid counting. All three detectors were coupled to a 4096-channel multi-channel analyzer, the latter was coupled to a mini-computer with a spectrum stripping software package.

Output data available to the operator consisted of a CRT display of the gamma spectrum for all three detectors and a "hard copy" printout from the mini-computer. The printout contained the individual's name, TLD number, social security number, date of count, duration of count, comment, i.e., purpose of count, and specific information relative to the radionuclides detected. The specific radionuclide information for each detector included the nuclide identification, total number of net counts in the photopeak areas, the organ burden ( $\mu\text{Ci}$  at the date of the count and at the date of the uptake), the percent of Maximum Permissible Organ Burden (MPOB), and a graphic display of the gamma ray spectrum for each of the three detectors.

The licensee's whole body counting program was described primarily in Procedure CRI-6, although part A of the Radiation Protection Operating Procedures contained general information on the subject. Procedure CRI-6 addressed the routine and special circumstances which warranted whole body counts, described the performance of daily energy calibration checks and background checks, and contained a detailed listing of all steps necessary for proper operation of the analyzer, mini-computer, and associated software. The procedure also addressed the annual primary calibration of the WBC using NBS-traceable sources. Section 3.2.6 of this report discusses the whole body counter quality assurance program.

Procedure CRI-6 recommended a whole body count semi-annually for FitzPatrick employees and entrance and exit counts for contractor employees. According to a licensee representative, the obtaining of exit counts for contractor personnel is ensured because the last 1/2 - 1 day of employment is utilized as a "processing out" day. Special whole body counts were recommended when an individual's nasal swab exceeded 10,000 dpm or when an exposure in excess of 40 MPC-hr/wk\* occurred. The frequencies for routine and special whole body counting were found to be consistent with those recommended in ANSI N343.

\* 1 MPC-hr is the value specified in Table 1, Column 1 of Appendix B to 10 CFR 20.



The auditor reviewed approximately 650 whole body counts performed from June 1979 to the date of this appraisal. Of that number, 99% were below 1% of the MPOB for Co-60 in the lung. The remaining 1% (6 individuals) had Co-60 lung depositions ranging from 1-4% of the MPOB. Some of the activity in these six individuals appeared to be external contamination which had been significantly reduced, but not yet eliminated, by up to two or three decontamination showers.

The WBC was operated by RES Technicians. Although nearly all RES technicians were, according to their supervision, able to operate the WBC, one technician appeared to be assigned this task on a routine basis. This technician had an A.S. Degree in Applied Science and had worked as a nuclear medicine technician in a hospital. Prior to joining the RES Department (approximately one year ago), he had received approximately one week of on-the-job training following the initial assignment of his responsibility to this area. Management overview of this program appeared to be continuous in that each of the 650 whole body count records scanned by the appraiser had been signed (indicating that a technical review had been performed) by the Radiation Protection and Radiochemistry Supervisor.

Temperature control of the WBC room appeared to be a problem during hot weather. The licensee monitored analyzer drifting (based on the position of certain photo peaks) between May 29 and July 27, 1980. On warmer days (> 90°F in the WBC room), the daily drifting ranged from 2 channels (tolerable) to 25 channels (intolerable). When the drifting became significant (4 or 5 channels or more), the operator had no recourse but to discontinue operation of the WBC. The licensee should examine the ventilation system which supplies the WBC room and implement a solution to the temperature control problem. This would assure the availability of the system for performance of a critical whole body count.

Based on findings in the above area, the following matter should be considered for program improvement:

- The modification of existing whole body count room ventilation or physical movement of the whole body counting system to an area or room not affected by temperature extremes.

- The establishment of an excreta analysis program (indirect bioassay program) for routine and emergency use.

#### 3.2.4 Internal Exposure Review

Management review of the whole body counting data was virtually a continuous process. Procedure CRI-6 contained the requirement to perform an organ dose calculation if the cumulative fractional organ burden (sum of all nuclides detected) exceeded 10% of the MPOB. Licensee representatives indicated this value had never been exceeded. This was verified on a sampling basis by the auditor.

The use of whole body count data for the estimation of MPC-hr exposures was discussed with licensee radiation protection management. The auditor indicated that although whole body count organ dose calculations appeared to be consistent with ICRP and NCRP practices, this data was never used to calculate the intake of radioactive material through inhalation or absorption through the skin. The procedures in this area made no provision for such an evaluation. The auditor pointed out that 10 CFR 20.103(a)(3) specifies that,

"... in addition, as appropriate ... (the licensee) ... shall use measurements of radioactivity in the body, measurements of radioactivity excreted from the body, or any combination of such measurements as may be necessary for timely detection and assessment of individual intakes of radioactivity by exposed individuals".

The auditor noted that in certain cases 10% of MPOB in a whole body count would exceed the amount observed in such a count of an individual who had an intake which exceeded the 40 hour control measure specified in 10 CFR 20.103(b)(2). This is illustrated in Table 3.1. Accordingly, it would not be possible to meet the requirement of 10 CFR 20.103(a)(3) to make a timely detection of intakes of these isotopes at the 40 hour level and substantially above it, utilizing the existing plant procedures.

Table 3.1

## Whole Body Count Values Compared to Inhalation Values

Isotope	10 CR 20 Appendix B Table 1 Column 1 Value ( $\mu\text{Ci}/\text{ml}$ )	Amount Inhaled 40 hr. work (1 week ( $\mu\text{Ci}$ )	Immediate Whole Body Count which indicates a 40 hr. intake 2) ( $\mu\text{Ci}$ )	Percent of Maximum Permissible Organ Burden ( $\mu\text{Ci}$ )		
				0.36	1.7	10
$^{60}\text{Co}$ (Insol)	$9 \times 10^{-9}$	.4356	.218	.036	.170	1
$^{137}\text{Cs}$ (Insol)	$1 \times 10^{-8}$	.484	.282	.108	.510	3

- 1) Value in footnote 3 to 10 CFR 20.103(a) divided by 13, which is  $6.3 \times 10^8$  ml divided by 13 =  $4.84 \times 10^7$  ml air inhaled in a 40 hr work week.
- 2) Assuming half of the material inhaled is immediately exhaled. If time has elapsed between intake and whole body counting the values in this column will be correspondingly smaller.

The performance of MPC-hr exposure estimates was necessary to ensure the requirements of 10 CFR 20.103 were met and MPC-hr values derived from WBC data can also be used to judge the effectiveness of respiratory protection and air monitoring programs.

MPC-hr data were generated by the licensee from air sample results pursuant to RES Department Standing Order No. 5, "MPC Hour Data Collection," however, since whole body counting data were not evaluated in terms of MPC-hr exposures, a comparison between the two data bases could not be made. This is also discussed in Section 3.2.5.3.

Investigations of personnel exposures to high levels of airborne radioactivity or contact with high levels of surface contamination are triggered by high air sample results, positive nasal swabs, or high skin contamination monitoring results. Licensee action following these events appeared to be timely and complete. Investigation thresholds contained in Procedure CRI-6 were consistent with ANSI N343 and are low enough to assure that no exposure in excess of the limits of 10 CFR 20.103 go uninvestigated. Documentation of these events appeared adequate. However, the licensee had no formal program to investigate personnel exposure in excess of 40 MPC-hrs in order to prevent recurrence. The auditor pointed out that such action is necessary to assure compliance.

Based on the above findings, improvement in the following area is required to achieve an acceptable program:



- The establishment of a formal program to identify, evaluate and take corrective actions to prevent recurrence of personnel exposure to airborne radioactive materials in excess of 40 MPC-hours.  
(50-333/80-20-08)

### 3.2.5 Internal Exposure Limitations

#### 3.2.5.1 Documents Reviewed

- a. PSO 2, Revision 1, "JAFNPP ALARA and Respiratory Protection Policies," dated November 4, 1980.
- b. AP 6.1, Revision 3, "Plant Chemistry, Radiation Protection and Environmental Control," dated November 4, 1980.
- c. RESS Standing Order No. 2, Revision 0, "Maintenance of Radiation Protection Records," dated April 11, 1977.
- d. RESS Standing Order No. 3, Revision 1, "Organization of Radiological and Environmental Services Department," dated November 4, 1980.
- e. RESS Procedures, Revision 4, "Radiation Protection Operating Procedures," dated December 26, 1976.
- f. ITP 14, Revision 0, "Respiratory Protection Training," dated October 1, 1980.
- g. Lesson Plan CRI 6, Revision 0, "Respiratory Protection Whole Body Counter Operation and Calibration," dated October 1, 1980.
- h. RTP 22, Revision 1, "Air Pak Air Compressor Instruction," dated January, 1980.
- i. RTP 23, Revision 2, "High Volume Portable Air Sampler," dated January, 1980.
- j. Procedure WAC 10.1.7, Revision 2, "Housekeeping and Cleanliness Control," dated May, 1980.
- k. QA Surveillance Report SR-613, "Radiation Protection Training," dated May, 1979.

- l. QA Surveillance Report SR-648, "Issue and Control of RWPs," dated March, 1980.
- m. Procedure F-OP-39, Revision 1, "Breathing, Instrument and Service Air System," dated July, 1980.
- n. JAF Scott Air Equipment Check List for period January thru November 1980.
- o. Control Room Breathing Air Monthly Maintenance Check List for period February, May, July, August and November, 1980.

#### 3.2.5.2 General

The licensee's methods for exposure limitations consisted of a combination of procedural controls, such as protective clothing, respiratory protection, and, to a lesser extent, engineering controls.

The Radiation Protection Operating Procedures (RPOP) contained a listing of surface contamination levels and the associated protective clothing requirements for different types of jobs to be done (ranging from observation of activities to direct contact maintenance work with highly contaminated components). Respiratory protection was prescribed by the RPOP when airborne radioactivity concentrations exceeded 25% of the 10 CFR 20, Appendix B values (for a 40-hour work week) and 14% of the Appendix B values for work situations which exceeded 40 hours/week.

The auditor noted the use of job-specific procedural controls relative to the use of protective clothing and respiratory protective equipment to be provided by Radiation Work Permits (RWPs). The protective clothing and respiratory protective equipment specified in the RWP appeared to be appropriate for various jobs observed by the auditor during tours of the facility.

The types of protective clothing in use by the licensee appeared appropriate to cover a broad spectrum of work activities and surface and airborne contamination levels. Procedures for donning of such clothing appeared adequate.

Posting of areas requiring protective clothing and respiratory protective equipment was also observed during tours of the facility. These appeared adequate.

The auditor evaluated the licensee's Respiratory Protection Program including air sampling; use of engineering and MPC-hour controls; medical qualifications; training, and maintenance and issuance controls for respiratory protection equipment. This was accomplished through the review of records as well as observations and discussions with licensee representatives within the radiation protection organization.

The FitzPatrick Respiratory Protection Program had received NRC approval and licensee representatives indicated it was operated in accordance with 10 CFR 20.103(c). This program has been in operation since 1977.

#### 3.2.5.3 Administrative

The Radiation and Environmental Safety Superintendent (RESS) is assigned responsibility for the Respiratory Protection Program. The RESS had received respiratory protection training and appeared capable of evaluating the various hazards requiring respiratory protection, recommending engineering controls, and specifying appropriate respiratory protection or forbidding its use if such conditions warranted it.

The Respiratory Protection Program, appeared to lack central supervision. It was noted that the RESS had been assigned responsibility for every major component of the Radiation Protection Program but there had been no formal documented delegation of the authority to supervise the respiratory protection program to any position below the RESS.

A plant policy statement was signed on November 4, 1980 by the Resident Manager and was approved by the Plant Operation Review Committee (PORC) on that date. The statement was added to an existing Plant Standing Order No. 2 which originally addressed only the ALARA Program. The policy statement dealt with the requirements of Regulatory

Guide 8.15 and referred the reader to the Radiation Protection Operating Procedures (RPOP) for details of the program.

The RPOP described respiratory protection equipment, its issuance, maintenance, selection, use, and return and which set forth the requirements for training and qualification of personnel who are to wear the equipment.

The appraiser noted that the procedures and standards were generally followed in the field with the single exception that training documentation did not appear to reflect all training which was received.

The licensee maintained sufficient records to permit the licensee to perform an evaluation of the Respiratory Protection Program effectiveness. However, it was not apparent that such an evaluation took place in any formal fashion. Data regarding exposure to airborne radioactivity generally was placed in an individual's file, and it did not appear that exposure determined by whole body counting was filed in an easily accessible form that would permit easy, logical assembly of data to observe trending or group exposure evaluation or ALARA based evaluations.

Whole body counting was used to verify the Respiratory Protection Program effectiveness, however, it was determined that the review was quite informal and not adequate to evaluate the program effectiveness. The auditor noted that Regulatory Guide 8.15 requires that a respiratory protection program be established that includes as a minimum, surveys, as appropriate, to evaluate an individual's exposures and to assess protection actually provided. The auditor noted the inadequate review would preclude identification and corrective action for airborne exposures in excess of 40 MPC-hours.

It was noted that only one type of filtered air mask (MSA) and one type of supplied air mask (Scott) was available for users of respiratory protection equipment.

Nasal swabs were frequently performed on people who had been using respiratory protection equipment as a means for verifying system integrity. Positive nasal swabs frequently resulted in whole body counting of the individual.

Based on the above findings, improvements in the following area is needed to achieve an acceptable program:

- The establishment of an approved, documented program to relate bioassays to the effectiveness of the respiratory protection program. (50-333/80-20-09)

Based on the above findings, the following matter should be considered for improvement of the program:

- Assignment of respiratory protection program responsibility to a qualified individual to provide day to day program monitoring and supervision.

#### 3.2.5.4 Respiratory Protective Equipment Use

Only MSHA/NIOSH approved respiratory protection equipment was being used at the FitzPatrick station. The station had previously used a Scott 282-TA-R combination cannister until October, 1980 but subsequently switched to the MSA/GMR combination filter charcoal cannister which was an approved type. No credit was taken for radioiodine protection for the combination cartridges. The SCBA and airline respirators used were also MSHA/NIOSH approved. Specialized equipment such as the welders' masks and eyeglass fit masks also meet the required approvals.

A medical evaluation of all respirator users was performed either by a screening group which came to the site semi-annually or otherwise by a local physician. The medical form and tests required to be passed by the screening group and the local physician were not included in the health physics personnel file of the workers. The auditor noted that Regulatory Guide 8.15 requires in Section 4.g that records sufficient to evaluate program adequacy be maintained.

The licensee representative stated that he had informed the local physical via telephone of the respiratory physical requirements, and that the screening medical group had claimed that they reviewed all of their requirements to assure that they were in accord with Regulatory Guide 8.15.

The licensee utilized file cards to indicate whether or not a wearer was qualified to wear respiratory protection. If the person who wrote RWP's or the leadman found that the person was not qualified, he removed the name from the list until he had been retrained and requalified. Such file cards also existed for annual SCBA retraining and for physical examinations for respiratory protection.

Procedures indicated that personnel requiring respiratory protection equipment would obtain the equipment before going to the job site. The appraiser observed that there were no provisions to preclude an unauthorized user from picking up respiratory equipment and using it. The equipment was stored in open bins and was readily accessible to all passers-by. There was no requirement to examine the authorization card after equipment pick-up, nor was the user required to be certified as the person for whom authorization had been extended. The auditor noted the Regulatory Guide 8.15 as referenced in 10 CFR 20.103(c), requires in regulatory position C.4.e that written operational and administrative procedures be established for control and issuance of respiratory protective equipment.

The auditor found that no quantitative respirator fit testing is done prior to an individual's use of respiratory protection equipment. A qualitative fit test performed with isoamyl acetate (banana oil) was used at one time, however, this was discontinued during 1979. The only fit testing at the time of the appraisal was qualitative fit testing for airline and SCBA respiratory protection equipment through the use of isoamyl acetate. Negative pressure fit checks were reportedly done but no check data was recorded. Data on the isoamyl acetate tests discussed above were not consistently recorded or attached to RWP's.



The auditor noted that reliance on the individual being tested to respond when he smelled banana oil was questionable, since the lack of a satisfactory respirator fit directly affected the individual's ability to earn his pay. The auditor noted that qualitative fit testing is a limited test method and is not considered appropriate for the purposes of assigning protection factors for use of respiratory protective equipment.

Based on the above findings, improvements in the following areas are needed to achieve an acceptable program:

- The establishment of an adequate control and issuance program for respiratory protective equipment. (50-333/80-20-10)
- The institution of quantitative fit testing and certification of the respiratory protection devices each effected worker is authorized to use prior to the use of protection factors in estimating his exposures to airborne radioactive material. (50-333/80-20-11)

Based on the findings in the above area, the following matters should be considered for improvement of the program:

- Maintain all records relative to personnel qualification for use of respirator equipment, including evidence of respiratory physical in one central location.

#### 3.2.5.5 Respiratory Protective Equipment Training

The appraiser reviewed the respiratory protection training program and found that the training, did not address the breath nor depth of subject matter required to be covered by Regulatory Guide 8.15 and NUREG-0041.

A licensee representative in the Training Department indicated that because of lack of personnel, the respiratory protection training had been tailored such that only the material deemed most essential for a wearer to know was covered. During the general employee training, the subject of respiratory protection was covered and equipment held

up and exhibited, however, no wearer training was performed in either the general employee training or the portion of training referred to as respiratory protection training. Some references were made to respiratory protection equipment wearing, donning, etc., in various commercially available videotapes which dealt with the general radiation protection program training.

The respirator wearers were, according to procedures, trained with use of a video tape and lecture, with subsequent instruction by their Radiation Work Permit (RWP) leadman for specifics such as visual communications when wearing respiratory protective equipment. When considerable voice communication was necessary, respiratory protection equipment was to be fitted with a commercially available speaking box.

Breathing resistance and adequacy of air supply was not covered either in station procedures or in the video training.

The RWP leadman was assigned the responsibility for the review of the use of respiratory protection equipment with the personnel working on the RWP with him. Where airline respirators were to be used, isoamyl acetate (banana oil) tests were to be performed to insure an adequate fit prior to entry into the airborne radioactive area. These tests were to be recorded on a form which was attached to the RWP. It was the responsibility of the leadman to prevent personnel who had facial hair which would interfere with the seal from using respirators. It should be noted that the only fit test performed for air filtration types of respirators is a negative pressure test. As previously noted, no quantitative fit test was performed.

Drills while utilizing respiratory protection equipment for the purposes of permitting respirator wearers to become experienced in donning, using, and removing such equipment prior to actual use, as described in NUREG-0041 had not been conducted by the training department since 1979.



The retraining frequency for personnel utilizing respiratory protection was not clearly specified in the Radiation Protection Operating Procedures. The material to be covered during retraining was not delineated in the respiratory protection training lesson plan nor Indoctrination and Training Procedure No. 14.

The auditor reviewed the qualifications of the Director of Training and noted that he had attended several U.S. Navy schools which involved the use of self-contained breathing apparatus. He also had been through several fire training schools in which self-contained breathing apparatus training was required. The Director maintains a knowledge of the respiratory protection requirements through reading. He has never been formally trained by any of the major manufacturers of respiratory protection equipment nor has he been through one of the national laboratory schools in respiratory protection. The RESS and the assistant to the RESS, on the otherhand, both received training at Los Alamos Scientific Laboratory (LASL) and had received subsequent SCBA retraining at the Niagara Mohawk Fire Training School.

Based on the findings in the above area, improvements in the following areas are needed to achieve an acceptable program.

- The establishment of a comprehensive quantitative respiratory protective equipment training program and procedures that meet the requirements of item C.4.b of Regulatory Guide 8.15. (50-333/80-20-12)

#### 3.2.5.6 Respiratory Equipment Maintenance and Quality Assurance Program

Respiratory protection equipment was presumed to be contaminated upon removal from an RWP area. No pre-decontamination survey as through the use of smears was performed on the equipment. All equipment was assumed to be contaminated and was decontaminated in the same fashion.

Radiological contamination limits were established for the reuse of equipment by procedure, and surveys, following decontamination and prior to

repair and inspection, were performed in a secondary laboratory section within the chemistry laboratory area.

The practices used for cleaning and disinfecting the respirators were observed and appeared to be acceptable. Adequate care was taken to prevent damage to the equipment during the cleaning and disinfecting process. The auditor noted the turn around time for issuance, use, decontamination and return for re-use to be adequate to maintain supply. Licensee representative indicated they could obtain respirators from Nine Mile Point if needed.

After decontamination, respirator repair, if needed, was performed by a person trained on the job by RESS personnel who had received specific training from various respirator representatives in the field.

Radiological and Environmental Services Departmental internal quality assurance as it pertained to respiratory protection, seemed to be lacking. No procedures existed for review of data obtained by air sampling equipment, or the quality assurance testing of reissued equipment to verify that it would perform in accordance with the manufacturer's specifications. The onsite quality assurance group only audited procedural adherence and did not review the program for adequacy.

Approved replacement parts for respirator equipment were obtained from the manufacturer of the equipment, however, there were no provisions to verify upon receipt, that the new equipment was, in fact, acceptable. A representative of the quality assurance department onsite indicated that since the class of the purchase was not one which they were required to review no audit was performed on such purchases. The auditor also noted that there appeared to be no systematic inventory of other than emergency respiratory devices. Chapter 10 of NUREG 0045, as referenced in Regulatory Guide 8.15, regulatory positions C.4.e, and f requires written operational and administrative procedures for control, inspection issuance, proper use and return.

The regulators and warning devices for the self-contained breathing apparatus were tested by donning the unit and breath testing them. Airline supply regulators were neither tested when initially received nor after decontamination following use. No records were maintained on air filtration or airline respirators. The units were not serialized and immediate identification of a specific unit was not practical under the present circumstances. As a consequence, changes of face plates, inlet valves or exhaust valves within a particular unit were not noted. Tests of the leakage subsequent to these changes were neither performed nor recorded. The auditors noted that failure to perform such tests was not in accordance with Regulatory Guide 8.15. Regulatory Guide 8.15 requires in Section C.4.d that procedures be established for inspection of respiratory protective equipment.

Regarding self contained and air supplied respiratory protective equipment, the auditor noted that the Radiation Protection Operating Procedures (RPOP) called for the use of self contained or supplied air for specialized circumstances. The station had a breathing air supply (Service Air) which was described in the Operating Procedure F-OP-39, "Breathing Instrument and Service Air System." The fittings and components for breathing air were standardized and unique, thereby, precluding the inadvertent introduction of other gases or depletion of air supply through such means as the use of air-driven tools.

The compressors used for the breathing air system had teflon seals and had been isolated from various air systems so as to preclude an introduction of oil, unburned hydrocarbons, and carbon monoxide from oil breakdown. The air compressors were monitored for carbon monoxide, oil vapor and other contaminants including radiologicals. The air was specified as Grade D breathing quality. A pressure activated isolation valve closes off the air line from the compressor to the breathing air hold-up tank. Alarms attached to the isolation valve alerted the control room. The operators were instructed, by procedures, to announce that breathing air systems had failed and all persons using airline respirators were to evacuate to a noncontaminated

area upon such an announcement. Compressed gas cylinders for refilling self-contained breathing apparatus bottles were appropriately labeled as containing breathing air.

According to licensee procedures, periodic equipment testing and inspection was supposed to be performed on a monthly basis for the self-contained breathing apparatus. Records were maintained for such inspections. Inspections were performed in two parts, one which was the control room breathing air, and the second part which was the station and emergency pack SCBA's. It was noted by the auditor that inspections of the control room breathing air, which were supposed to be performed monthly were, according to licensee inspection records, only performed in February, May, July, August and November of 1980. Inspections of the station and emergency units were performed on a monthly basis as evidenced by inspection of the records.

In a review of the compressed station air and supplied air manifold systems, it was noted that the manifold outlet valves were typically not covered to preclude the introduction of foreign matter, dirt, oil or other potentially hazardous materials. The auditor noted that this should be rectified in future maintenance programs at the facility.

During a review of the breathing air supply system, it was noted that the portable manifolds, which are linked back to a bulkhead-mounted distribution point, contained filter units. These filter cartridges were capable of particulate removal, moisture removal, and removal of several other components because of the charcoal filtration unit. No installation data nor test data was noted on the cartridge cannister within the field manifold. Nor was there an installation noted on the filter unit in the cartridge at the bulk-head mounted outlet. It was recommended to facility personnel that procedures be implemented to assure an indication of the last date of installation of a filter.

Licensee representatives indicated that since filter units were mounted on the bulk-head, cartridges would probably be removed from the portable distribution field manifolds. It was indicated that, in all probability, the empty cannisters would be labeled as not containing a cartridge, inasmuch as filtration was available at the last hard plumbed area coming away from the bulk-head.

During a review of the air filtration type respirators used by the licensee, it was noted that that licensee personnel reused a cartridge up to four times, marking it with an X or some other delineating mark, each time it was removed from a mask for decontamination. However, the licensee was not performing pressure drop or other filtration quality tests prior to the reuse of cartridges. The auditor noted that such tests should be performed in order to assure that cartridges have a reasonably lengthy life before reuse, as discussed in NUREG-0041.

Airline respirators and air filtration type respirators were not adequately stored. Respirators were bulk-loaded into top loading tool cribs such that units may be stacked as many as 15 deep. Storage in this manner was not in conformance with Chapter 9 of NUREG 0041, referred to in Regulatory Guide 8.15, Section C.4.d, as it could not prevent damage or misshaping by adjacent equipment pressure.

Protection of respiratory protection equipment against heat, cold, sunlight, or moisture was provided through the retention of the equipment in fairly well-controlled environments, the prevention of exposure to sunlight and the protection of the units prior to use through the sealing of them in plastic bags.

Based on the findings in the above area, improvements in the following areas are needed to achieve an acceptable program:

- The establishment of an adequate program or procedures for respiratory protective equipment testing, storage, issuance and control that meet the requirements of item c.4.c, d and e of Regulatory Guide 8.15 (50-333/80-20-12).



### 3.2.5.7 Engineering Controls

Licensee representatives indicated that gloveboxes, hoods and tents were being employed where practicable to contain potential airborne radioactivity. The auditor noted that the use of such tents and containment systems did not seem to be consistently considered. This is evidenced by the findings discussed later in this section.

The licensee utilized continuous air monitors as trending devices with alarm points set at sufficiently low levels to indicate a change in the ambient airborne radioactivity and cause radiation protection surveillance to be increased.

The auditor reviewed the designated air ventilation and cooling systems in order to establish that air flows were from areas of low to potentially high airborne radioactivity. Supporting documents reviewed were the Plant Operating Procedures F-OP-51, 52, 53, 54 and 70. No apparent problems were found with the ventilation design and operation as described.

A review of measurements of the face velocities of the sampling hoods, chemistry fume hoods, and other inlet air requirements as it pertained to airborne radioactivity control was conducted. The radiation protection group produced a record of survey performed in October 1980, which indicated that face velocities, as measured with a velometer, were above the minimum requirements.

Based on discussions with licensee representatives and inspection of work locations, the limitation of exposures to airborne radioactive materials through the use of engineering controls (e.g., auxiliary ventilation systems) appeared to be limited. For example, work involving sawing contaminated equipment (radwaste filter septums) was performed inside of a tent, but the tent was not equipped with exhaust ventilation. This caused one entire elevation of the radwaste building to become an airborne radioactivity area and the licensee required the use of respirators. The airborne radioactivity concentrations generated were measured to be 32% of the 10 CFR 20 Appendix B value inside the enclosure and 29% of it outside (Survey #35047, November 10, 1980). This serves to illustrate a situation where the



use of auxiliary ventilation would have prevented the elevated concentration outside of the enclosure. For this particular job, air line respiratory protection was worn inside the enclosure and filter respirators were worn outside. It was noted by the auditor that several maintenance tasks performed during the 1980 refueling outage (May-August) involved significant airborne radioactive material concentrations and might have benefitted from the use of auxiliary ventilation systems. As an example, maintenance work performed in the drywell which included repair of various valves, such as packing, removal and grinding of welds, (RWP Nos. 1766, 1858, 2441, 2449 and 2537) involved exposure to ambient airborne radioactive material concentrations ranging from 100 to 2000 times the 10 CFR 20, Appendix B values. Although respiratory protection was worn in all instances, it appeared to the auditor that the use of auxiliary ventilation systems with HEPA filtration had not been sufficiently considered.

Licensee representatives indicated that some large (4,000 CFM) temporary ventilation systems were available and could be used with flexible duct work to ventilate areas to reduce the concentrations of airborne radioactivity and hence, reduce the need for respiratory protection equipment during outage conditions. One such portable ventilation system was found in the facility during tours but was not seen in use.

The auditor selected several radiation work permits from the licensee's files to review for various aspects including the use of respiratory protection equipment, air sampling, nasal smears, maximum permissible concentration evaluations and the general use of Engineering Controls. Radiation work permits selected for review were for work performed on or about June 21 and June 22, 1980, (RWP's 3054(S), 3064(S), 3074(S), 3120(S), and 3123(S)), and on or about August 4, 1980, (RWP's Nos. 6015(S) and 5982(S)). The RWP selected for complete follow-through was 3123(S), which covered work performed on June 22, 1980 at 1630 hours.

Work was conducted in the drywell and was for packing removal on 12 MOV 15 and for surveys in that area to establish the airborne contamination

levels. The surveyor was in the area with an air filtration type mask for approximately 10 minutes. The two persons who worked under the permit were listed as wearing airline respirators and had been in the area for 2 to 2.5 hours. An air sample was commenced on the job at 1625 and terminated at 1752. The sample was taken with a high volume air sampler through a charcoal and filter combination cannister and the cannister was counted with a GeLi spectrometer. The associated computer program calculated the MPC % to be 7.0352 million percent (that, is the concentration was 70,352 times the appropriate value from 10 CFR 20 Appendix B Table 1, Column 1.) Protection factor utilized for the airline respirators was 2000. The effective air concentration (ambient concentration divided by the protection factor) was 35.176 times the applicable 10 CFR 20, Appendix B concentration and an individual using an airline respirator could remain in this concentration for 1.7 minutes before exceeding the 40 hour control measure referred to in 10 CFR 20.103(b)(2) and 22 minutes before exceeding the maximum permissible intake specified in 10 CFR 20.103(a)(1).

The maximum protection factor authorized for full face air purifying respirators is 50. Consequently  $70,350 \div 50 = 1407$  MPC-hours and an exposure for 2.5 seconds would constitute the maximum exposure time before the 40 hour control measure was exceeded and 33.3 seconds would be the maximum exposure time for an individual so protected before the maximum permissible intake specified by 10 CFR 20.103 was exceeded.

The auditor noted that 10 CFR 20.103(c) requires that those licensees making allowance for respiratory protection equipment use such equipment in accordance with Regulatory Guide 8.15. Regulatory Guide 8.15, specifies in Section C.2 that respiratory protective equipment be selected must provide a protection factor greater than the multiple by which peak concentrations are expected to exceed the values specified in 10 CFR 20, Appendix B. The peak concentration of airborne radioactivity in the above case was 70,352 times the applicable Appendix B values while the respiratory protective equipment used only provided a protection factor of 2000, i.e., the device's protection factor was about  $1/35$  of the peak concentration. The

auditor noted the use of a device with a lower protection factor than the peak airborne concentration was not consistent with the Regulatory Guide requirements.

Nasal smears were performed on the two workers and the surveyor. Results of which for the two workers with airline, atmosphere supplying respirators) were 151 and 328 dpm and for the surveyor with the air purifying respirator the results was 1236 dpm.

Section C.2 of Regulatory Guide 8.15 states, in part;

"If a respirator user's intake of radioactive materials is later determined by other measurements to have been greater than that expected from initial estimates of radioactive materials in the air the user inhales, the greater quantity is to be used in evaluating exposures; if it is less than that initially estimated, the lesser quantity may be used in evaluating exposures"

The three persons were sent to the whole body counter to obtain data to estimate the quantity of radioactive material inhaled. Results of these whole body counts ranged from 0.36-1.7% of the MPOB. The lung burden of both workers had diminished to less than 10% of the original value within a short period of time, and the surveyor's lung burden was reduced by a factor of 2 in a period of two months.

As shown previously in Table 3.1, 1.7% of MPOB for cesium 137 indicates that an intake in excess of the 40 hour control measure of 10 CFR 20.103(b)(2) has occurred. In such an instance the regulation requires that the licensee shall,

"make such evaluations and take such actions as are necessary to assure against recurrence. The licensee shall maintain records of such occurrences, evaluations, and actions taken in a clear and readily identifiable form suitable for summary review and evaluation."

No form was available to indicate that the personnel received a banana oil qualitative test of respirator fit, although the RWP indicated

that such a test was required. During a review of the training, the appraiser learned that although the leadman was supposed to consider engineering controls before working in areas having potentially high airborne radioactivity, no steps were taken to prevent airborne activity from being blown out into the work environment. Specifically, no tenting or containment type devices were utilized prior to applying an air stream to remove the packing, nor did procedures prohibit the use of an air stream to remove the packing. The auditor noted the pressurized air stream used apparently blew radioactive contamination into workers breathing zone.

10 CFR 20.103(b)(1) states,

"The licensee shall, as a precautionary procedure, use process or other engineering controls, to the extent practicable, to limit concentrations of radioactive materials in air to levels below those which delimit an airborne radioactivity area as defined in 10 CFR 20.203(d)(1) (ii)."

The appropriate portion of 10 CFR 20.203(d) states,

"As used in the regulations in this part, "airborne radioactivity area" means ... any room, enclosure, or operating area in which airborne radioactive material composed wholly or partly of licensed material exists in concentrations which, averaged over the number of hours in any week during which individuals are in the area, exceed 25 percent of the amounts specified in Appendix B, Table I, Column 1 of this part.

The auditor noted that a process control such as a procedural prohibition against directing an airstream against the contaminated dirt and packing material was a practicable precautionary procedure that could have been used in this case.

The auditor noted that the use of the air stream produced a concentration of radioactive material which, assuming an occupancy time of 2.5 hours, exceeded the value specified by 10 CFR 20.203(d)(1)(ii) by a factor of 17,588.

The above incident and subsequent exposures were recorded in the workers' files as it pertained to their MPC-hour exposure. The check of the records indicated that the users were qualified to utilize airline respirators. In a review of the data utilized to establish the protection factors of equipment, it was noted that there was an inconsistency between the respiratory protection lesson plan and the Radiation Protection Operating Procedure values for protection factors. Protection factors in the RPOP were consistent with those recommended by Regulatory Guide 8.15.

Based on the findings in the above area, improvements in the following areas are needed to achieve an acceptable program:

- Establish means to ensure respiratory protective equipment is not routinely used in airborne radioactively concentrations whose MPC fraction is greater than the protection factor of the equipment. (50-333/80-20-13)
- Establish a program and procedures to assure that process or other engineering controls are used to the extent practicable to limit the concentrations of airborne radioactive materials. (50-333/80-20-14)

### 3.2.6 Internal Dosimetry Quality Assurance Program

The licensee's quality assurance program for the internal dosimetry program (whole body counting) was reviewed with respect to the program recommendations specified in ANSI N-343, "Standard for Internal Dosimetry for Mixed Fission and Activation Products." Procedure CRI-6 described the Quality Assurance Program for the whole body counter.

Review of the whole body counter operation indicated the following operational and quality control checks were performed.

- A daily five minute background count. Normal background count rates for each detector were sufficiently familiar to the operator so that off-standard conditions were easily recognized.
- A short (one-half minute or less) count of either Ba-133/Co-60 button sources ( $\sim 0.6 \mu\text{Ci}$ ) or a Na-22 liquid source ( $\sim 0.6 \mu\text{Ci}$ ) to ensure that the photopeaks fell in (or sufficiently near) the proper channels.

- An annual primary calibration, using NBS-traceable sources, for nine radionuclides of reactor origin and K-40. The WBC vendor performed the initial calibration in 1978, as well as subsequent calibration in 1979 and 1980. The licensee, however, did not have any record of these calibrations. (The vendor did not furnish a report for the 1978 and 1979 calibrations; the 1980 calibration had just been completed by the vendor prior to the appraisal and a report had not yet been furnished). The appraiser was unable to complete a review of this area.

The above operational and quality control checks did not, however, include any counts of check sources on a frequent basis (e.g., daily or weekly) to ensure that the analyzer was quantifying activity on a consistent basis (e.g., verifying that a given number of counts result under selected photo peaks). This was noted to be inconsistent with ANSI N343, Section 15.3.3. In addition, review of Procedure GRI-6 indicated that the annual primary calibration referred to a single activity level for each radionuclide, rather than a range of activity between 60 and 20,000 nCi, as recommended by ANSI N343 Section 15.2.

Based on the above findings, improvement in the following area is needed to achieve an acceptable program:

- The establishment of a whole body counting calibration and QA check program meeting the recommendations of ANSI N343. (50-333/80-20-15)

### 3.3. Surveillance Program

#### 3.3.1 Documents Reviewed

- a. "Radiation Protection Operating Procedures," Part A, Revisions 0-4, June 1978, December 1979
- b. RTP-1, "Area Radiation Monitor Calibration," Revision 1, June 1979
- c. RTP-3, "Victoreen Model 740F Operation and Calibration," Revision 1, January 1979
- d. RTP-4, "Teletector Operation and Calibration," Revision 1, January 1979
- e. RTP-6, "E-120 Operation and Calibration," Revision 1, August 1978



- f. RTP-12, "Sealed Source Leakage Test," Revision 0, November 1978
- g. RTP-16, "RM-16 Operation and Calibration," Revision 0, January 1979
- h. RTP-17, "PNR-4 Operation and Calibration," Revision 0, January 1978
- i. RTP-18, "PNC-4 Operation and Calibration," Revision 0, January 1978
- j. RTP-19, "Eberline Model RO-5A Operation and Calibration," Revision 0, January 1980
- k. RTP-20, "RM-14 Operation and Calibration," Revision 2, May 1980
- l. RTP-23, "High Volume Portable Air Sampler," Revision 2, April 1980
- m. RTP-24, "Model 1000B Calibrator Operation," Revision 1, October 1980
- n. RTP-27, "Digi/Master Operation and Calibration," Revision 1, October 1979
- o. RTP-28, "E-520 Operation and Calibration," Revision 1, October 1979
- p. F-OP-32, "Area Radiation Monitoring System No. 18," Revision 1, September 1980
- q. Plant Standing Order No. 17, "Use of Digimaster Survey Meter," Revision 0, January 1979
- r. RES Department Standing Order No. 2, "Maintenance of Radiation Protection Records," Revision 0, April 1977
- s. RES Department Standing Order No. 8, "Routine Plant Patrols," Revision 0, October 1979
- t. "Air Sample Log," April - November 1980
- u. "Radiation Survey Log," April - November 1980
- v. "Instrument Calibration Log," January 1979 - November 1980

- w. "Radiation Work Permit Log," May - November 1980
- x. Memo from Eric Mulcahey to RES Techs dated August 21, 1979, subject, "Surveillance Requirements" (RES-79-103)

### 3.3.2 General

The documentation of routine radiation surveillance program requirements was somewhat fragmented, i.e., some were contained in the August 21, 1979 memo to Radiological and Environmental Services (RES) Techs (referenced above) and some were referenced in the Radiation Protection Operating Procedure. The latter document appeared, however, to have been written primarily for those individuals qualified to perform self-monitoring.

Self-monitoring was practiced routinely at the FitzPatrick plant. Such monitoring was permitted after an individual had received the appropriate training and was listed on the extended Radiation Work Permit (RWP) for the individual's department. Such RWPs were for inspection activities, observation of work, and other similar activities which did not involve maintenance work on components or equipment where the dose rate exceeded 5 mrem/hr. Individuals qualified in self-monitoring were permitted to measure dose rates (gamma and/or beta) up to 10 rem/hr and monitor surface contamination levels up to 50,000 dpm per 100 cm<sup>2</sup>. The auditor did not note any problems with the use of this monitoring methodology.

Responsibilities for various facets of the surveillance program appeared to be defined in an August 21, 1979 memo from the RES Department Superintendent to the RES Technicians. Licensee representatives indicated the memo was reissued approximately every six months to reflect changes in assignments.

It appeared to the auditor that this system for personnel assignment to the surveillance program was not sufficiently formal, i.e., changes in assignments had been made since August 21, 1979, but were apparently oral, rather than reflected in an updating of the memo. Definitions of responsibilities for the several contract radiation protection technicians on duty during the appraisal were also informal. These contract technicians were given responsibility for most of the routine (non-radiation work permit related) surveillance activities. These included surface contamination surveys of clean areas, step-off pads, and tools and other equipment being removed from contamination or potentially contaminated areas; routine airborne radioactivity surveys; and sharing of the surveillance workload in support of

radiation work permit, (RWP), issuance and followup. Complex surveillance tasks were generally performed by RES technicians. Direction and first-line supervision for surveillance activities for all technicians were normally provided by the Senior Technician, Radiation Protection. This individual was responsible for scheduling of routine surveillance activities, assignment of technicians to perform surveys in support of the RWP program, and generally, (53% of the time, based on a sampling of 272 RWPs), was the individual who signed the RWP for the RES Department.

Technical procedural guidance for the performance of surveys (as opposed to guidance relative to operating an instrument) appeared to be contained primarily in the RPOP. These procedures appeared to be directed toward the individual qualified to perform self-monitoring and not towards a radiation protection technician. Thus, specific types of surveys which self-monitors, as a rule, did not perform, e.g., neutron dose rate measurements and air sample collection, were not discussed in any detail in the RPOP.

Procedural guidance for the RES technicians for the performance of these types of surveys, in particular, were lacking. In addition, the RES technicians had little procedural guidance regarding the specific surveys to be performed for an RWP.

Records of surveys appeared, in general, to be adequate, although a number of instances were noted where a sketch of the area surveyed would have been of more benefit to the user of the survey results than the manner in which the data were presented. The Survey Log was a collection of all types of surveys performed arranged in order of survey number (approximately chronological). The auditor concluded that some benefit might be realized by segregating the surveys by building and then further by level, such that routine and even certain non-routine surveys could be recorded on a building map for the particular elevation surveyed. A category of special surveys, for example, could also be established for such measurements as nasal swabs and other contamination surveys of individuals.

Surveys performed by the licensee for both routine and non-routine situations included direct radiation, airborne radioactive materials, and surface contamination measurements. All surveys were performed either by licensee personnel or by contractor HP technicians.

Licensee representatives indicated that alpha measurements were made on approximately 10% of all surface contamination measurements (smear surveys) and on 100% of smear surveys of spent fuel and waste shipments. However, a random sampling by the auditor, of approximately 140 surveys disclosed only one alpha measurement (associated with a waste shipment). Although a goal of performing alpha measurements on 10% of all smear surveys may be more than normal, likewise not performing these surveys any more frequently than was disclosed by the auditor's sampling of records was noted to be too infrequent.

Direct beta radiation surveys were normally performed using ion chambers essentially at contact with the source. Source geometry correction factors of approximately 3 for one model of instrument and from 6 to 12 for another instrument were posted on the case of each instrument. The only major drawback with the licensee's program for beta dose rate measurements was that they were not always performed in those situations for which such measurements were warranted. The appraiser noted several instances during the 1980 refueling outage where beta surveys did not appear to have been performed: RWP #423(S), tear down of "D" MSIV, 5/12/80; RWP #3201(S), control rod drive disassembly, 6/24/80; and RWP #3346, open/repair valve located in steam tunnel, 6/25/80. Increased management oversight appears needed in this area.

Gamma radiation surveys were made with the same ion chambers used for beta measurements. No significant problems were noted in this area.

Neutron radiation surveys were normally performed with a commercially available rem-meter. However, during the appraisal, it was noted that the licensee's only rem-meter was at the manufacturer's facility undergoing repair. Neutron surveys performed during this period were accomplished with a moderated instrument which did not directly indicate a dose rate in rems/hr, rather it indicated a count rate due to high energy neutrons and another due to thermal neutrons. The dose rate was then conservatively estimated from these count rates. A licensee representative recognized that the backup neutron instrument was not totally adequate, and stated that a second rem-meter was on order.

Airborne radioactive materials were collected both from individual's breathing zones and from ambient air at several key locations around the facility. Both particulates and radioiodines were measured. All samples were counted on a GeLi/multi-channel analyzer system. The auditor noted the lack of a procedure for operation and

calibration of the Staplex air samplers used to collect breathing zone air samples. The licensee's air sampling program did not include lapel samplers, reportedly on the basis that the small flow rate associated with such units would not provide a sufficiently low minimum detectable activity. The auditor noted that lapel air samplers may be the preferred method for situations where high concentrations are anticipated, when concentrations are subject to significant changes, or when other means for obtaining a breathing zone sample are difficult to employ.

The auditors noted that Technical Specification 6.8, indicates that written procedures are to be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5, "Facility Administrative Policies and Procedures", of ANSI 18.7-1972 and Appendix A of Regulatory Guide 1.33, November 1972." Section G of Appendix A to Regulatory Guide 1.33, 1972 lists procedures for limiting material released to the environment and limiting personnel exposure. Section G.5 recommends procedures for surveys and monitoring.

Section H of Appendix A recommends procedures of a type appropriate to the circumstances be provided to assure that among other items, instruments and other measuring devices are properly calibrated and adjusted at specified periods to maintain accuracy. The auditors noted that the lack of operation and calibration procedures for the Staplex air sampler, an instrument used for limiting personnel exposure to airborne radioactive material was not consistent with the Regulatory Guide recommendation.

Surface contamination measurements (smear surveys) were primarily made by using kraft paper discs approximately 1" in diameter. The smears were then counted with a pancake probe GM counter.

Other than alpha contamination measurements and the use of kraft paper, (which is discussed later), the overall surface contamination measurements program appeared to be adequate, including smear surveys of contaminated areas in support of the RWP program, smears of areas outside of those where radioactive materials were handled/stored, and smears of step-off pads. The auditor noted that materials, tools and other equipment which had been either carried through or used in contaminated or potentially contaminated areas were routinely smeared at the controlled area access point prior to exiting the clean area.



As part of the independent measurements performed during this appraisal, a comparison of smear survey methods was made between kraft paper (routinely used by the licensee) and filter paper. The results indicated that on the average surface contamination levels were a factor of two higher when filter paper is used. The kraft paper-quantified activity was used as a basis in setting protective clothing and other contamination control requirements. However, in all cases where kraft paper was used the procedural requirements were set at very low levels and levels twice as high would not create a contamination problem. The use of kraft paper for this purpose was, therefore, satisfactory. The use of kraft paper where greater accuracy is required, such as contamination surveys of radioactive waste shipments, may result in the introduction of differences between sender/receiver contamination values and the possibility of exceeding contamination limits.

Based on the above findings, improvements in the following areas are needed to achieve an acceptable program:  
(50-333/80-20-16)

- The establishment of a formally documented and approved routine plant radiation and contamination surveillance program.
- The establishment of formal guidance for technicians as to the type of radiation surveys required prior to issuance of radiation work permits.

Based on the above findings, the following matters should be considered for improvement:

- Use of an improved smear survey medium.
- Segregation of routine radiation and contamination survey results.

### 3.3.3 Instrument Suitability and Use

#### 3.3.3.1 Portable Radiation Monitoring Instruments

The licensee appeared to have a sufficient number and appropriate types of instruments available for performing measurements of beta and gamma radiation levels. Two types of portable GM counters and two types of ion chambers were used for most of the measurements made as part of the routine program. These instruments were frequently supplemented by a compact GM-type instrument which provided a digital indication



of the gamma exposure rate. High range (up to 1000 R/hr) instruments having a telescoping probe were also in the licensee's inventory. The licensee's procedure for tagging inoperative instruments and initiating a work request appeared satisfactory.

Although the high range portable instrumentation (discussed above) provided a high enough range for routine use, the instruments may not be adequate for use in accident situations where portable instruments up to 10E4 rad/hr are recommended to be on-hand as specified in ANSI N320, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation."

Calibrations of the above instruments were performed quarterly with a commercially available calibrator containing Cs-137 sources having a total activity of 186 curies. The auditor evaluated the licensee's ability to properly use the instrument calibrator by requesting a representative of the licensee to expose the auditor's instrument (an ion chamber which had previously been calibrated with an NBS-traceable source) to several specific exposure rates. The results were as follows and appeared satisfactory:

<u>Calculated exposure rate, mR/hr</u>	<u>Actual exposure rate, mR/hr</u>
20	17
220	225
650	700
3700	3800

Calibrations of the licensee's neutron survey instruments were performed annually by the manufacturer. An operational check source for beta-gamma instruments was available near the Health Physics control point and appeared to be routinely used.

Alpha monitoring, discussed earlier from the aspect of smear counting, should be supplemented by a direct measuring alpha survey meter. The licensee currently does not own such an instrument. The Appraisal Team recommended that consideration be given to purchase such an instrument.

With respect to area monitoring instrumentation, the appraiser noted that the licensee employed both a fixed system (original equipment having 30 monitoring points) and several semi-fixed monitors. While no problems were noted with the use of either of these systems, the Appraisal Team informed the licensee that the upper limit of the fixed system (maximum of 1 R/hr, except for monitor No. 30, which had an upper limit of 10E3 r/hr) was much less than the 10E4 R/hr recommended by ANSI N320-1979. A licensee representative stated that he did not believe this system was to be designed for use in an emergency.

Based on the above findings, the following matter should be considered for improvement:

- The evaluation of the current supply of portable radiation survey instrumentation in light of the recommendations of ANSI N320 and the selection of additional instrumentation as needed.

#### 3.3.3.2 Airborne Radioactivity Sampling Instrumentation

The air sampling/monitoring equipment used by the licensee consisted of continuous air monitors (CAMs), portable low flow rate air samplers, and portable high flow rate air samplers. The number of each type appeared to be adequate.

Breathing zone samples were collected with a high flow rate (5 cfm with a charcoal cartridge) air sampler. The charcoal cartridges used in conjunction with this air sampler were approximately 2 1/2" in diameter and about 3/4" thick and were threaded so that they would fit into the ring adapter placed over the intake of the air mover. The particular charcoal cannister used had been on the market for many years. The auditor computed a "rough" estimate of the residence time of iodine in the air being sampled and the charcoal cartridge (based on a flow rate of 5 cfm, an effective area of 1.3 sq. in., and an effective thickness of 1 in.) and determined it to be about 10 msec. This value was much smaller than the 250 msec per 2 inches of bed depth recommended by Regulatory Guide 1.52. The Appraisal Team concluded that the type of charcoal used by the licensee was not the type that could

be used with a high flow rate air mover, and therefore the licensee could be underestimating airborne iodine concentrations. A licensee representative stated that an order had recently been placed for 10 high flow rate air movers and charcoal cartridges compatible with a high velocity air stream, as well as silver zeolite cartridges (P. O. #20785-80, dated 11/12/80). No other problems were identified with the licensee's air monitoring program.

Based on the above findings, improvements in the following area is needed to achieve an acceptable program:

- Purchase and use of sampling equipment and media with known airborne radioiodine sampling, collection and retention efficiencies. (50-333/80-20-17)

#### 3.3.3.3 Personnel Contamination Instrumentation

The licensee's program for personnel contamination monitoring addressed all of the usual monitoring devices available, viz., portal monitors, hand and shoe monitors, and friskers. One item noted by the Appraisal Team was that the only type of these devices which was both operating and being properly used by personnel was the hand and shoe monitor (and even this system, in one case - the exit to the dosimetry trailer - was inoperable).

A commercially available portal monitor device was installed at the main guard house - a location through which all personnel must pass upon leaving the plant. This system, however, was not operating at the time of the appraisal. Licensee representatives stated that it was both inconvenient to use and lacked the sensitivity judged by the license to be adequate. Upgrading of this system has been in progress since at least August 21, 1979 when a memo from the RES Department Superintendent assigned the responsibility of upgrading the system to a senior technician. However, little appears to have been done as of the time of the appraisal (approximately one year later). There was a monitoring device in operation at the guard house, however, it was not suitable for personnel monitoring and was admittedly used for safeguards purposes rather than contamination monitoring. Licensee represen-

tatives stated that a liquid scintillation detector was being considered as a replacement for the existing portal monitor.

The auditor also reviewed the licensee's system and procedure for use of frisking devices. Although the system appeared adequate in terms of number of units and locations, direct observation of frisking techniques by the auditor revealed them to be inadequate. Specifically, of 50 individuals observed exiting the HP control point during the period covered by this appraisal, only two individuals surveyed areas of their body beyond their hands and shoes. The 2 individuals still did not perform a complete frisk. While no contamination was detected by any of the fifty individuals, the practice of not surveying beyond the hands and shoes increases the likelihood of personnel contamination being carried offsite - a problem exacerbated by the non-operating portal monitor. Improvements in this system, such as locating additional friskers near the HP control point and setting up frisking lanes so that several individuals could perform whole body frisks simultaneously, were discussed with the licensee. Additionally, the auditors indicated that Technical Specification (T.S.) 6.11 requires that procedures for personnel radiation protection be prepared and adhered to for all plant operations. The licensee's Radiation Protection Operating Procedures developed in accordance with T.S. 6.11 requires in Section III.C.1 that a complete contamination survey of the body be performed by passing the probe slowly over the body. The licensee representatives were noted to direct the radiation protection technician at the control point area to observe personnel and ensure a complete body frisk is performed.

In further reviewing the area of personnel contamination monitoring, the licensee was noted to be utilizing thick wall ( $\sim 30 \text{ mg/cm}^2$ ) GM tubes for this purpose. The use of the thick wall tubes would preclude identification of low-level skin contamination. The auditors noted thin wall ( $\sim 7 \text{ mg/cm}^2$ ) GM tubes were available for low-level skin contamination monitoring and would be of value for monitoring low energy radiation.

Based on the above findings, improvements in the following areas are required to achieve an acceptable program:

- enforcement of personnel contamination self survey requirements and the provision of adequate portal monitors at appropriate locations. (50-333/80-20-19)

Based on the above findings, the following matter should be considered for improvement:

- The use of thin window GM tubes for use in performing personnel whole body frisking.

#### 4.0 Radioactive Waste Management

##### 4.1 Documents Reviewed

- a. Standing Order No. 7, "Solid Waste Isotopic Analysis - Indirect Method," dated June 7, 1979.
- b. Procedure No. PSP-8, Revision 4, Radioactive Material Shipping Procedure dated August 11, 1980.
- c. Procedure No. 21A, Revision 2, "Shipping of Radioactive Waste," dated October 3, 1980.
- d. Procedure No. 21B, Revision 2, "Emergency Notification - Radwaste Shipment Trouble\*," dated October 3, 1980.
- e. Operations Department Standing Order Procedure No. 9, Revision 1, "Radioactive Waste Shipment," dated October 3, 1980.
- f. "Operating Procedure for CNSI Solidification System Units - FitzPatrick," Revision 1, dated May 2, 1980.
- g. Operating Procedure No. F-OP-20, Revision 5, "Standby Gas Treatment System," dated August 26, 1980.
- h. Operating Procedure No. F-OP-24A, Revision 5, "Off Gas System," dated March 11, 1980.
- i. Operating Procedure No. F-OP-31, Revision 4, "Process Radiation Monitoring System," dated August 10, 1979.

- j. Operating Procedure No. F-OP-32, Revision 1, "Area Radiation Monitoring System No. 18," dated February 27, 1979.
- k. Operating Procedure No. F-OP-48, Revision 4, "Solid Radwaste System," dated October 17, 1980.
- l. Operating Procedure No. F-OP-49, Revision 6, "Liquid Radioactive Waste System," dated June 11, 1980.
- m. Operating Procedure No. F-OP-50, Revision 2, "Equipment and Floor Drainage System", dated July 7, 1980.
- n. Operating Procedure No. F-OP-51, Revision 2, "Reactor Building Ventilation and Cooling System", dated July 12, 1979.
- o. Operating Procedure No. F-OP-52, Revision 1, "Turbine Building Ventilation", dated April 25, 1980.
- p. Operating Procedure No. F-OP-54, Revision 1, "Radwaste Building Heating and Ventilation System", dated April 28, 1980.
- q. Operating Procedure No. F-OP-55B, Revision 2, "Control Room Ventilation and Cooling", dated March 26, 1980.
- r. Work Activity Control Procedure No. 10.1.11, Revision 0, "Handling Procedure for CNSI-14-195 Cask", dated November 17, 1980.
- s. Temporary Procedure No. 44, Revision 0, "Procedure for Determining Waste Shipments Can Be De-Watered to Less Than One Percent," dated November 19, 1980.
- t. Process Survey Procedure PSP-4, Revision 1, "Waste Water Sampling and Analysis," dated July 14, 1980.
- u. Process Survey Procedure PSP-5, Revision 1, "Radioactive Airborne Sampling, Analysis and Equipment Calibration", dated May 18, 1979.
- v. Process Survey Procedure, PSP-6, Revision 1, "SBGTS and Crevass Filter Testing", dated July 10, 1980.
- w. Process Survey Procedure, PSP-10, Revision 1, "Auxiliary Boiler System - Sampling and Analysis," dated July 1, 1980.
- x. ANSI/ANS-55.1 1979, "American National Standard for Solid Radioactive Waste Processing System for Light Water Cooled Reactor Plants."
- y. ANSI/ANS-55.3 1976, "American National Standard Boiling Water Reactor Liquid Radioactive Waste Processing System."



- z. ANSI/ANS-55.4 1979, "American National Standard for Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants."

#### 4.2 Program Responsibility

During the course of the Health Physics Appraisal, the Radioactive Waste Management Program was evaluated. This evaluation consisted of a review of: assignment of program responsibility; staffing; solid, liquid and gaseous waste processing systems; waste disposition; effluent/process instrumentation; and personnel training and qualification.

The Radiological and Environmental Supervisor has responsibility for radiological control and radiochemistry. This included effluent monitoring and control. Although plant Standing Order Number 3 indicated this individual had responsibility for surveillance of radioactive material shipments leaving the site, this individual provided limited oversight of shipping activities. This extent of oversight appeared to be limited to providing radiation surveys of the package and radionuclide concentration analysis of the waste.

The Water System Supervisor essentially manages all radioactive waste processing at the FitzPatrick facility. This individual is responsible for operation of the liquid and solid radioactive waste systems, solid radioactive waste handling and preparation and packaging for shipment, solid radioactive waste.

The Water Systems Supervisor reports to the Operations Superintendent. This latter individual reports to the Station Resident Manager through the Superintendent of power. Although the Radioactive Waste Program responsibility was assigned at a sufficiently high level and the reporting chain appeared adequate for proper attention, review and management oversight, the appraisal team's review of the program indicated little apparent management oversight of the radioactive waste program.

The auditors noted that although the Water System Supervisor is managing the preparation and packaging of radioactive waste for shipment, the Radiological and Environmental Services Superintendent was designated as responsible for radwaste shipping. As discussed above, this latter individual provided little oversight of actual shipping activities. The auditors noted that plant management should act to clarify this situation.

Sampling and analysis of effluents is provided by the Radiological and Environmental Services Group. A member of this group, a laboratory technician initiates a liquid waste discharge permit. This permit is given to the shift supervisor who, through his signature of the permit signifies his approval of controlled discharge of the liquid waste.

Based on the above findings, improvement in the following area is necessary to achieve an acceptable program:

- Formally designate the responsibilities for the preparation and packaging of radioactive waste for transport and formally assign responsibility for approval of delivery of packaged waste for transport. (50-333/80-20-20)

#### 4.3 Liquid and Gaseous Waste Processing/Disposition

##### 4.3.1 Liquid Radioactive Waste

##### 4.3.1.1 Liquid Waste Processing

The liquid radioactive waste system at the FitzPatrick plant is designed to treat radioactive liquid waste by several different methods. These methods include: filtration; demineralization; concentration; and neutralization. The liquid waste is classified prior to and after processing based on the liquid's conductivity, chemical or detergent content. Control of liquid waste is from a separate radioactive waste control room located in the rad waste building. Most operations are automatic and do not require operator attention.

High Purity (low conductivity) waste originates primarily from facility equipment drain sumps and from recycle and process lines in radwaste. This liquid is treated by the waste collector system which utilizes precoat filters and deep-bed demineralizers as the means of treatment. The waste collector system can be augmented through cross ties with the floor drain collector system and fuel pool demineralizers.

Low Purity liquid waste (high conductivity) primarily originates as liquid collected in floor drain sumps. These sumps are pumped to the floor drain collector tank and routed to either the waste collector tank, floor drain sample tanks or waste neutralizes tank via the floor drain sample tanks.

Chemical and detergent wastes are both treated by use of neutralization while the detergent wastes are also treated by concentration.

The licensee does not routinely experience problems with handling liquid capacity. However, this capacity is reduced when the waste concentra-

tor is removed from service. Only one of two concentrators was indicated as being operational at the time of the appraisal. When the concentrator is out of service, filtration is used in-lieu of concentration.

Although the radwaste facility was designed for peak loads (usually during startup or shutdown), the licensee has at some times experienced too high a water inventory which resulted in the release of several hundred thousand gallons of liquid. The licensee normally releases liquid waste from the laundry at the rate of approximately 2000 - 3000 gallons/day.

Discussions with the Water Systems Supervisor indicated that difficulty has been encountered with process systems failing to provide the expected treatment decontamination factors. An example is the thin film evaporator which passes high conductivity water as a result of carry-over. This carry-over has caused plugging of one evaporator eductor. As a result of these problems, this evaporator is not used. The licensee has also experienced problems with the centrifuge in that sand carry-over from a sand filter has damaged the centrifuge. The licensee has also experienced difficulty with powder resins breaking through the deep bed demineralizers. This has resulted in frequent backwashing of the condensate demineralizers to remove the powder resin. No routine program is in place to periodically evaluate radionuclide decontamination factors.

Auditor discussions with the licensee's Maintenance Superintendent indicated no routine maintenance program exists for the radwaste system. The discussions indicated the radwaste system was apparently built on a "low bid concept." This has resulted in a radwaste system whose design and construction did not consider frequency or ease of maintenance, both of which affect personnel exposure. Because of this low bid design, the radwaste system is not standardized, i.e., different pumps, valves etc. may be used in the same system thereby increasing the difficulty in maintenance.

The licensee's Maintenance Superintendent has, as a result of the above, been concentrating on performance of modifications to standardize and

upgrade the radwaste system components. The Maintenance Superintendent indicated that valve packing is being changed, and where possible mechanical seals are being installed to reduce liquid waste. Also, during preparation for work, system prints are reviewed to determine if any additional work can be performed on components within the isolation boundary established for the initial work. This allows upgrading and maintenance of components during one work and repair session. This concept also would appear to provide lower overall man-rem exposures. This program is however not formalized.

#### 4.3.1.2 Liquid Waste Disposition

The licensee does not normally release large quantities of liquid waste. Those releases which are made originate primarily from laundry waste. The sampling, analysis, movement and discharge of liquid waste is performed in accordance with licensee approved procedures. Technical Specification release limits were written into the appropriate procedures.

Procedures were in place for both computer and manual analysis of the sample activity data. A liquid radioactive waste permit is completed prior to release of liquid wastes. The permit is signed by a shift supervisor to indicate discharge approval, discharge. The procedures and permit appeared adequate to assure that controlled liquid releases were within Technical Specification and 10 CFR 20 limits. Liquid waste is discharged through a monitored line. The discharge permit specified setting trip alarms to terminate the release in the event of higher than anticipated activity.

The licensee had procedures in place for sampling potential sources of radioactive releases. This included a procedure for sampling the auxiliary boiler system and a draft procedure under development for sampling the sanitary sewage system effluent.

The procedures appeared adequate to meet Technical Specification surveillance requirements and 10 CFR 20 release limits.

#### 4.3.1.3 Liquid Monitoring System

The liquid systems monitored included the reactor building closed cooling water loop, service water, and radwaste discharge.

Procedures were in place for calibration and functional testing of the radwaste effluent, service water and reactor building closed loop cooling water loop monitors. The procedures provided for quarterly source checks and yearly calibrations with known source activities. The quarterly source check instrument reading is based on the reading calibration. The quarterly checks for 1980 were noted to be consistent with the value obtained at the yearly calibration. Three different known source activities were used for the yearly calibration. The sources ranged from  $1E-5$  to  $1E-2$  uCi/ml and provided a "K" factor (uCi/ml per count/second) for future use. A sample cannister containing demineralized water is used for background determination. The monitor's useful range (using the K factor) were from approximately  $1E-7$  to  $1E-1$  uCi/ml. This lower limit met the minimum detectable level for Co-60 and Cs-137 as specified in ANSI N13.10-1974.

Auditor discussions with licensee Radiological and Environmental Services personnel indicated the residual heat removal services water monitor has apparently been out of service for 5 years due to inability of samples to reach the detector, discharge of service water from this system is monitored by the main service water monitor.

Review of the radioactive liquid effluent monitor calibration with respect to the licensee's Appendix B Technical Specification 2.3.A.7 indicated the monitor was being calibrated quarterly with a known radioactive source as required by the specification.

#### 4.3.2 Gaseous Waste Processing/Disposition

##### 4.3.2.1 Gaseous Waste Processing

The principal gaseous waste processing system at the FitzPatrick facility is the offgas system. This system receives offgas from the main condenser air ejector and processes it through a recombiner/charcoal system. With the recombiner in service,

the gas is heldup for 5 hours which allows for significant decay of the radioactive gases prior to release. With the recombiner out of service, the gas due to the increased volume, undergoes only approximately a 30 minute holdup.

The facility has charcoal (~ 17 tons) filled tanks to provide additional holdup of the gases. Use of the charcoal for holdup results in a 98% reduction in the discharged gas activity as compared to the tanks inlet gas activity. The licensee has not been utilizing the recombiner or charcoal beds since August 1980 due to leaks in the off gas system and problems with the gas dryers. With the recombiner and charcoal beds out of service, the off gas release rate from the stack is several hundred times higher than with these items on. This was confirmed through review of gaseous release rate data. On several dates, gaseous release rates were noted to be several thousands of uCi/sec with the recombiner off versus release rates in the low tens of uCi/sec with the recombiner and charcoal beds in service. Discussions with licensee radiological and environmental service personnel indicated the higher release rates were due to the recombiner and charcoal beds not being used.

The facility utilizes a conventional Standby Gas Treatment System (SBGTS) to filter and exhaust reactor building atmosphere via the stack during secondary containment isolation conditions. Gases discharged from the primary containment during inerting and deinerting are also passed through the SBGTS before being released to the atmospheres. The SBGTS and Control Room filters and charcoal adsorbers are tested (removal efficiency and adsorption) in accordance with Procedure PSP-6. Review of the procedure indicated testing was being performed by a contractor.

In-place testing and laboratory charcoal testing appeared consistent with Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." The auditor noted that Procedure PSP-6 did not have a test data sheet attached, rather the "final" data was provided to the licensee by the contractor. The auditor noted that the licensee



should develop a procedure data sheet with the various allowable test parameters and acceptance test limits indicated on the data sheet. This would provide a convenient method for test result and parameter comparison with acceptance criteria. This data sheet should also include results of system visual inspections prior to testing.

The procedure used by the contractor for in-place testing is referenced in the licensee's filter testing procedure. The auditor noted this procedure was apparently not reviewed and approved by the licensee. In view of the fact that the procedure is used for periodically testing components (i.e., control room ventilation and SBGTS) that are important during accident conditions, this procedure should be reviewed for adequacy by the licensee. (50-333/80-20-21)

The auditor also noted that this licensee's Technical Specifications do not require an air flow distribution test across the HEPA filters or iodine adsorbers of the SBGTS or control room ventilation system, rather a pressure drop test was utilized. The auditor noted that if system design permits, a flow distribution test should be performed. This test would indicate non-uniform loading of filters and non-uniform flow to charcoal adsorbers. This latter condition would promote faster iodine breakthrough during accident conditions.

The licensee did indicate portable ventilation systems are onsite and available for exhausting tents. These are discussed in Section 6.3.5 of this report. The auditor could not determine if removal efficiency testing was performed for these portable ventilation systems.

#### 4.3.2.2 Gaseous Waste Disposition

Since the FitzPatrick facility is a boiling water reactor, gaseous waste is released continuously during operation. The licensee has established procedures for required sampling, analysis and monitoring of particulate, halogen and gaseous effluents.

Because the licensee's facility is in close proximity to Niagara Mohawk's Nine Mile Point Nuclear Facility, the airborne effluents from that facility are included in allowable release rate calculations for the FitzPatrick facility. Discussions with Radiological and Environmental Services Technicians indicated gross activity release rates are obtained each morning from the Nine Mile Point Facility while halogen and particulate release rates are obtained each Wednesday. These values are then used in determining compliance with Technical Specification release rate limits.

Procedures were in place which provided guidance for meeting Technical Specification surveillance requirements including allowable release rates, required sample analysis and compositing.

The procedures appeared adequate to meet Technical Specification effluent release surveillance requirements.

#### 4.3.2.3 Gaseous Monitoring System

The gaseous monitoring system consists of the air ejector off-gas radiation monitors, main stack radiation monitors, drywell airborne monitoring system and the ventilation radiation monitoring system. The ventilation monitoring system consists of a turbine building exhaust, radwaste building exhaust, refueling floor exhaust, reactor building (lower floor) exhaust and control room intake monitor. The refuel area and reactor building ventilation detectors monitor effluents that eventually combine and exit one release point. The turbine building and radwaste building exhaust detectors monitor effluents that exit separate release points.

The main stack and ventilation monitors consist of a gaseous monitoring system with particulate and iodine cartridges for removal and counting. The air ejector off-gas monitor uses gamma ionization chambers for monitoring off-gas activity. The drywell airborne monitor uses a beta scintillation detector to monitor gaseous and particulate activity while a gamma scintillator monitors iodine activity. The particulate and iodine monitors of the drywell system view the collection media.

Procedures were in place for calibration and setting of alarms for the gaseous monitoring systems.

The main steam lines are monitored by four gamma sensitive ionization chambers. Procedure PSP-14 provides instructions for calibration of these detectors. The procedure indicates "source cans" are to be used which provide radiation levels in the range 10 to 100 mR/hr, 100 to 900 mR/hr and 1000 mR to 75% of the HiHi alarm setpoint. The radiation sources are not identified in the procedure nor is "75% of HiHi" identified. An R-chamber and cutie-pie are used to measure the dose from the source cans. These detectors are calibrated each refueling outage in accordance with Technical Specification requirements.

Procedure PSP-14 also provided guidance for calibration of the steam jet air ejector monitors. These were calibrated at three points with known dose rates each refueling outage with the source cans used for calibration of the main steam line monitor.

Review of Procedure PSP-5 indicated the off-gas monitors were also being calibrated by collection of a sample and determining a K factor ( $\mu\text{Ci}/\text{sec}$  per mR/hr). This was performed periodically during a calendar quarter and served as the method for quarterly calibration and alarm point setting. A yearly calibration was performed with sources in the range 50-4000 mR/hr. No source to be used was identified in the procedure nor was there any indication the source was to be NBS traceable. Procedure PSP-5 discusses review of off-gas release rates ( $\mu\text{Ci}/\text{sec}$ ) versus power level to determine if release rate has changed with changing to power level. Changes greater than  $\pm 20\%$  are to be reported to a Radiological and Environmental Services supervisor. The procedure does not provide instruction for review of the K factor unless the new value is greater than 110% of the previous K. The new K factor is reported to control room in this event or once each month, whichever comes first. The auditor noted changes in the K factor without changes in power would indicate detector/monitoring system malfunctions or possible fuel problems.

The stack gaseous monitoring system was being calibrated in accordance with Procedure PSP-5 in a similar manner (K factor). No external sources, other than a Cs-137 internal check source, were used. An annual calibration similar to the quarterly calibration was being performed.

The vent monitors were also being calibrated by collection of a sample and determination of a K factor. These monitors had internal check sources but no indication of source values was contained in the procedure. The calibration data sheet did have a location for values due to exposing the check source both before and after calibration. However, no acceptance limit for "before" versus "after" readings, (e.g.  $\pm 10\%$  etc.) was found in the procedure.

The auditor noted during a review of the stack and vent monitoring system calibrations that a sample is taken of the appropriate ventilation input and a K factor ( $\mu\text{Ci}/\text{sec}$  per cps) is obtained. The K factor provides information relative to a single point on the detector's range and does not provide any indication of system linearity over the range of the instrument.

Technical Specification Section A.1.F.2 defines an instrument calibration as an adjustment of an instrument signal output so that it corresponds within acceptable range, and accuracy to a known value(s) of the parameter which the instrument monitors.

The auditor noted that Regulatory Guide 1.21 recommends that calibrations be performed for the full range of the readout device for continuous radioactivity monitoring systems. It further sets forth the need to establish a relationship between concentration and monitor readings over the full range.

The range of the stack monitor (gaseous channel) is from approximately .2 to  $2 \times 10^6$   $\mu\text{Ci}/\text{sec}$ . The stack gas K factor, in use during the appraisal period was 2.0  $\mu\text{Ci}/\text{sec}$  per cps. Using this factor and assuming the factor was appropriate for the stack gas reading during the period, a release rate of  $\sim 5 \times 10^3$   $\mu\text{Ci}/\text{sec}$  is obtained. This K factor was being used over the range of the instrument without verification that the K

factor was linear over this range and acceptable for use. The auditor could not identify any procedural guidance for review of K factors other than that previously described.

The auditors noted that calibration of the gaseous effluent monitors was consistent with Appendix B Technical Specification Section 2.3.B.9 which requires that these monitors be calibrated quarterly by means of a check source and annually with a known radioactive source. Excluding the off-gas, monitors which are calibrated annually at three points with known radioactive gaseous effluent monitor calibration at one point does not appear to provide a indication of radiation source response over the entire range of monitor readout, consequently the licensee should consider checking these instruments' responses at several points with known radioactive sources e.g. yearly.

Based on the above findings, improvement in the following area is needed to achieve an acceptable program:

- Expedite repair of Off Gas Treatment System dryers and system leaks to allow use of this system (Section 4.3.2.1).  
(50-333/80-20-22)

Based on the findings in the above area, the following matters should be considered for program improvement:

- Establishment of a formal maintenance program for radioactive waste systems (Section 4.3.1.1).
- Review of radioactive waste system design and operation to assure adequate standby capacity is available in the event of critical component failure (Section 4.3.1.1).
- Review and approve contractor inplace testing procedure for Standby Gas Treatment System on control room ventilation system (Section 4.3.2.1).
- Perform flow distribution tests on Standby Gas Treatment and control room ventilation system if system design permits (Section 4.3.2.1).

- Include acceptance criteria and test parameters for ventilation system tests in procedures for in-place and laboratory testing (Section 4.3.2.1).
- Review portable ventilation units to ensure removal efficiency tests are performed on the units (Section 4.3.2.1).
- Calibration of Gaseous effluent monitors at more than one point (Section 4.3.2.3)

#### 4.4 Solid Radioactive Waste

##### 4.4.1 Systems and Storage

The licensee processes both wet and dry solid wastes. Wet solid wastes consists primarily of spent resins from the waste and condensate demineralizers and includes backwash sludge from various filters and demineralizers. Dry solid waste consists of contaminated material such as rags, paper and other material.

Wet solid waste is discharged to the phase-separator or directly to the waste sludge tank. Decontaminated liquid is pumped to the waste or floor drain collector tank for processing. The wet solid waste, upon reaching the desired concentration in the sludge tank, is fed to the concentrated waste tank for processing. This processing consists of solidification by a contractor firm which solidifies the waste onsite prior to shipment. A 10 CFR 50.59 review of the system was completed on December 20, 1978 and indicated that the operation of the system would not involve an unreviewed safety question.

The dry solid waste is compacted into drums or LSA boxes by means of a hydraulic press (compactor). The compactors are fitted with high efficiency filtration systems. Procedures were in place for operation of these compactors.

Review of solidified and compacted radwaste storage indicated the licensee would, if required to store waste onsite, have sufficient storage capacity for approximately one years storage of LSA waste (compacted low level waste). The licensee's Water Treatment Supervisor indicated that little storage space is available for high level waste storage. A block wall area outside the main building has been constructed and was indicated as being able to hold 6 liners. The Water Treatment Supervisor indicated that approximately six months shielded storage capacity for high level waste is onsite. This storage capacity was



noted to be greater than the 30 days recommended in ANSI/ANS-55.1-1979. A low level radioactive waste storage building is to be constructed to accommodate low level waste. This building is anticipated to have a capacity of 6 to 8 months storage.

The block wall storage structure discussed above, was apparently reviewed and documented in plant operating review committee meetings as not needing a documented 10 CFR 50.59 review because the structure was "Temporary." This structure was built approximately 4 months prior to this appraisal.

A reverse electroplating device is being used by the licensee for decontamination of metals. A licensee representative stated that from 40,000 to 50,000 pounds of metals have been recovered and not disposed of as radioactive waste through the use of this device.

The licensee established a Radwaste Committee early in 1980. Part of the responsibilities of this committee was to reduce the volume of radwaste being shipped offsite. The committee placed an individual at the main control point to observe material being taken into the controlled area and to ask those workers carrying in material which would become radwaste such as packaging, boxes, etc. not to take the material in unless it was essential to the performance of the work.

The auditors noted that the committee held 4 meetings after which the committee stopped meeting apparently due to time constraints imposed by outage work. The individual that monitored material being taken into the controlled areas was removed from that job and placed on the trash separation crew.

The auditors noted that an outage was an important time to have a Rad Waste Committee in effect since an outage results in generation of significant volumes of radioactive waste. The trash separation crew discussed above separated contaminated from noncontaminated trash. The auditors noted that control of material taken into potentially contaminated areas should be used in lieu of trash separation for radwaste volume reduction.

The licensee's worker training program was indicated as containing instructions that unnecessary material not be brought into the controlled area. No instructions regarding material control were posted at potentially contaminated area entry points.

#### 4.4.2 Solid Rad Waste Shipment

##### 4.4.2.1 General

The licensee normally ships, on the average, 2 to 3 concentrated solidified radioactive waste shipments per month, 1 dewatered resin shipment every other month and approximately 2 to 3 boxes of LSA waste per month. The number of concentrated waste shipments during the 1980 period was substantially higher due to problems with powdex resin breakthrough of condensate demineralizers resulting in frequent resin regeneration.

##### 4.4.2.2 Quality Assurance Program

The licensee has included Quality Assurance (QA) of Radioactive Waste shipping activities in his 10 CFR 50, Appendix B, Commission-approved quality assurance program. Quality Assurance Procedure QAP No. 2.1, "Quality Assurance Program Scope" includes packaging of radioactive material for transport and transportation of radioactive material as an item to which the QA program applies.

Audits of radioactive waste shipping are performed in accordance with Quality Assurance Procedure QAP 18.1, "Quality Assurance Audit Program - Plant," Revision 5. This procedure describes performance of standard and surveillance audits. The standard audits are used to provide information relative to satisfactory completion of a procedure in its entirety while the surveillance audit is used to "spot check" selected procedural requirements.

The auditor reviewed various radwaste audits (standard and surveillance) performed by the site QA organization. These included audits of radwaste shipping and conformance to IE Bulletin 79-19, "Packaging and Shipment of Low-Level Radioactive Waste."

The auditor determined through discussions with the Site Quality Assurance Engineer that neither he nor his auditors had any expertise in rad waste shipping, rather it was assumed that the individual performing radwaste handling, solidification and shipping was proficient in this area. It was likewise assumed that the existing radwaste procedures were adequate.

The QA Department did not conduct a routine, review of each radwaste shipment to ensure that the type of cask used and shipment procedures performed were being used in accordance with appropriate requirements. No fixed schedule was identified for the review each type of shipment and cask.

As previously discussed the individual assigned responsibility for "surveillance" of radioactive waste shipments leaving the site (Radiological and Environmental Services Superintendent per Standing Order No. 3) was providing limited oversight of radwaste shipments. The auditor noted that the individual who appeared to be actually responsible for shipments (Water Treatment Supervisor) was utilizing shipping procedure check lists to provide QA oversight of shipments. The procedure check list used, (OP-48 Table III), was general in nature and not sufficiently detailed to assure compliance with applicable requirements. These included burial site acceptance limits, 10 CFR 71.12 general license requirements, weight limitations, and cask handling requirements such as lid torque limits.

The licensee was utilizing an onsite contractor solidification system (urea-formaldehyde) to solidify liquid waste. Auditor discussions with licensee QA representatives indicated that there was no QA oversight of this system other than a determination that the system was being operated without approved procedures. The auditor determined that the licensee had not evaluated, nor had the licensee's QA group identified the need to evaluate, the solidification system's product to ensure it met burial site requirements. For example, the amount of free standing water and amount of transuranics in the solidified waste being shipped had not been evaluated as of the time of the appraisal.

The auditor noted that 10 CFR 30.41 prohibits transfer of byproduct material unless it is in a form authorized by the recipient's NRC or Agreement State license and that South Carolina License No. 097, an Agreement State license prohibits receipt of solidified waste which contains detectable free standing water or transuranic concentrations per gram of waste which exceed specified values.

Through an interview with a licensee representative the auditor established that between January 1, 1976 and November 10, 1980, licensed byproduct material in solidified waste and dewatered resins, was transferred to the holder of South Carolina License No. 097 and no verification was made that the limits on transuranic concentrations and detectable free standing water were not exceeded.

Upon auditor identification of this item, licensee representatives suspended radioactive waste shipments pending completion of free-standing water and transuranic content evaluation. The licensee subsequently established a test procedure (Temporary Procedure No. 44) for use in determining the amount of water remaining in a radwaste liner after a typical de-watering operation. The results of the test indicated that the de-watering operation would result in less than 1% free-standing water. Licensee Radiological and Environmental Service personnel performed a preliminary evaluation of total transuranic content of radwaste and determined the content would be under 10 nanocuries per gram.

During the review of this area, the auditors reviewed several radwaste shipments made by the licensee and performed independent radiation dose measurements of several shipments.

The shipment loading and review indicated the following deficiencies:

- The shipping cask (CNSI-14-195) routinely used by the licensee for shipping greater than Type A waste was being loaded and closed without approved loading and closure procedures. 10 CFR 71.54 requires these procedures to be in place. As of the time of the appraisal, licensee procedures had not been established and approved. The licensee suspended shipments by this cask pending establishment of approved procedures.
- 10 CFR 71.12(b) requires that a copy of all documents referenced in the cask certificate of compliance, (in this case Certificate of Compliance Number 9094), be on hand. A cask containing 14.5 curies was shipped on November 10, 1980, and at other times

without all referenced documents being on hand. A cask drawing, used to verify that the correct cask is being used was not on hand.

- The auditor determined from questioning a licensee representative that during the last two years several shipments of greater than type A quantities had been made in single packages. He further determined that no records had been made of the results of routine determinations, required by 10 CFR 71.54, among other things, that the packages were undamaged, that package closure and sealing gaskets were present and defect free and that they were loaded and closed in accord with written procedures.

The latest example of this failure to generate and maintain such records concerned a shipment made on November 10, 1980. This shipment consisted of a single package containing 14.552 curies of radioactive material, 10.301 curies of which were transport group III radionuclides (cobalt 60; 9.1 curies; cesium 134, 0.695 curies and cesium 137, 0.506 curies). The licensee representative stated that no record of the 71.54 routine determinations had been made. 10 CFR 71.62(a)(10), requires the licensee, to maintain records of the results of routine determinations performed in accordance with 10 CFR 71.54. The auditor noted that as of the time of the appraisal the licensee was not routinely maintaining records of these determinations. These records, are to be maintained for each shipment in a single package of a greater than Type A quantity of radioactive material and are to include records of the determinations that: the package has not been significantly damaged, the closure of the package and the sealing gaskets are present and free from defects, and the package has been loaded and closed in accordance with written procedures. In addition to a shipment of greater than Type A material in a single package on November 10, 1980, (discussed without all referenced documents being on hand. A cask drawing, used to verify that the correct cask is being used was not on hand.

- The auditor determined from questioning a licensee representative that during the last two years several shipments of greater than type A quantities had been made in single packages. He further determined that no records had been made of the results of routine determinations, required by 10 CFR 71.54, among other things, that the packages were undamaged, that package closure and sealing gaskets were present and defect free and that they were loaded and closed in accord with written procedures.

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of greater than Type A material in a single package without records of routine determinations being maintained.

Based on the above findings, improvement in the following areas are needed to achieve an acceptable program:

- Establish and implement radioactive waste shipping cask loading and closure procedures to meet the requirements of 10 CFR 71.54. (50-333/80-20-23)
- Establish and implement means to maintain and update all documents required to be on-hand prior to shipment of radioactive waste. (50-333/80-20-24)
- Establish and implement a radioactive waste shipping records program which meets the requirements of 10 CFR 71.62. (50-333/80-20-25)
- Review all radioactive waste storage areas to assure a documented 10 CFR 50.59 evaluation is on file including temporary storage areas which had not been previously reviewed. (50-333/80-20-26)
- Establishment and implementation of a quality assurance program sufficient to assure radioactive waste is packaged, transported and transferred in accordance with applicable regulatory requirements (Section 4.4.2.2). (50-333/80-20-27)

Based on the above findings, the following matters should be considered for improvement of the program:

- Establish a formal radioactive waste volume reduction program (Section 4.4.1).

## 5. ALARA Program

### 5.1 Documents Reviewed

Administrative Procedure No. 6.1, "Plant Chemistry, Radiation Protection and Environmental Control", Revision 1, April 28, 1980

Plant Standing Order No. 2, "JAFNPP ALARA and Respiratory Protection Policies", Revision 1 November 4, 1980

### 5.2 General

At the time of the appraisal the licensee did not have an ALARA program in effect. When the plant effort was compared against the aspects of an ALARA program described in Regulatory Guides 8.8 and 8.10 the only points which could be identified were:

Plant Standing Order No. 2 states, in Section 7, that the Power Authority of the State of New York is committed through its radiation protection program to maintain occupational radiation exposures as low as reasonably achievable (ALARA). It further states that as part of the training program, aspects of the radiation protection program dealing with maintaining exposures, ALARA shall be discussed and that the employees shall be made aware of the Authority's and their own responsibilities and of the Authority's commitment to meet this end. The Auditor attended a typical training program and noted that these points were covered.

Aside from the above, the auditors could find no evidence of a formalized ALARA program within the Radiological and Environmental Services Department. It was observed that the individual responsible for maintenance activities had an appreciation for the importance of ALARA principles as they applied to the workers in his organization. He closely monitored accumulating doses and planned and took actions that were designed to keep collective doses ALARA. However, the auditors could find no evidence that these concepts were consciously applied in other departments. One senior technician had been designated as responsible for ALARA but this was not a full time assignment and it was not possible to identify any formal framework of policies and procedures within which his work was carried out.

Based on the above findings improvement in the following areas is required to achieve an acceptable program: (50-333/80-20-28)

- Establish, document and implement a formal corporate and plant ALARA program that conforms to the guidance in Section C of Regulatory Guide 8.8, and to Regulatory Guide 8.10.
- Full-time professional level manning plus the necessary supporting personnel must be provided to operate the plant ALARA program. The necessary corporate level manpower should be provided.

- Procedural action levels in radiation work permit review, planning and job review, consistent with good ALARA principles for individuals, as well as the collective worker exposure group are needed.

## 6.0 Health Physics Facilities and Equipment

### 6.1 Health Physics Facilities

#### 6.1.1 Sample Counting Area

The licensee maintained a laboratory in which several types of analytical counting instruments were used. GeLi gamma ray spectrometers were (8.9% and 16.2% efficient) used for both health physics and chemistry sample counting. A well-type 2"x2" NaI crystal spectrometer attached to a single channel pulse type analyzer was available for backup. The facility's beta and alpha counting for analytical data was performed with a PC-5 windowless gas flow proportional counter. The analytical laboratory detectors were shielded with lead. The back wall of the analytical laboratory was part of the 3'6" shield wall over the main steam line and feedwater lines to the reactor, and was equivalent to 8 inches of lead. This wall should be reviewed for shielding adequacy if leaking fuel becomes a problem. The counting area appeared to have adequate work space for performing counting activities.

#### 6.1.2 Instrument Calibration Facility

The facility's instrument calibration laboratory was located in an area remote from traffic, however, the area was dusty. The calibration device was suitable for the types and ranges of instruments used at the facility and there were and had suitable jigs for accurate and reproducible placement of instrumentation into the gamma ray fields. Calibration of neutron rate and neutron rem meters was performed offsite. Calibration of beta measuring instrumentation was performed with a uranium slab. Radiation sources were NBS traceable. No alpha emitting sources were used as no portable alpha detecting instrumentation was onsite.

#### 6.1.3 Personnel Decontamination Facility

An area close to the health physics technician's office at the 272 ft. level was used for personnel decontamination. A sink, and an immediately adjacent shower for whole body decontamination, hair washing and major decontamination processes were available.

The appraiser noted that the facilities were in a state of disrepair, and that the only supplies were soaps, and a cloth towel. No spare towels were in the immediate area. The shower did not contain the same variety of decontamination media as did the sink just outside the shower area. No shower mats were available. Paper suits for persons to dress into were not available at the shower facility.

#### 6.1.4 Protective Clothing Change Areas

The auditor observed during tours through the facility that change areas were frequently placed very close to the work sites. It was noted that radiation exposure rates to personnel in some change areas were appreciably above background. Change and rest areas were not located consistent with ALARA practices. Although a locker room facility was located near the health physics office and access control point on the 272' level and had very low radiation levels, it was utilized only for changing from street clothes into work clothes. Although with minor modification, this area could have been used for donning and removing protective clothing, it was not used for this purpose.

#### 6.1.5 Access Control Points - Health Physics Office Areas

The main access control point outside of the health physics office had been designated as the place where personnel performed their final frisk and had tools and other materials wipe tested. This produced a situation where potentially contaminated items were brought unnecessarily close to the final crossover point to clean areas. It also produced substantial congestion, particularly at shift change. This situation encouraged inadequate contamination surveys of both personnel and equipment. Existing equipment, namely, the 30 mg/cm<sup>2</sup> sidewindow G.M. tube is inappropriate for frisking for low levels and low energy beta activity. The number of instruments available for frisking at this point was inadequate relative to the number of workers who needed to use such instrumentation.

Office space for health physics technicians staff was cramped. A single office 11'6"x13' was the only place provided for the health physics technicians, foremen and workers requiring radiation protection services to exchange information, sign-in, sign-out and read radiation work permits, review and write. The proposed change in the arrangement of the radiation protection areas would seem to be consistent with an improved flow and communication of information associated with the safe operation of the facility.

#### 6.1.6 External Dosimetry

The areas where dosimeters were issued and read were at some distance from the general health physics areas. Consideration should be given to locating these areas closer together for more consistent, consolidated operations when area redesign or space reassignment is done. The dosimeter reading area was a wire cage erected in the cable spreading area. The area was quite dusty when personnel were working around that area. The TLD reader, associated computer hardware and software, and TL dosimeters would all be more appropriately housed in a dedicated facility with a controlled environment, such as more dust-free atmosphere, better temperature control. Closer proximity of the issuance point to the reader and input/output terminal would increase speed and ease of communication afforded.

#### 6.1.7 Internal Dosimetry

The whole body counter was the major method used to perform internal dosimetry, and was housed in an area immediately adjacent to the respiratory mask decontamination area. A half-inch lead shield had been added to the wall between the storage area for the masks and the whole body counter to reduce radiation background in the whole body counting area.

Used and contaminated respirator masks come to a room immediately adjacent to the whole body counter. It was noted that contaminated materials often come in at a level which may impact the background of the whole body counter. Some consideration should be given to a physical separation of the whole body counter and the potentially higher background radiation area which contained used and contaminated respirators.

#### 6.1.8 Respirator Fitting, Testing and Decontamination, Protective Clothing Laundry

The respirator fitting and testing program booth, was found to be suitable only for qualitative fittings. The booth had not been used in approximately one year. It was located outside of the existing training facilities and was locked. A licensee representative felt that it was in a questionable state of repair.

The facility for decontamination of respiratory protection equipment appeared to be adequately designed to preclude the spread of contaminated wash material to the work environment.

The final check of respirator equipment for contamination prior to its repair and bagging was performed in part of the chemistry laboratory. A separate area should be provided for this activity. Greater separation between radioactive and nonradioactive areas in the chemistry laboratory could be achieved in the absence of the respiratory protection equipment bagging station. The licensee was not performing quantitative tests on repaired respiratory protection equipment to verify adequate repair.

Protective clothing was decontaminated in a laundry. There was adequate separation between the washing and drying facilities. Appropriate friskers appeared to be used consistently to prevent the passage of garments with contamination levels greater than those specified in station procedures.

#### 6.1.9 Training Facilities

The auditor performed a walk-through inspection of the training facility and conducted several review sessions with training personnel for the purpose of evaluating the adequacy of training facilities and training materials within specialized areas. The training facilities appeared to be adequate, conducive to learning, satisfactorily equipped. The area also contained a limited library of reference materials and training equipment adequate for the basic levels of general employee training, respiratory protection training and much of the operator training performed onsite.

The training coordinator indicated that space was available for the Training Department to expand. The planned expansion was for the purpose of mock-up and simulator type instruction training for ALARA considerations, and pre-operation supervision of functional operations.

#### 6.1.10 Tools and Parts Decontamination

In tours through the facility, the appraiser noted areas where tools and parts were decontaminated. Appropriate lay-down areas were set aside for such work. Controls for entry and exit from the area were consistent with good health physics practices.

Based on the findings in the above area, the following improvements are needed to achieve an acceptable program:

- Locate change areas and access control points consistent with ALARA principles (Section 6.1.4). (50-333/80-20-29)



- Provide additional personnel contamination frisking stations at appropriate locations to create conditions under which the procedure for personnel contamination self-surveys can be conscientiously followed and to permit better application of the ALARA principle. (50-333/80-20-30)
- Place suitable quantitative fit testing equipment in service and use to qualify workers for appropriate respirators to retest repaired respiratory protection equipment. (Section 6.1.8). (50-333/80-20-31)
- Provide additional personnel contamination frisking stations (Section 6.1.5). (50-333/80-20-32)

Based on the findings in the above area, the following matters should be considered for improvement:

- Improve personnel decontamination facilities readiness by providing additional supplies and in finishing the structural facilities (Section 6.1.3).
- Improve plant Radiation Protection Office facilities, particularly for technicians and foremen (Section 6.1.5).
- Movement of External and Internal Dosimetry Operations to areas with environments better suited for operation of the associated equipment (Sections 6.1.6 and 6.1.7).

## 6.2 Chemistry Facilities

### 6.2.1 Analytical Laboratories

The chemistry group had a large analytical laboratory which was utilized for water chemistry, atomic absorption, pH analysis, ion analysis, and photospectrographic analysis, as well as analytical nuclear counting. The latter facility was also utilized for health physics analytical counting. The radioactive and nonradioactive areas of the laboratory had contiguous surfaces. Consideration should be given to methods of reducing the probability for cross-contamination.

### 6.2.2 Sample Storage

Adequately shielded sample storage was not available in the area. Licensee representatives stated that local shielding was erected at the time of need, for example, when reactor coolant samples were brought in for analysis. It was noted that the final aliquots of composite samples were small in volume and were kept out of the way in a cabinet storage area beneath the bench tops while the remainder of the hot sample was poured down the hot sink which connected to radwaste.

### 6.2.3 Sampling Areas

The auditor reviewed areas for collecting samples of primary coolant, airborne effluents, containment environments, and the secondary coolant for monitoring reactor coolant chemistry. All systems had suitable air collection hoods where required, and air sampling devices were exhausted back into the original environmental downstream of the sampling point. Shielding appeared adequate for sampling during normal operation. Readouts for some monitoring devices were found in the immediate localities. Several devices read out in the control room. However, the laundry tank sample line to the sampling hood was reported to be plugged. This resulted in at least 5 entries per week to manually sample the tank which was located in a 30 millirem/hr radiation field.

Based on the above findings, this portion of the licensee's program appears acceptable, but the following matter should be considered for program improvement:

- Improve shielded storage facilities for radioactive samples (Section 6.2.2).
- Keep sample lines clear so that sampling programs are consistent with ALARA concepts (Section 6.2.3).

## 6.3 Protective Equipment

### 6.3.1 Respiratory Protection Devices

The respiratory protective equipment used by the licensee is described in the licensee's Radiation Protection Operating procedures and is discussed in Section 3.2.5.4 of this report. Equipment used included self-contained breathing apparatus, airline supplied and high efficiency filter equipped full face equipment. Airline supplied hoods and bubble suits also were used. Welder's respiratory protective equipment was also utilized. Review of selected devices indicated NIOSH/MSHA approved equipment was being used.

Review of respiratory protective equipment storage areas indicated an adequate supply of equipment appeared to be on hand. Since no major work was in progress during this appraisal, the auditors could not comment on the adequacy of current supplies for outage conditions.

### 6.3.2 Anti-Contamination Clothing

Auditor discussions with licensee representatives and review of supplies indicated an adequate supply of protective clothing, hoods, booties, etc., for normal operations and for off-normal operations, such as outages, was available. A laundry facility was maintained onsite.

Licensee representatives indicated that a stock of PC's was required to be maintained in the warehouse. When it appeared that a shortage was going to occur during peak usage times, an interchange with the adjacent facility at Nine Mile Point had been used. Monitoring of the PC's after laundering was routine, and appropriate limits had been established for release for reuse.

### 6.3.3 Temporary Shielding

Review of various locations throughout the facility indicated shielding, including lead brick, lead sheet, and lead blankets, was used for hot spots.

### 6.3.4 Containment Materials

Plastic sheets, tents, etc., were being used to prevent the spread of contamination. Suitable lay-down areas were established for decontamination or storage of contaminated materials.

Based on the above findings, this portion of the licensee's program appears acceptable, however the following matter should be considered for program improvement:

- Review portable ventilation systems to ensure an adequate number are available and that the systems are maintained and tested properly.

## 7. Exit Interview

The Appraisal Team met with licensee representatives (denoted in Annex A) at the conclusion of the appraisal on November 21, 1980. The Appraisal Team summarized the scope and findings of the appraisal. The findings were grouped into categories:

- a. Significant appraisal findings are summarized at the conclusion of the applicable sections or subsections of this report and are contained in Appendix A to the letter forwarding this report. The licensee's response to these findings, to be submitted in writing, will be reviewed upon receipt.

- b. Findings of lesser significance, but which are considered instrumental to improvement of the licensee's program, are summarized at the conclusion of the applicable sections or subsections of this report.

ANNEX A

Persons Contacted

## ANNEX A

### Persons Contacted

- \* J. P. Bayne, Senior Vice President - Nuclear Generation
- \* R. J. Pasternak, Resident Manager
  - J. J. Kelly, Corporate Health Physicist
- \* R. Baker, Superintendent of Power
- \* M. N. Brosee, Maintenance Superintendent
- \* E. Mulcahey, Radiological and Environmental Services Superintendent
- \* W. Fernandez, Technical Services Superintendent
- \* M. Cosgrove, Site Quality Assurance Engineer
- \* A. McKeen, Assistant to the Radiological and Environmental Services Superintendent
  - J. P. Flaherty, Assistant to the Instrumentation and Controls Superintendent
- \* V. Childs, Assistant to the Resident Manager
- \* D. Tall, Training Coordinator
  - D. Zimmerman, Radiological and Environmental Services Supervisor
  - M. Hunt, Nurse
- \* R. Baker, Superintendent of Power
- \* D. M. Thomison, Training Manager
- \* M. Kelleher-Paris, Radiological Engineer
- \* R. Converse, Operations Superintendent
- \* B. Mays, Water Systems Supervisor
- \* R. J. Vargo, Shift Technical Advisor
- \* G. Nott, Outage Coordinator
- \* C. Patrick, Information Officer
- \* W. Fernandez, Technical Services Superintendent

### NRC Personnel at Exit Interview (Other than Appraisal Team Members)

- \* J. C. Linville, USNRC Resident Inspector

\* denotes those present at the exit interview on November 21, 1980.

The auditors also held discussions with and interviewed other licensee and contractor employees. They included engineering, operations, quality assurance/control, training, maintenance and radiological controls and emergency planning personnel.



ANNEX B

James A. FitzPatrick Nuclear Station TLD Badge Irradiation Test

## ANNEX B

### James A. FitzPatrick Nuclear Station TLD Badge Irradiation Test

#### 1. TLD Irradiation Test

A limited TLD irradiation test was conducted to determine the ability of the licensee's TLD badges and processing system to accurately monitor the radiation doses received by workers. For the purpose of this test, the licensee submitted 22 TLD badges of the type routinely used for whole body monitoring of personnel. Two of the submitted badges were used as in-transit controls.

The badges were irradiated with gamma radiation from a calibrated NBS traceable Cs-137 beam irradiator while mounted on a 15 centimeter thick curved lucite phantom to simulate worker/badge geometry during irradiation. Because the TLD badge clip prevented the badge from being mounted flush with the phantom, radiation exposure dose rates 9 mm close to the source were utilized.

All irradiations were performed at the United States Department of Energy's, Radiological and Environmental Sciences Laboratory, Idaho Falls, Idaho. The irradiations were conducted in accordance with the Health Physics Society Standards Committee recommendations as contained in "Draft American National Standard Criteria for Testing Personnel Dosimetry Performance," ANSI N13.11 published July, 1978.

The results of the whole body TLD badge tests are presented in the attached table (James A. FitzPatrick Nuclear Station TLD Irradiation Test Data.)

#### 2. Test Evaluations

The performance criteria presented in ANSI N13.11, were utilized for the test evaluation.

Section 4 of the ANSI N13.11 indicates that personnel dosimetry performance in a given radiation category is considered adequate if for all applicable test range intervals and applicable phantom depths, the following relation is satisfied:

$$|B| + 2S < L$$

where  $|B|$  is the absolute value of the bias,  $S$  is the standard deviation of the performance index, and  $L$  is the tolerance level.

Radiation categories, test ranges and tolerance levels are presented in Table 1 of ANSI N13.11 while relations for the bias, standard deviation and performance index are presented in Section 2 of the standard.

Section 2 defines the bias of the values of the performance index  $P_i$  as

$$B \equiv \bar{P} \equiv (1/n) \sum_{i=1}^n P_i$$

and defines the standard deviation of the values of the performance index  $P_i$  as

$$S \equiv \frac{\sum_{i=1}^n (P_i - \bar{P})^2}{n - 1}^{1/2}$$

where the performance index  $P_i$  for the  $i$ th dosimeter is defined as

$$P_i \equiv \frac{(H_I^1)_i - (H_I)_i}{(H_I)_i}$$

and

$(H_I^1)_i$   $\equiv$  testing laboratory assigned dose equivalent index (millirem)

$(H_I)_i$   $\equiv$  Processor's reported dose equivalent index (millirem)

$\bar{(H_I)}$   $\equiv$  Average value of dose index chosen in each dose interval (millirem) (See Test data which follows)

Section 4.1.2 of ANSI N13.11 requires values of  $S$  and  $B$  to be obtained from the performance index in specified dose intervals in each test category.

The TLD test performed utilized radiation in Test Category I, i.e., photon radiation with an average between 300 Kev and 3 Mev. Two points were selected in the protection range (0.03 to 10 rem). The deep dose tolerance level is 0.3 or  $6/(\bar{H})^{1/2}$  (whichever is larger) in this range. The two test point groups were separated to evaluate each group.

- a. Test Group 1 (342 millirem - Cs-137)

$$\text{Bias} = B = 0.035$$

$$\text{Standard Deviation} = S = 0.089$$

$$\text{Tolerance Level} = L = 0.3 \text{ or } 6/(\bar{H}_I)^{1/2}, \text{ whichever is larger}$$

$$\bar{H}_I = 342 \text{ millirem}$$

$$L = 0.324$$

$$\text{Therefore } |B| + 2S < L \\ |.035| + 2(.089) = 0.213 < .324$$

Test Group meets tolerance level.

- b. Test Group 2 (2490 millirem - Cs-137)

$$\text{Bias} = B = 0.09$$

$$\text{Standard Deviation} = S = 0.053$$

$$\text{Tolerance Level} = L = 0.3 \text{ or } 6/(\bar{H}_I)^{1/2}, \text{ whichever is larger}$$

$$\bar{H}_I = 2490 \text{ millirem}$$

$$L = 0.3$$

$$\text{Therefore } |B| + 2S < L \\ |.09| + 2(.053) = 0.196 < 0.3$$

Test Group meets tolerance level.

### 3. Test Summary

The results of this limited test indicate the licensee is able to evaluate mid-energy Category I photon radiation in the protection range adequately.

James A. FitzPatrick

TLD Irradiation Test Data

<u>Test Group</u>	<u>Badge Number</u>	<u>Cs-137 Test (<math>I</math>)<sub>i</sub></u> <u>Dose Delivered (R)</u>	<u>Licensee* (<math>H</math>)<sub>i</sub></u> <sup>1</sup> <u>Readout (R)</u>
1	2475	0.342	.373
	2477	0.342	.355
	2484	0.342	.381
	2486	0.342	.347
	2487	0.342	.365
	2488	0.342	.358
	2491	0.342	.290
	2497	0.342	.391
	2499	0.342	.364
	2500	0.342	.315
	2.	2476	2.49
2478		2.49	2.71
2480		2.49	2.77
2482		2.49	2.64
2483		2.49	2.80
2485		2.49	2.88
2490		2.49	2.46
2493		2.49	2.74
2496		2.49	2.60
2498		2.49	2.66
Control	2479	-	.023
Badges	2494	-	.030

\* Note: Average of Control badge dose has been subtracted.