DOCKET NO. 50-201
EXTRA

SAFETY ANALYSIS

SPENT FUEL PROCESSING PLANT

Part B
of
License Application

Volume II

Copy No. 75

U.S. Atomic Energy Commission
Licensing & Regulation

NUCLEAR FUEL SERVICES, INC.

July 1962

L 7785

VI. ENGINEERING ANALYSIS OF PLANT

VI - ENGINEERING ANALYSIS OF PLANT

VI ENGINEERING ANALYSIS OF THE PLANT

6.1 In this section the salient features of a number of the engineering aspects of the plant are discussed including:

a) Ventilation

b) Sampling and Analysis

c) Maintenance

d) Shielding

e) Monitoring

f) Utilities

g) Criticality

Summary

6.2 The above subsections may be summarized briefly:

a) Ventilation There are four separate ventilation systems: 1) the general building ventilation, 2) the process ventilation, 3) the process vessel system, and 4) the dissolver off-gas system. The systems are designed to maintain more active areas at lower pressures than less active areas. All the systems are treated as needed and finally discharged through a 200-foot stack in sufficient quantity and velocity to assure adequate atmospheric dilution.

b) Sampling and Analysis The sampling methods used have been proved satisfactory for use in radiochemical plants. Most of the highly active samples are taken completely remotely and dilutions are made before the sample is brought out from behind shielding. Analytical work is done in hot cells backed up by radiochemical laboratories.

c) Maintenance Equipment which is subject to the highest contamination levels and which will require considerable maintenance is collected in two remotely operated and maintained cells -- the Process Mechanical Cell and the Chemical Process Cell. In the PMC major repair and replacement can be accomplished by remotely operated equipment or the equipment can be removed for disposal or for transfer to the maintenance area for decontamination. In the CPC defective equipment must be removed and replaced. In all other areas maintenance is done by contact

methods. In these cells the equipment, and sometimes the cells themselves, must be decontaminated sufficiently to allow entry of personnel to make the necessary repairs. Mechanical equipment such as pumps, which may be expected to require frequent maintenance, is located in shielded niches in the warm equipment aisle. Each of these may be isolated, decontaminated, and repaired without having to decontaminate any other piece of equipment.

d) Shielding Shielding has been designed to assure that backgrounds in normal access areas will not exceed 1 mr/hr. There have been sufficient safety factors included in these calculations that it is expected that most areas in the plant will have a background much lower than this.

e) Monitoring There is an extensive area and personnel monitoring system with both fixed and mobile units set up to check on shielding, maintenance and operating procedures.

f) The plant utility systems have been designed for loads somewhat in excess of the plant design. Dual systems are provided in critical areas.

g) Consideration for the prevention of critical incidents are discussed.

Ventilation

6.3 The several ventilation systems are of two types, those for general area ventilation and those for handling off-gases. There are four area ventilation systems; one for the Office Building, one for the Process Building and Laboratories, one for the FRS and one for the Cold Chemical Makeup Area. There are three off-gas ventilation systems; one to handle dissolver off-gases, one for process vessel off-gases and one for the high level waste storage tanks. Schematics of the area ventilation systems are to be found in Figures 6.3a and 6.3b. A schematic of the dissolver off-gas system and the process vessel off-gas system is shown in Figure 6.3c. The dissolver off-gas system is described in Paragraphs 4.26 through 4.28. The high level waste tank off-gas system is shown schematically in Figure 6.3d.

6.4 The design philosophy of the systems incorporates the following points:

a. The total volume of air and gases entering the systems is held to the least possible amount consistent with comfort of personnel and integrity of operating equipment.

b. Fresh air entering the systems is filtered, conditioned for special uses, and distributed to normally occupied spaces. Distribution to controlled and normal access areas is so designed that air flow cascades from the less active to the more active areas, thus preventing entry of air from the more active to the less active areas. Process areas are maintained at the least pressure consistent with safe practice and economy.

Figure 6.3a

P&ID Controlled Ventilation System to Elevation 131'

Drawing 4413 15R-A-74 Revision 2

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Figure 6.3b

P&ID Controlled Ventilation System Above Elevation 131'

Drawing 4413 15R-A-75 Revision 2

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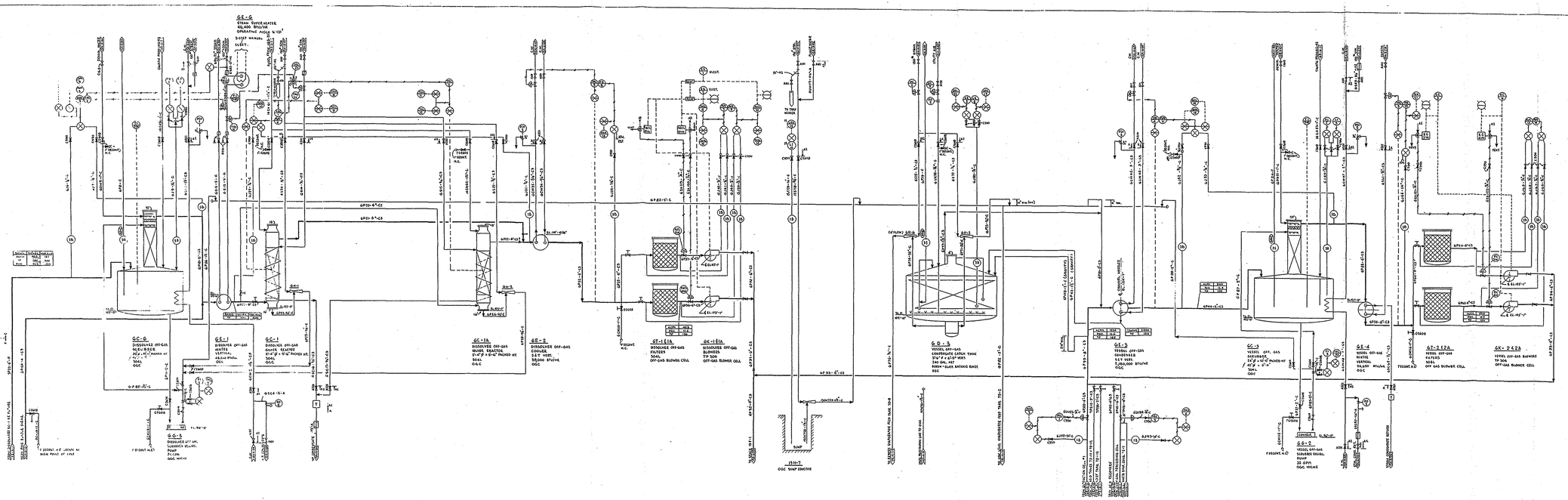
Figure 6.3c

PI&FD Process Off-Gas and Vent System

Drawing 4413 6R-A-1 Revision 5



NOTE
FOR P.&I.D. LEGEND & GENERAL NOTES
SEE DWG. 15J-A-2A

DOCKET NO. 50-201

A	AS-BUILT REVISION PER ENG. CH. 1-4-6				
A	AS-BUILT REVISION PER CHANGE				
A	PRELIMINARY FOR CONSTRUCTION				
A	DESIGN REV. 12/28/66				
A	ISSUED FOR PRELIMINARY REVIEW				
A	DESIGN REV. 12/28/66				
A	ISSUED FOR PRELIMINARY REVIEW				
A	REVISED FOR PRELIMINARY REVIEW				

BECHTEL CORPORATION
SAN FRANCISCO

NUCLEAR FUEL SERVICES, INC.
SPENT FUEL PROCESSING PLANT

PIPING, INSTRUMENT / FLOW DIAGRAM
PROCESS OFF-GAS & VENT SYSTEM
SPENT FUEL PROCESSING PLANT

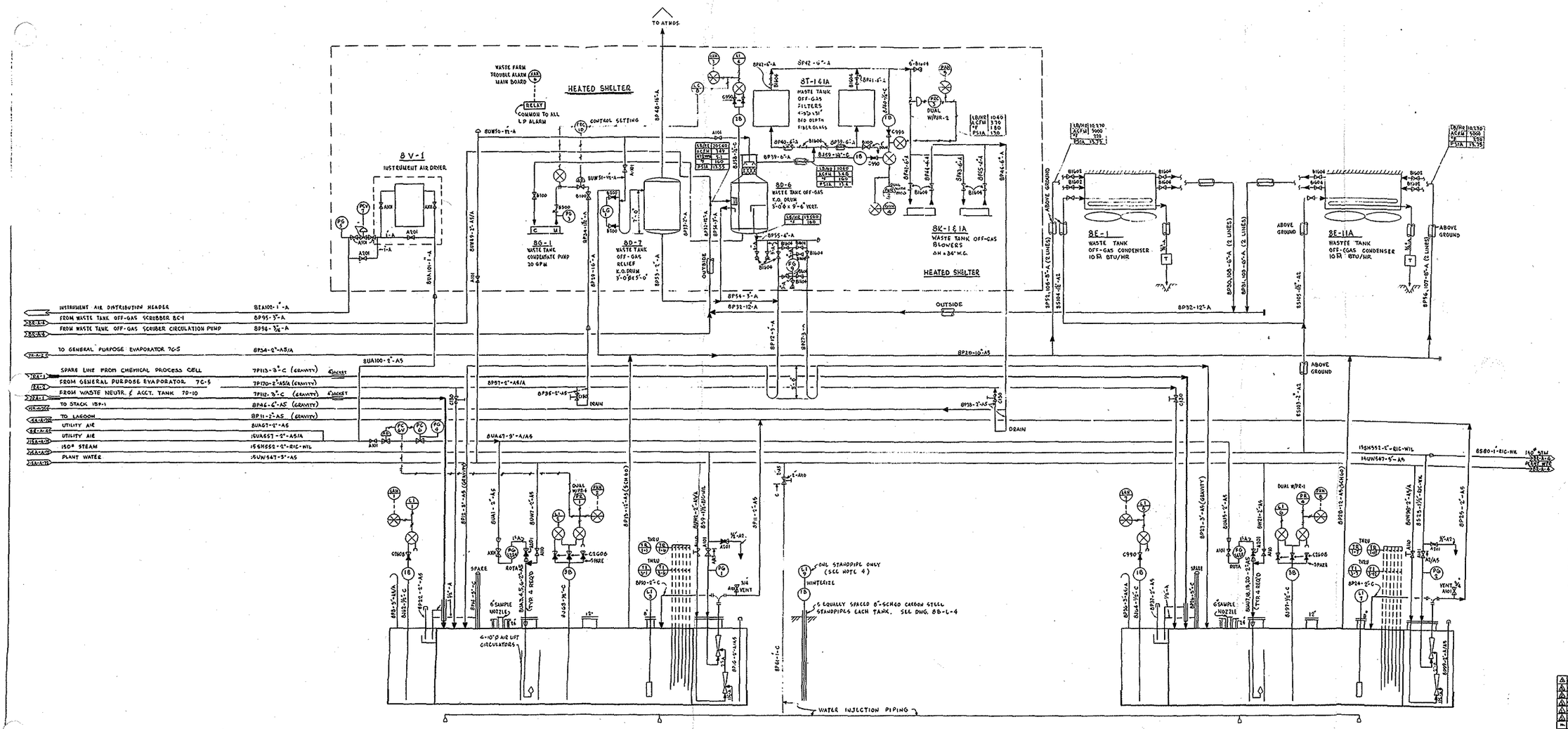
4413 GR-A-1

3540

Figure 6.3d

Flow P&ID Waste Tank Farm

Drawing 4413 8R-A-1 Revision 5



NOTE

1. FOR P&ID INDEX, LEGEND & TYPICAL DETAILS SEE DWG. 1GA-A-D6
2. PANEL MOUNTED INSTRUMENTS ARE ON LOCAL PANEL EXCEPT AS NOTED.
3. ALL INSTRUMENTS ON THIS DRAWING CARRY PREFIX "8" EXCEPT AS NOTED.
4. ONE STANDPIPE TO BE FITTED WITH DIRECT READING TOP MOUNTED LEVEL GAGE POSITIONED TO PROVIDE EASY READING FROM PLANT WASTE VALVE STATION.

A	8-10-66	REVISED TO INCLUDE COMMENTS				
B	5-12-66	ISSUED FOR CONSTRUCTION				
C	1-18-65	ISSUED FOR APPROVAL				
H	7-17-64	ISSUED FOR ADD. I				
G	12-1-64	ISSUED FOR DESIGN SCOPE				
F	11-20-64	ISSUED FOR DESIGN SCOPE				
No.	DATE	REVISIONS	BY	APP.	DRAWN	DESIGNED

SCALE NONE — DESIGNED — DRAWN

BECHTEL CORPORATION

NEW YORK — SAN FRANCISCO — LOS ANGELES

ENGINEERING BY BECHTEL ASSOCIATES

NUCLEAR FUEL SERVICES, INC.
SPENT FUEL PROCESSING PLANT

FLOW, PIPING & INSTRUMENT DIAGRAM
WASTE TANK FARM
SPENT FUEL PROCESSING PLANT

JOB No. 4413 — DRAWING No. 8R-A-1 — REV.

DOCKET NO 50-201

c. Conventional air recirculation methods have been avoided to facilitate cascading flow in order to prevent activity buildup.

d. Normal access openings between areas of differing activity potential are either isolated from other areas by means of air locks or are ventilated from the less active to the more active at velocities sufficient to preclude intrusion of air or gases from the more active areas into the less active areas.

e. Gases vented from process and laboratory equipment are, insofar as practical, maintained in segregated systems to allow special treatment, and monitoring is provided on the total off-gas prior to its release to the atmosphere.

f. Elimination of toxic and radioactive aerosols is accomplished in a manner so as to produce the least volume of of residue requiring isolation or subsequent treatment.

g. Final exhaust to the atmosphere is accomplished in sufficient volume to insure the proper dilution of irremovable gases and at sufficient velocity head to insure secondary dilution and distribution in the atmosphere.

6.5 Office Ventilation

The Office Building has a conventional and separate ventilation system completely isolated from the process building except that the air supplied to the laundry, the hot lobby and change rooms from the office air supply is exhausted into the process building exhaust system. The balance of the Office Building exhaust air does not go out through the stack.

6.6 Process Building Ventilation

Fresh air is filtered, conditioned and distributed at an average rate of 10 air changes per hour on a once through basis to the least active areas which includes the control room, stairways, analytical aisles, and the process building offices. Air from these areas is distributed through adjustable louvers and diffusers to areas slightly less inactive. These areas which include the operating aisles and laboratories are maintained at a slightly negative pressure of -0.1 inch WC compared to a slight positive pressure in the areas providing the air intake. Operating and other aisles are ventilated at a nominal average rate of 4.5 air charges per hour without recirculation. The exhaust air is drawn through adjustable registers and distributed into more active areas. These areas which include the warm equipment aisles and sample aisle are maintained at a somewhat lower pressure of approximately -0.3 inch WC. The air from these areas is exhausted into the process building exhaust system. This exhaust is routed through back-pressure control dampers to protect these areas from backflow due to a sudden increase in pressure in the exhaust system from any source. The building exhaust goes through a train of prefilter, absolute filter and exhauster which are in duplicate.

6.7 The laboratories are supplied with 10 air changes per hour, some of which is exhausted as described in Par. 6.6 but most of it exhausts through the fume hoods. The fume hoods exhaust to the cell exhaust system which is part of the building exhaust

system except that before it reaches the building exhaust the air from this system passes through an air washer. The fume hood exhaust enters the cell exhaust system through a back-pressure control damper to prevent blow back. After it is washed it is heated to prevent condensation and subsequently enters the process building exhaust system.

6.8 The mass spectrometer laboratory, emission spectrograph laboratory and the counting room are air conditioned by conventional means. They exhaust into the Process Building exhaust system through a back-pressure control damper.

6.9 The process cells are isolated from access areas with air locks. They are maintained at negative pressures ranging from -0.5 to -1.0 inch WC by exhausting the inleakage from adjacent areas to the process building exhaust system. The exhausts from the remote cells are provided with roughing filters and routed through the air washer. The exhausts from contact maintenance cells and glove boxes ordinarily bypass the air washer but may be routed through it as required. Sufficient air is passed through the analytical cells, off-gas cell and acid recovery cell to remove process heat so that the average cell temperature does not exceed 120 F. In the PMC, GPC and CPC process heat is removed so as not to exceed 120 F by circulating cell air through remotely replaceable in-cell coolers.

6.10 The principal source of dry particulate material occurs at the fuel shear. The shear is essentially a closed system with an inert gas flow directed into the fuel chute. The exhaust from the fuel chute is filtered through a readily replaceable stainless steel

cartridge filter followed by a cartridge type carbon filter. The gas then exhausts to the cell exhaust system.

6.11 The Maintenance cells, pump niches, decontamination rooms, and scrap removal area exhaust into the adjacent process cells and are thus maintained at negative pressures between -0.4 and -0.5 inch WC. Access to the maintenance cells is through air locks. During occupancy they are ventilated on a non-recirculating basis at an average nominal rate of 4.5 air changes per hour by opening supply dampers from adjacent low activity areas. This air is exhausted into the adjacent process cell. The equipment decontamination room and scrap removal area maintain a relatively large inflow of air through truck access doors when they are open. These doors are not opened when hatches or shielding doors to the adjacent process cells are open.

Fuel Receiving and Storage Ventilation

6.12 The fuel pool area is provided with a filtered air intake and heater which normally exhausts through roof exhausters. These exhausters are equipped with isolation dampers which close when the exhauster motor shuts off. The area can be exhausted into the stack at a rate that provides reasonable and practical velocities through the carrier access doors. A duct is provided that exhausts to the air washer. A portable hood can be attached to provide area exhaust to the air washer.

Cold Chemical Ventilation System

6.13 The cold chemical makeup area is provided with filtered fresh air and a heater. This area exhausts through roof ventilators.

Off-Gas Ventilation Systems

Process Vessel Ventilation

6.14 The various process vessels and equipment pieces are connected to a separate vent header and ducted to a scrubber, then to a filter and exhauster which are in duplicate to permit uninterrupted flow during filter change out. This arrangement is shown in Figure 6.3c. The exhaust connects to the stack.

Dissolver Ventilation

6.15 The dissolvers each vent through a total reflux condensor then the off-gases duct through a common header to a caustic scrubber which removes acid fumes. The scrubber is followed by a silver reactor to remove iodine after which the gases are filtered and exhausted to the stack. The silver reactor, filter and exhauster are in duplicate as shown in Figure 6.3c.

Tank Farm Ventilation

6.16 The off-gas from the waste tanks passes through a knockout drum to remove entrained liquid. It then passes through a fiber glass filter followed by a blower and these are paralleled with identical units as shown in Figure 6.3d. The exhaust line from the blower connects to the stack.

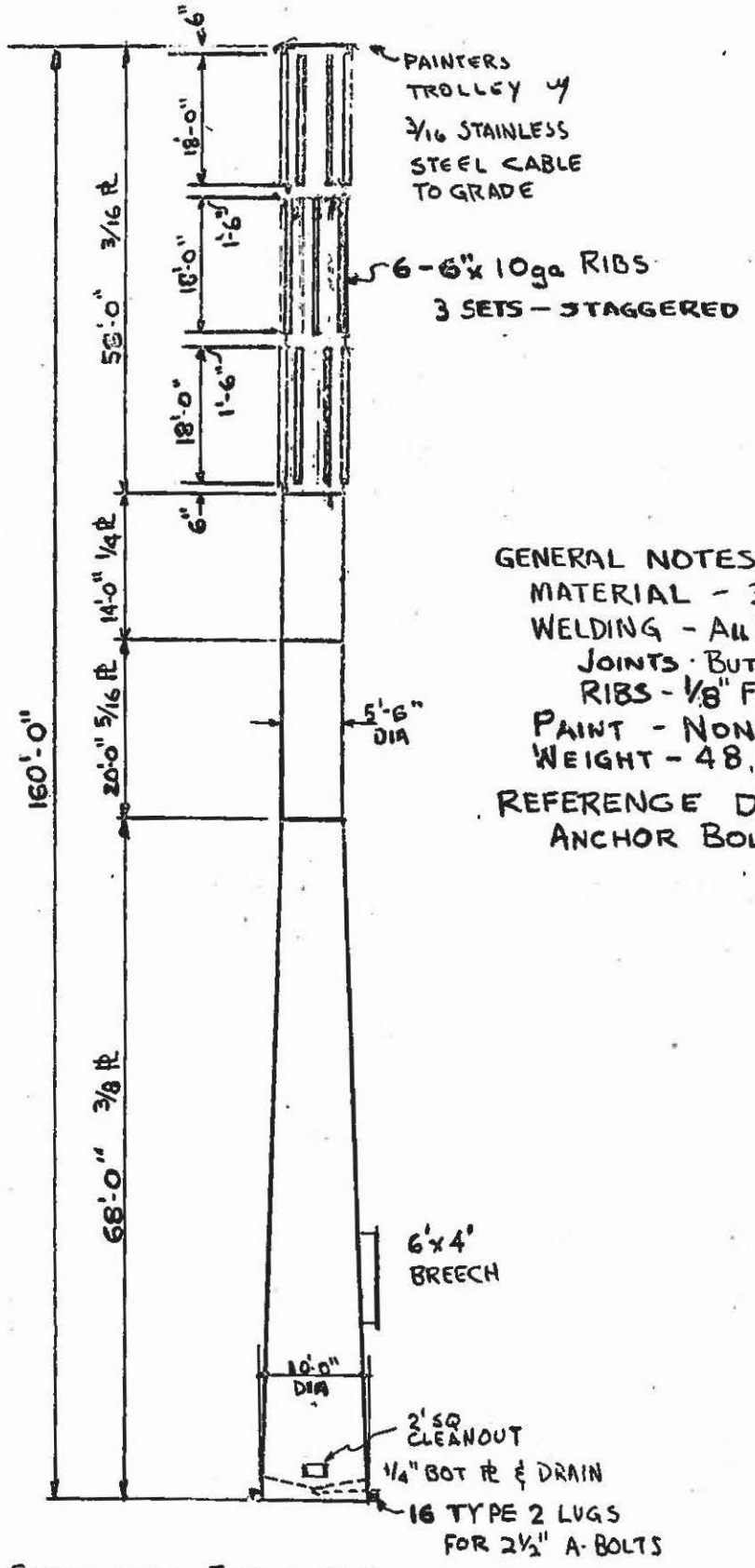
Stack

6.17 The ventilation stack extends to an elevation of 202 feet above building grade. It is a self supporting, free standing, stainless steel structure. Provision is made for decontamination by means of spray nozzle rings at various levels. A sketch of the stack is shown in Figure 6.17. Calculations of the dilution capabilities of the stack are given in Sections VII and VIII.

Figure 6.17

Off-Gas Stack

Drawing 4139 15 CSK-F-1



GENERAL NOTES:

MATERIAL - 304 L STAINLESS STEEL

WELDING - ALL LONGITUDINAL & CIRCUMERENTIAL JOINTS · BUTT WELDED

RIBS - 1/8" FILLET 2" @ 10" C/C EACH SIDE

PAINT - NONE

WEIGHT - 48,000#

REFERENCE DRAWING

ANCHOR BOLT LUG & BASE PL C-C-510

SCHEMATIC ELEVATION

△								
△								
△								
△	DATE 6/1/61		DR. F. JONES	CH'K.	DR. SUP.	ENG.	CHIEF ENG.	CLIENT

BECHTEL CORPORATION

NUCLEAR FUEL SERVICES INC

OFF-GAS STACK 15-F-1

JOB No. 4139

DRAWING No. 15CSK-F-1

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FORM E-5

Control at the Exhaust Systems

6.18 Cells are maintained at pressures lower than adjacent areas by controlling the pressure in the Process Building Exhaust System into which the ventilation exhausts terminate. This is accomplished (a) by providing back-pressure control dampers in all ventilation exhausts except cell exhausts. (b) by providing an emergency air washer by-pass pressure control damper. This control damper is normally closed but will open at sudden or large air surges exhausting from any cell. (c) by common automatic pressure-sensor control of the exhauster inlet vanes to maintain a constant negative pressure. More active areas are maintained at pressures lower than less active areas by cascading from the less active areas and adjusting air flow at inlets only. Back pressure control valves can be set at the desired pressure in the more active area and will automatically adjust for an increase or decrease in flow. In the operating aisles, no release of radioactive or toxic material is contemplated. However, should such contamination occur, the supply and return ducts to each of the general areas can be closed down to isolate the area of occurrence and allow decontamination of the area without contaminating the entire system. In more active areas, where the possibility of contamination is contemplated, manual duct dampers are installed to allow increasing the flow of air toward an area which is contaminated, as required.

6.19 Under normal operation, one exhaust train consisting of a prefilter, absolute filter, and exhauster is on stream and the duplicate train is isolated by means of butterfly valves. Pressure sensors in the suction plenum provide alarm signals at pre-determined pressures

and at the same time provide switching to actuate and engage the duplicate equipment train. The differential pressure across each filter bank is measured and high or low differential pressures will actuate alarms and start up the duplicate equipment train. Normal operation will start on one train with clean filters. It will continue on this train until the prefilter is loaded to the normal level, as indicated by the local pressure gages and by the recorded differential pressure at the control console. As a filter loads up, the fan load will increase to satisfy this pressure requirement until the control limits are exceeded, at which time the alternate train is started.

Emergency Operation

6.20 The operating exhaust fan in each system is provided with emergency means of operation during a power failure and in such an event all ventilation systems would be restored within fifteen seconds. Automatically actuated butterfly valves in the exhaust trains are equipped with hand operators and position locks which may be individually operated when the linkage to the operator is disconnected. The parallel exhauster can be started and brought on the line manually. Vanes and valves can be driven to the open or closed position manually, regardless of the signals from sensors.

6.21 The normal air supply is fresh intake air. The supply fan pushes this air across an automatic roll-type filter bank. Should the supply drop due to an excess pressure differential across the filter or due to power failure, a gravity type damper connected to the atmosphere will open to maintain supply air flow to the operating aisles.

The failure of the exhaust filter bank would be signaled by a sudden decrease in the differential pressure or an increase in the reading of the radiation stack monitor, either case actuating an alarm. The alternate exhaust train would take over and the failed exhaust train would be automatically isolated from the system.

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Figure 6.25

Glove Box Sampling Stations

Drawing 15B-T-58

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Figure 6.26

"C" Type Sample Cell

Drawings 15A-L-57
15A-T-59
15A-T-60

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Figure 6.27

Process Sampling System

Drawing 15R-A-82

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Figure 6.28

Mechanical Flow Diagram
Analytic and Sampling Operations

Drawing 15R-A-186

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Figure 6.31

Floor Plan and Schedules
Laboratory Area Process Building

Drawing 41-A-R-11

Revision 1, May 30, 1964

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Figure 6.32

Equipment Arrangement and P & ID's
For Analytical Cells

Drawings 15A-A-181,
182
183
184
15R-A-201

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Table 6.36a

Accountability Sample Summary

The uranium and plutonium "input" in a processing lot shall be determined by adding the quantities present in each batch of dissolver solution, the measured quantities in undissolved residues, and the quantities present in the processing system before processing of such lot.

The uranium and plutonium "output" in a processing lot shall be determined by adding the quantities in each batch of product, waste streams, and residue for such lot, and by adding to the sum of such quantities the amount remaining in the processing system after the processing of such lot is complete, minus the amount present in the processing system before processing of the lot was begun.

The "material balance" in per cent is the plutonium or uranium output divided by plutonium or uranium input x 100.

A processing lot will be composed of several batch dissolutions. Accountability analyses are proposed on a batchwise basis for dissolver solutions and dissolver residues. Batch identity loss occurs in flowing streams; therefore, waste and final product accountability analyses will be made as necessary throughout a processing lot to determine material balances.

Revision 1, May 30, 1964

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Shielding

6.59 The primary protection for the plant personnel from penetrating radiation is the shielding interposed between the radioactive material processed in the plant and the plant operators. Shielding materials used include water, steel, lead, concrete, heavy concrete, and earth. For the purposes of design the plant has been divided into three zones as follows:

Zone 1, Normal Access Area

This is an area in which there may be occupancy at any time during the work week. Exposure levels will be known, controlled, and recorded. The normal radiation levels in these areas will not exceed 1.5 mr/hour. This zone begins at the exclusion fence for the plant site area and includes, except as specifically posted, the area and yards surrounding the plant proper, the waste tank farm area, the burial grounds, storage lagoon, and all parking areas. In the plant building this zone includes all areas which are or may be occupied during full operation.

Zone 2, Limited Access Area

This is any plant working area which requires routine access for inspection or maintenance but which is not continuously occupied. The radiation levels in this zone will not exceed 100mr/hour. This zone is set up to cover situations which are expected to be temporary.

For instance, during the removal of hulls, scrap, or contaminated equipment from the plant and the transfer of these to the waste burial area, it is expected that plant areas will have to be marked off as limited access zones.

Zone 3, Controlled Access Area

This is a plant area which normally has no access at all but which, on occasion, must be accessible for short periods of time under carefully controlled conditions. The

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Figure 6.67a

General Locations of Radiation Monitoring and Sampling Systems

Drawing 4413 15A-J-2

Revision 1, March 16, 1964

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Figure 6.67b

General Locations of Radiation Monitoring and Sampling Systems

Drawing 4413 15A-J-16

Revision 1, March 16, 1964

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Figure 6.67c

General Locations of Radiation Monitoring and Sampling Systems

Drawing 4413 15A-J-17

Revision 1, March 16, 1964

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Figure 6.67d

General Locations of Radiation Monitoring and Sampling Systems

Drawing 4413 15A-J-18

Revision 1, March 16, 1964

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Figure 6.67e

General Locations of Radiation Monitoring and Sampling Systems

Drawing 4413 15A-J-19

Revision 1, March 16, 1964

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Figure 6.67f

General Locations of Radiation Monitoring and Sampling Systems

Drawing 4413 15A-J-20

Revision 1, March 16, 1964

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Figure 6.67g

General Locations of Radiation Monitoring and Sampling System

Drawing 4413 15A-J-35

Revision 1, March 16, 1964

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Table 6.67a

Health and Safety Equipment

1	Tracerlab remote radiation alarm system with 15 channels.
1	Tracerlab local radiation alarm with 5 channels.
1	Tracerlab stack particulate and iodine-131 monitor.
4	Tracerlab liquid in-line monitors.
2	Eberline beta-gamma hand and foot counters.
2	Science Associates weather monitoring stations.
2	Stream Gauging and sampling stations.
24	Nuclear-Chicago beta-gamma personnel station monitors.
18	Nuclear-Chicago alpha personnel station monitors.
4	Eberline portable gas proportional alpha counters.
1	Eberline gas proportional alpha floor monitor.
4	Victoreen Thyac II GM survey meters.
1	Nuclear-Chicago deep hole monitor.
1	Eberline beta-gamma floor monitor.
8	Technical Associates CP survey meters.
2	Technical Associates CPTP high range survey meters.
1	Victoreen low energy survey meter.

Revision 1, March 16, 1964

Table 6.67a (cont'd)

1	Nuclear-Chicago portable neutron survey meter.
	Nuclear-Chicago film badge service.
400	Nuclear-Chicago indirect reading gamma dosimeters.
1	Nuclear-Chicago charger-reader.
25	Nuclear-Chicago direct reading gamma dosimeters.
1	Nuclear-Chicago charger.
1	Thermoluminescent dosimeter readout.
30	Lithium fluoride dosimeters.
3	Tracerlab environmental air monitoring systems.
1	Gelman Nuclear Air sampler.
73	Station in-plant sampling system
1	Nuclear Measurements programmed alpha continuous air monitor.
1	Nuclear Measurements fixed filter alpha continuous air monitor.
5	Nuclear Measurements fixed filter beta-gamma continuous air monitor.
1	MSA Universal Testing Kit for toxic fumes in air.
1	MSA gascope for detection of natural gas in air.
1	Nuclear-Chicago thyroid monitor.
1	Nuclear-Chicago plutonium gamma monitor.
1	Set New England Nuclear calibrated gamma disc sources.

Revision 1, March 16, 1964

Table 6.67a (cont'd)

1	Set New England Nuclear calibrated simulated iodine-131 sources.
3 ml	Tracerlab calibrated cesium-137 solution.
1	Tracerlab calibrated 10 millicurie cobalt-60 source.
3	Tracerlab calibrated strontium-90 disc sources.
1	Set Eberline calibrated plutonium standards.
3	Eberline calibrated plutonium standards.
1	Simpson wide band oscilloscope.
1	Simpson VOM.
1	Atomic Accessories pulse generator.

Table 6.67b

Health and Safety Counting Room Equipment

1	Packard automatic liquid scintillation counting system.
1	Nuclear Measurements automatic scan gamma pulse height analyzer system.
3	Nuclear-Chicago automatic low background sample counting systems.

6.69 An area air particulate sampling system is provided to sample on a continuous basis air in all the operating areas for possible contamination. The system is powered by a vacuum pump with piping system terminating in 54 open filter paper holders. A flow meter is provided at each sample station to set the flow air through the filter paper. The sampling stations are located in operating, equipment and sample aisles and other occupied areas as shown in Figures 6.67a through 6.67f. Space is provided for a spare central vacuum pump and the pump is on the emergency power system to insure air sampling in all areas. Health and Safety personnel will change and analyze the filters for radioactivity as described in Section IX.

6.70 A cell air particulate sampling system is installed to permit sampling of in-cell atmosphere for air-borne particulate contamination. This system will be useful as a prelude to maintenance and inspection. At each cell a one-inch diameter, type 304 stainless steel tube penetrates the cell wall. The tubes are suitably placed considering radiation streaming and are sloped downward into the cell to facilitate self-draining during cell decontamination. On the outside wall of the cell the tubes connect through a closed filter paper holder and air flow meter to the central vacuum system which exhausts into the building ventilation system. Each sampler is double-valved so that it may be turned off when not in use. These 19 sample stations can be located in Figures 6.67a through 6.67f.

6.71 A stack gas monitoring system is provided to sample the gaseous effluent of the plant operations. The stack gas monitoring system can be located on Figure 6.67e. The stack monitor consists of a two-channel monitoring system for particulate and gaseous iodine-131 activity. Particulate monitoring is done by means of a moving tape filter and an end window GM tube. The filter tape transport is based on the Brookhaven design. A solid capstan with milled slots rides on a Teflon shear valve which limits the air bypass around the filter paper to less than 2 per cent. The filter paper is held against the rotating capstan by the pressure differential across the paper and is moved by the rotation capstan. The detector is positioned to view the particulates at the deposition point providing immediate response to radioactivity. The detector is shielded by two inches of lead. The detector count feeds into a ratemeter with switch selected three decade (10 to 10^4 CPM) scale on a $4\frac{1}{2}$ inch panel meter. Meter accuracy is $\pm$ 2 per cent in terms of meter current. Ratemeter output is to a two pen chart recorder mounted at the central control panel. The recorder is equipped with an adjustable alarm point

for each pen and an audible alarm. The iodine-131 monitor consists of a fixed charcoal filter which receives air from the particulate filter. The charcoal filter is monitored by a gamma scintillation detector shielded by three inches of lead. The detector output is to a ratemeter as described above with the addition of a spectrometer unit to selectively count the gamma radiation from iodine-131. Ratemeter output is to the chart recorder described above. The particulate monitor will detect 10^{-11} $\mu c/cc$ mixed fission products but will be set to alarm at 10^{-10} $\mu c/cc$. The iodine monitor can be set to alarm within 30 minutes at a concentration of 1×10^{-10} $\mu c/cc$. The actual alarm point will be set in the field after obtaining experience with it.

6.72 Two weather monitoring systems are provided which automatically record wind direction, velocity, and temperature. One of the systems is located near the base and one near the top of the stack. A 2-channel liquid in-line radiation monitoring system is provided; one channel for the effluent to the seepage basin and one channel for the cooling water return line to the cooling water tower. These are described in Section IX and can be located in Figure 6.67b. These monitors act as a final check for protection of the public in the operation of the storage lagoon and the cooling water tower. Two other liquid in-line monitors are used to check the weak acid from pump 7G-1 and the condensate line to the utility room. These systems are described in Section IX and can also be located in Figure 6.67b.

6.73 Six continuous air monitors are provided. One of these units is installed in the product packaging and shipping area specifically for detection of air-borne plutonium contamination. Four of these units are to detect particulate and gaseous fission products and are located in the fuel storage building, the hot lobby, the mechanical operating aisle and the general purpose cell operating aisle. The sixth unit, an alpha, beta, gamma monitor is located in the analytical aisle. These units are described in Section IX and can be located in Figures 6.67a, 6.67b, and 6.67d.

6.74 Three site perimeter environs radiation monitoring stations are provided to detect and record the possible accumulated contribution to background radiation of uncontrolled public areas through the operation of the NFS plant facility. It is described in Section IX and may be located in Figure 6.67g. These stations are self-contained except for electrical requirements and are sheltered from the weather. They will be serviced weekly and furnish a continuing record of the effect of plant operations on the environs.

6.75 A stream monitoring system is provided to gage creek flow and continuously sample creek flow of Cattaraugus and Franks Creek. The stations are installed to provide a continuing record of the contribution, if any, of NFS plant operations to uncontrolled public areas. They will be serviced weekly.

6.76 A fixed in-cell area radiation measuring station system is provided to permit intermittent inspection and detection of gamma background radiation inside the operating cells. This system is composed of 21 separate stainless steel tube penetrations into the cells, suitably sized for intermittent reception of the deephole probe. On the inside, the tubing extends 6 inches into the cells. In the PMC, the inside end of the tubing terminates in suitably remotely removable end caps. In all other cells the tubing terminates with fixed end caps. On the outside surface of all cell walls, the tubing terminates approximately flush with fixed end caps. On the outside surface of all cell walls, the tubing terminates approximately flush with the cell wall and is equipped with suitable removable caps. Radiation streaming is considered in the tubing placement. The system is located in Figures 6.67a through f.

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Export Controlled Information

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Export Controlled Information

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Export Controlled Information

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Utility Section of the Safety Analysis

Utilities

6.93 The plant utility systems are summarized in Table 6.93.

6.94 The plant water supply is taken from two lakes which have been created by the construction of two dams near the south end of the site. The two interconnected lakes created by damming up the watershed run-off from about 3500 acres of land, contain 1009-acre feet of water in 25 acres. The pump house which contains two 400-gpm pumps is located just inside the northernmost dam and is connected to the plant by 6000 feet of 8" pipe along the railroad spur. During normal operation the pumps are operated alternately through an alternator switch actuated by a timing device. The pump not in operation drains to prevent freezing. Power to the raw water pumps is not furnished by the plant electrical systems. In the event that power should fail at the pumping station, it would be noted by loss of line pressure at the plant. History shows that power failures are of extremely short duration, but should an extended power outage be indicated in freezing weather, the line can be drained manually.

6.95 A clarifier is installed at the plant to receive the raw water, treat it with chloride of lime, soda ash, and alum as is required. The effluent, 400 gpm, is then filtered and stored in a 475,000 gallon storage

Table 6.93

Summary of Utilities

System	Units	Normal Demand	Installed Capacity
Raw Water Supply	gpm	220	400 Continuous 800 Intermittent
Raw Water Treatment	gpm	220	400 Continuous
Domestic Water - Process and Administration Area	gpm	150	150
Demineralized Water	gpm	16	16 Continuous 32 Maximum
Cooling Water - Open System	gpm	1785	2250
Heat Transfer	Btu x 10^6	29.3	31
Steam Generation	lb/hr x 10^3	50	80
150 psig	lb/hr x 10^3	11	-
25 psig	lb/hr x 10^3	44	-
Fuel Gas	Btu/hr x 10^6	52	110
Fuel Oil	gpm	0	15
Compressed Air	scfm	1400	2800
Plant Air	scfm	850	2200
Instrument Air	scfm	550	600

tank. The filtered water storage tank holds 300,000 gallons in preferential storage for the firewater system and the remaining 175,000 gallon storage volume is for the plant systems. The plant water is furnished by two-250 gpm pumps boosting the water pressure to a minimum of 75 psig.

6.96 The domestic water system is allocated 150 gpm from the plant system and is chlorinated for potability as the water is delivered to a 1000-gallon accumulator tank. A storage heater takes water from the accumulator and is capable of processing 1710 gph of 140 to 180F hot water available to the domestic system. Cooling tower makeup is taken from the plant system; about 50 gpm is required at present and 75 gpm is allocated for the future makeup requirement. The demineralized water system will normally make 16 gpm of demineralized water and may make 32 gpm maximum makeup to the 18,000-gallon demineralized water storage tank.

6.97 The cooling water system is an open cooling tower presently operating at 2250 gpm making 80F cooling water available from 115F water returned to the tower. A two-cell induced draft, counter-flow tower is installed with a basin suitable for 4500 gpm future flow requirements. Separate chlorination and other chemical feed equipment is installed to support this system.

6.98 Two 40,000-lb/hr boilers are installed to provide the present requirement of approximately 50,000 lb/hr and provision for a third 40,000-lb/hr boiler is made for the future total requirement of approximately 75,000 lb/hr. In the case of a power failure provision has been made for a safe and orderly shutdown of the plant through the use of low

pressure steam driven turbines and auto-start facilities. Utility commodities such as the boiler auxiliaries, cooling tower circulating pump, process building exhaust fans, emergency generator, air compressors, condensate return pump, plant water pump, etc. are provided with emergency turbine drives. In the event that a power failure should occur, an emergency auto-start, turbine driven generator is provided to furnish electric power to the ventilation system and minimal lighting. Total low pressure steam requirements are available through a pressure reducing station. Condensate is collected at the steam users, monitored for radioactive contamination in the common line to either of the two 17,500 gallon condensate receiver tanks. The tanks are used alternately and the condensate is analyzed for quality before it is pumped back to the 80,000 lb/hr deaerator to be reused for boiler feed. Should the condensate stream be contaminated it will be diverted to the waste treatment plant for disposition.

6.99 An inert gas generator fired with propane gas and suitable for 3,000 scfh is provided together with a compressor to boost this gas pressure to 80 psig for distribution and use. An LPG storage system is required to furnish fuel to the inert gas generator, and on the basis of fifteen days supply on hand at normal use rate, a 3,000-gallon total capacity (1375 gallons midsummer) propane bullet is furnished together with required regulating equipment and piping.

6.100 Natural gas is available from the Iroquois Gas Company at the site and a 150,000-cfh supply is being negotiated for total future plant requirements.

6.101 A storage tank of 10,000 gallons capacity is installed for the diesel oil requirements of the boilers and the engine driven fire pump. Required transfer pumpage, day tanks, and piping are installed.

6.102 Two 300-hp compressors are installed to furnish the instrument and plant air requirements for the present and future requirements. Each compressor, one motor driven and one turbine driven, has a maximum capacity of 1400 scfm at 100 psig and each is of sufficient capacity to furnish total air requirements. The compressors are of nonlubricated construction and as such the compressed air supply is suitable for direct process injection and/or breathing air without treatment. An instrument air dryer is furnished for 600 scfm at 90 psig and 0F dew point for the instrument requirements.

Submission No. 22--Final Safety Analysis Report

NFS Operating License Application

AEC Docket No. 50-201

Section on Criticality - Paragraphs

6.103 - 6.167 and Appendix

October 31, 1964

Criticality

6.103 The steps being taken to avoid a criticality event anywhere in the NFS plant are detailed in this section. The design of the plant detailed in this document is such as to reduce the probability of a criticality event anywhere in the plant to an acceptable minimum. Even so, sufficient shielding is provided so that, if a criticality incident should occur, the health and safety of the general public and of the employees would not be endangered. (See Paragraphs 7.30 through 7.34 and 8.24 through 8.29.)

Criteria

6.104 The criteria chosen to provide criticality control in the NFS plant have been taken from Nuclear Safety Guides, TID-7016, and from experience with other facilities including the NFS plant at Erwin, Tennessee, and various AEC plants and laboratories. The calculations and conclusions reached are supported by published guides and experimental data and these are referenced herein. *

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* References for this section follow Paragraph 6.166.

6.105 Calculations are based upon the assumption that the fissionable isotope content and enrichment of each fuel element processed is that of the original element before irradiation. However, for all plant fuels the reactivity level at the time the fuel is introduced into the processing operation is less than that of the original fuel assembly before irradiation. This is shown in Table 6.105. An additional factor of safety, in addition to factors discussed below, is therefore available.

6.106 Criticality is controlled by various methods, some of which include geometry, concentration, mass and poisons either liquid or solid. With the exception of the first, some degree of administrative control is generally associated with each of these. Throughout the plant various combinations of these methods are used. Except where it is possible to use geometric control, two or more independent safeguards are operative at every step.

6.107 Wherever concentration control is the primary safeguard, the concentration permitted is 50 to 60 per cent of that calculated to be the minimum critical concentration for the system involved depending on the measure of control provided at each step. This provides a large safety factor in each step.

6.108 Wherever the primary control is other than geometry, process control instruments and other administrative procedural controls are provided to assure that the calculated conditions are maintained. Process control instruments in many process steps give immediate indication of any change in conditions. These are backed up with instruments and alarms, where needed, for purposes of criticality control only. Solution

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analyses are an invaluable aid to positive control and are made use of throughout the process. Administrative procedures and controls, which are the backbone of any operation, are detailed, specific, and rigidly enforced. The use of these is discussed in Paragraphs 9.5 through 9.8.

Geometry

6.109 Wherever possible, and particularly at the end of the process where there is close contact between personnel and the final product, extensive use is made of geometric control. This method does not lend itself to those parts of the process where solution volumes are large.

Concentration

6.110 Throughout most of the process the fuel is in solution in relatively large volumes. The process lends itself to close control of concentrations at all points and, thus, criticality is most easily controlled by limiting the concentration of fissionable isotopes. The selected limits are maintained by process control and analytical determination backed up by instrument controls and alarms with the appropriate limits and actions specified in the procedures.

Mass

6.111 Mass control is practiced at a few points such as some steps in FRS and PMC where only a single fuel assembly will be moved at a time or in the waste evaporators where every feed to the step has been analytically proved to have negligible concentrations of fissionable materials.

Poisons

6.112 Limited use is made of soluble neutron poisons such as boric acid as a secondary control to increase the safety factor on concentration

control in dissolution of enriched fuels and in the rework evaporator. It is also used for emergency flooding to quench a criticality incident should one occur. Recent data from Rocky Flats and ORNL have demonstrated that the use of solid poisons results in large savings in space, tank economy, and in reduced handling and maintenance. These data have been conservatively applied to make use of solid poisons as secondary safeguards for vessels which will contain normally barren solutions, to provide an increased safety factor in some areas where the primary control is concentration, and to provide practical storage for purified product solutions.

6.113 In subsequent paragraphs the process is examined from the receipt of the fuel to the shipment of the product and the disposal of the waste, and the methods used to avoid criticality at each step are explained in detail.

Fuel Receiving and Storage (FRS)

6.114 The FRS area is described in Paragraphs 3.6 through 3.10. The process steps carried out therein are described in Paragraphs 4.2 through 4.8. Fuel is brought into the plant in shipping casks which have been licensed by the AEC. Criticality control is one of the prime factors in the review of these shipping casks so that the risk of a critical incident upon cask receipt has already been reviewed and deemed acceptable. The cask is surveyed upon receipt and is placed in an unloading pool which contains 44 feet of water after which the lid is removed. Thereafter the fuel assemblies are removed with the cask unloading crane. Only one fuel assembly can be handled at a time and

there is no assembly which can go critical by itself. The assembly is placed in a storage canister which is in a temporary storage rack located in the fuel unloading pool. These racks will accommodate only one 20" X 21" canister per station and the stations are in a single vertical plane and are located 20 inches center to center which gives 12 inches minimum spacing edge to edge. The assemblies are placed vertically in the storage canisters (see Figure 5.6). Short assemblies may be stacked end to end, and fuel assemblies of small cross section may be placed adjacent to each other inside the 8 inch diameter fuel guide which centers the particular assemblies in the canister. The number of fuel assemblies permitted in the canister will be limited to that giving a Keff. of 0.85. This number is determined with the reactor owner for a specific fuel. The canisters are then fitted with guides which will permit insertion of only the number of elements prescribed.

6.115 The storage canisters are then picked up by the canister crane (see Figure 5.8a). This crane is the only one equipped with the required grappler to lift the storage canisters. Only one canister can be handled at a time. There is a limited vertical lift to this crane so that a storage canister cannot be lifted high enough to reduce the amount of water shielding to less than 11 feet or permit movement over the top of other canisters. With this crane the storage canister is moved over into the fuel storage pool and stored in the storage rack (see Figure 5.9). The pool contains no drains precluding loss of water reflection. The canister is brought into the storage pool through an aisle and then moved at right angles into the desired empty storage space in the desired row. There are spacers on the outside of the

storage canisters which provide a 20 inch diameter. Fuel assemblies are supported in the center of the storage canisters by guides. Thus, it is physically impossible to have the center of any fuel element closer than 20 inches from the center of any other. With the largest fuel elements to be processed, this will give a minimum of 12 inches edge to edge spacing in the storage racks or at any time during the movement of a basket into or out of the storage array. Thus, the attainment of a critical array with any physically possible arrangement of any fuel to be processed is deemed impossible in the fuel storage sequence.

6.116 Each position in the fuel pool storage rack is numbered as is each storage canister. As a fuel assembly is loaded into a canister and placed in the storage rack, the identity of the assembly, the customer, enrichment, uranium and plutonium content, and fuel type are entered in a log record. The canister number and storage rack number are also recorded. Before an assembly is released for processing, a letter of authorization approved as indicated in Paragraph 9.7 specifying the exact fuel assemblies to be included in a batch must be executed. A dissolver batch will be made up based on the accurately known uranium content of the fuel assemblies as originally fabricated.

6.117 One storage canister at a time is moved with the fuel pool crane and affixed to the underwater transfer conveyor (Figure 5.11). This conveyor moves the canister into position below the PMC and the fuel assembly is lifted out of the canister with the PMC crane.

Process Mechanical Cell (PMC) & General Purpose Cell (GPC)

6.118 The PMC has been described in Paragraph 3.11; the processes carried on therein are described in Paragraphs 4.9 through 4.15, and

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Product Packaging and Shipping

6.162 Plutonium solution and highly enriched uranium solution are packaged in 5-inch O.D. (maximum) polyethylene bottles not over 4-feet long. These are placed inside 5-foot sections of schedule 40 stainless steel pipe made with welded bottoms and flanged gasketed tops that seal. This system is safe fully reflected to 500 g/l Pu^7 and for all concentrations of enriched uranium solutions. These are mounted in bird cages which are approved under 10 CFR Part 71.

6.163 Low enriched uranium solution is packaged at 10 g/l U-235 or less in tank trucks of the same specifications as those used by Savannah River operations for shipment of highly enriched uranium in safe concentration.

Off-Gas Scrubbers

6.164 The vessel off-gas scrubber (6C-3) is located in the Off-Gas Cell. The vessels venting to this scrubber contain, at most, trace quantities of fissionable material except the rework evaporator which is at always safe concentration. To reach 6C-3 this vapor must first pass through the rework evaporator condenser and then through a line equipped with a high pressure alarm which sounds at an increase in pressure due to condenser failure. All of the off-gas to 6C-3 must first pass through 6E-3, the vessel off-gas condenser, which removes contained liquids. These liquids recycle to the rework evaporator feed tank. Since nothing in this routing concentrates the uranium in the vapors and since the uranium contained would be only that carried along with the water vapor by entrainment (which is 10^{-3} to 10^{-4} of that in the dissolver,) the uranium content and concentration will be much less than the always safe concentration and the system is safe.

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Export Controlled Information

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VI – PROTECTION OF THE PUBLIC

Before the

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For Licenses for a Spent Fuel Processing Plant

Under Sections 53, 63, 81, 104 (b), and 185 of the

Atomic Energy Act

AEC Docket No. 50-201

Submission No. 15 - Final Safety Analysis Report

Revision of Section VII - Protection of the Public

Revision 2, August 20, 1964

VII PROTECTION OF THE PUBLIC

Summary

7.1 The plant and process which have been described in detail in preceding sections are designed to operate so that, under all normal operating procedures, any discharge of radioactivity to the environment will be well within the limits set forth in 10 CFR Part 20.

7.2 Radioactivity can be lost from the process complex at the following points:

1. Stack
2. Waste storage tanks
3. Storage lagoon
4. Burial ground
5. Egress of personnel and material
6. Product shipment

In subsequent paragraphs, each of the above possibilities is analyzed to show that the statement of Paragraph 7.1 is valid. Some of the detailed calculations are shown in Appendicies as noted.

7.3 Further, this plant and its site are shown to be so designed and located that, in the unlikely event of the most serious accident which could possibly be deemed credible, there will be no discharge to the environment which results in levels of exposure in excess of those set forth in Sections 100.11(a)(1), (2) and (3) of 10 CFR Part 100; and further that steps can be taken to assure that, even in the event of such an accident, the discharges to surface waterways at the site boundary can be kept within the limits specified in 10 CFR Part 20 through the use of reasonable correction measures after the accident or release has occurred.

7.4 The following abnormal events have been postulated:

1. The complete rupture of a waste tank releasing 600,000 gallons of high-level waste.

2. A criticality incident anywhere in the plant involving a total of 10^{19} fissions in a single burst or a multiple continuing event totalling 10^{20} fissions.

3. A criticality incident in the fuel storage pool which sets up a 10-mwt boiling water reactor which operated for as long as 3 hours before it can be shut down.

4. A chemical explosion in the plant which is assumed to rupture a vessel containing a full day's charge of the maximum fission product content possible.

5. The complete failure of the iodine removal equipment so that for a period of up to one day the complete charge of iodine is lost to the atmosphere.

The rationale for the selection of these events has been to select for the plant, the stack, and the tank farm events which represent the upper limit of catastrophe which could occur in each of these areas, even though we believe that the likelihood of occurrence is very small. In subsequent paragraphs, each of the above possibilities is analyzed.

7.5 Throughout this section, a number of assumptions recur. Values for such recurring assumptions are collected in Table 7.5. Assumptions specifically related to a particular calculation are included in the calculation.

Normal Operations

Stack

7.6 As explained in Paragraphs 6.3 through 6.21, the ventilation systems are designed to assure that, under normal operating conditions, flow of air is always from areas of least contamination into those of higher contamination. There are separate systems for vessels, dissolvers, and the cells themselves. These join together and are filtered before discharge through a 65-meter stack. The total volume of air discharged is 32,000 cfm. Iodine removal facilities are dsigned to collect 99.5% of the incident iodine. It is assumed that all of the noble gases in a daily charge escape during the course of the day. Under normal operating conditions, the amount of solid fission products taken into the gas stream is assumed to be low enough that the filtering of this stream will reduce them to the point where they are negligible in comparison to the gaseous activity. Calculations are based on an average fuel which we may expect to process in this plant represented by the following parameters:

Table 7.5

Assumptions Used in Calculations

Sections VII & VIII

1. The dispersion parameters used are those given in Table 2.14 and in "Nuclear Safety", Volume 2, No. 4, June 1961. Figures V-1 and V-2 provide horizontal and vertical dispersion coefficients respectively for distances up to 10^5 meters and for meteorological conditions ranging from "extremely unstable" to "moderately stable". In all calculations performed in this section, "slightly unstable" coefficients have been assumed to represent average conditions and "moderately stable" coefficients have been assumed to represent inversion conditions.

   Wind velocities of 1 meter/second for inversion conditions and 4 meters/second for average conditions have been used.

   The following wind distribution data has been used:

   Wind Distribution

   (Per Cent Per Octant)

Wind Direction	Summer	Winter	Average
N	9%	8%	8.5%
NE	4	2	3
E	5	2	3.5
SE	17	9	13
S	23	21	22
SW	13	25	19
W	9	12	10.5
NW	20	21	20.5

2. Fuel is cooled 150 days before processing.

3. High-level waste is stored at 410 gallons per ton which is equivalent to:

132 c/gal	Sr-90
166 c/gal	Cs-137
57 c/gal	Ru-106

   at the time of storage.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Table 7.5 (Cont'd)

4. The rate of travel in the surficial till is 1.0 foot/day. The rate of travel in the silty till is 5×10^{-5} foot/day.

5. 90% of Sr-90 is associated with sludge in the tank.

6. 99.9% of Sr-90 is adsorbed on soil on passage through it.

7. 99.99% of Cs-137 is adsorbed on passage through the 700 feet of soil.

8. No Ru-106 is adsorbed at all.

9. Tritium is assumed to go 25% to stack, 10% to waste tanks, 65% to steam.

10. For long-lived isotopes the fission products are taken as 70% from U^{235} - 30% from Pu^{239}. For short-lived isotopes they are taken as 60% from Pu^{239} - 40% from U^{235}.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Burnup	20,000 mwd/ton
Specific Power	32 mw/ton
Irradiation Time	2 years
Load Factor	85 per cent
Cooling Time	150 days

Using these parameters the input activity to the plant was calculated. The gaseous activity* input is:

Kr-85	6.3×10^3 curies
I-129	0.022 curie
I-131	1.8 curies
Xe-131m	1.0 curies
Xe-133	3.8×10^{-3} curie
Tritium	50 curies

Under the conditions stated above, the total daily discharge from the stack using the average activity level fuel contemplated will be:

Kr-85	6.3×10^3 curies
I-129	1.1×10^{-4} curie
I-131	9.0×10^{-3} curie
Xe-131m	1.0 curie
Xe-133	3.8×10^{-3} curie
Tritium	50 curies

7.7 The concentrations of each of these isotopes at various distances and under various meteorological conditions are calculated from the following formulae:

For short-term calculations:

$$X = \frac{Q}{\pi \sigma_y \sigma_z u} \exp\left[-\frac{h^2}{2\sigma_z^2}\right] \quad (7.7a)$$

For Long-period average concentration:

$$\frac{2^{\frac{1}{2}}\, 0.01 f\, Q}{\pi^{\frac{1}{2}} \sigma_z u x} \frac{2\pi}{8} \exp\left[-\frac{h_1^2}{2\sigma_z^2}\right] \quad (7.7b)$$

* At 150 days cooling these are the only significant gaseous isotopes.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Where X = concentration in curies/m^3 ($\mu c/cc$)
Q = emission rate in curies/second
σ_y, σ_z = dispersion coefficients in meters
h = stack height in meters
u = wind velocity in meters/second
x = distance downwind in meters
f = wind frequency in per cent/octant

The calculation has been carried out for both inversion and average conditions over the range 1500 to 51,000 meters (see Appendix 7.7). The results of these calculations are presented in Table 7.7. The maximum concentrations are given for both the average and inversion conditions. For average conditions the maximum concentration occurs at the site boundary; under inversion conditions the maximum concentration occurs over the range of about 4000 to 10,000 meters downwind from the stack. It can be seen that all of the concentrations are well within the MPC values with the exception of the Kr-85 concentration under inversion conditions. The inversion concentrations given are centerline concentrations and include no wind diversity factor; they are not expected to persist for more than a few hours at a time. The yearly average concentration, which is permitted under 10 CFR Part 20, will not be significantly increased by these occurrances.

7.8 Although 10 CFR Part 20 contains no provision for limits on the deposition of radioiodine on pasturage, the plant is designed to release iodine at concentrations lower than the MPC for concentration in air in order to protect those areas surrounding the plant site which are used for dairying. Using the long-period average concentration and a deposition velocity of 0.01 meter per second, the deposition rate has been calculated (Appendix 7.8). Since yearly average concentrations are used, it is reasonable to assume that the equilibrium conditions are reached; i.e. the rate of deposition equals the rate of decay. The south, southwest and northwest octants have the highest yearly average wind frequencies, ranging from about 19 to 22 per cent. Therefore, a wind frequency of 25 per cent per octant has been used in these calculations. It was found after the Windscale incident that a grazing area contamination level of 1 μc per square meter resulted in about 0.1 μc/liter of milk*. Using this relationship the resultant activity levels in milk have been calculated. The milk activity levels are shown in Table 7.8.

7.9 The Federal Radiation Council has established a <u>Radioactivity Intake Guide for Iodine-131</u> of 100 $\mu\mu c$ per day, based on the uptake by children as the most sensitvie segment of the population. As can be seen from Table 7.8, the consumption of about five liters of milk per day from dariy cattle grazing immediately adjacent to the site boundary would be required to equal the level of intake as established by

*TID-8206, Page 56

Table 7.7

Maximum Concentration of Gaseous Isotopes Under

Inversion and Average Meteorological Conditions

Isotopes[d]	Curies/Second	X, μc/cc		MPC[c] μc/cc
		Inversion[a]	Average[b]	
Kr-85	7.3×10^{-2}	7.3×10^{-7}	1.6×10^{-8}	3×10^{-7}
I-129	1.3×10^{-9}	1.3×10^{-14}	2.8×10^{-16}	6×10^{-11}
I-131	1.0×10^{-7}	1.0×10^{-12}	2.2×10^{-14}	3×10^{-10}
Xe-131m	1.15×10^{-5}	1.15×10^{-10}	2.5×10^{-12}	4×10^{-7}
Xe-133	4.4×10^{-8}	4.4×10^{-13}	9.7×10^{-15}	3×10^{-7}
Tritium	5.8×10^{-4}	5.8×10^{-9}	1.3×10^{-10}	2×10^{-7}

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a Maximum concentration occurs at about 6000 meters from the stack; concentration within about 10% of the maximum occur from about 4000 to 10,000 meters from the stack.

b Maximum concentrations occur at the site boundary (1500 meters).

c Table II, Appendix B, 10 CFR Part 20.

d At 150 days cooling, these are the only significant gaseous isotopes.

e Based on 1 triton produced per 10^4 fissions (reported as 1 in 1 to 4×10^4) with 25% lost up the stack, 65% lost in liquid waste effluent, 10% to storage tanks.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Table 7.8

Iodine Deposition and Milk Concentration[a]

Distance in Meters	Ground Concentration $\mu\mu c/m^2$	Milk Concentration $\mu\mu c$/liter
1500	200	22
2000	150	15
5000	31	3.1
10000	8.9	0.89
20000	2.6	0.26

a See Table 7.5 for assumptions and Appendix 7.8 for detailed calculations.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

this Guide. This rate of consumption is higher than any that can be expected, probably by a factor of at least four. In addition, no credit is taken for dilution (during processing) by milk containing lesser (or no) amounts of radioiodine or the fact that cattle are pastured in Western New York State only about half the year.

Waste Storage Tanks

7.10 The design of the waste storage tanks has been discussed in detail in Paragraphs 5.50 through 5.56 and in Submission 1 date July 1, 1963. These tanks are built in a "cup-and-saucer" design. Operating procedures call for monitoring of the annular space between the tank and its saucer and of the water introduced under the tanks. If there is significant leakage from the tank into the saucer, the entire tank contents will be transferred into a spare tank kept for that purpose. Thus, under normal operating conditions there will be no loss of activity from these tanks.

Storage Lagoon

7.11 The very low-level wastes from this process--overheads from acid fractionation, solvent wastes, and miscellaneous wastes--can be put through the general purpose evaporator and the overheads from this can be put through ion exchange columns if necessary. It is expected that the normal activity content of the overheads from the general purpose evaporator will contain about 10^{-6} μc/cc of activity. This can be further reduced by a factor of 30 by the use of simple, non-regenerated cationic ion exchange resulting in a concentration of 3×10^{-8} μc/cc. The expected volume of these wastes is 40,000 gal/day. The average available flow in Buttermilk Creek is 41 cfs which is equal to 2.7×10^{7} gal/day. Thus, the available on-site dilution factor is 6.8×10^{2}. In Cattaraugus Creek an additional dilution factor of about 8.5 is available. The concentration in Cattaraugus Creek would be expected to be about 10^{-10} μc/cc. Furthermore, the residual activity in this stream will be largely Ru-106 and I-131 with some Zr-Nb-95. The MPC's for these isotopes are 1×10^{-5}, 2×10^{-6}, 6×10^{-5}, and 1×10^{-4} μc/cc rather than 1×10^{-7} for unknown activities when radium is absent. Therefore, the available factor of safety is about 10^{3} without any analyses of the effluent and about 10^{4} if we choose to carry out specific fission product analyses on this effluent stream. This stream will also carry about 130 curies per day of tritium since there is no known way to process it to remove the tritium. The concentration of tritium on-site in Buttermilk Creek will average 1.3×10^{-3} μc/cc. The on-site MPC is 10^{-1} μc/cc. In Cattaraugus Creek the tritium concentration is expected to average 1.5×10^{-4} μc/cc; the MPC here is 3×10^{-3} μc/cc.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

7.12 This low-level stream can be discharged directly to Buttermilk Creek and the level of activity at the site boundary will remain well within the MPC levels of 10 CFR Part 20. In addition there will be a series of lagoons available for use as an emergency holdup area. Their use will permit time for the decay of shorter-lived isotopes and will allow the adsorption on the soil of some of the longer-lived isotopes. The low level waste streams from the plant discharge into the interceptor, a concrete pit of 50,000 gallons capacity, which is designed for batching of wastes. A valved interceptor drain line will permit collection of one days output from the plant which will then be sampled for gross alpha, beta, gamma and tritium. The pH of the sample will be checked and the interceptor contents neutralized if necessary to pH to 6 to 8. A line is available for pumping the interceptor contents back to the plant for further processing. Normally, after sampling, the interceptor drain valve will be opened and the contents allowed to drain by gravity to the first holding pond, a 300,000-gallon settling basin with a high level overflow to the second pond. The second and third ponds each have capacity of about 2.3 million gallons. Between the second and third ponds will be a high level overflow and a valved drain line about 18 inches above the bottom of the pond. A valved drain line from the third pond will discharge to the creek. The capacity of the ponds above the overflows will allow complete holdup of 100 days output from the plant.

7.13 In view of the factors of safety available, no hazard will be presented by the routine handling of this aspect of the operation.

Burial Ground

7.14 Two types of wastes will be buried in the ground in conjunction with the operation of this plant. One is low-level solid trash of all sorts coming either from the plant operation itself or shipped in for burial from off-site users of radioactivity. The other is high-level solid trash in the form of leached hulls or equipment discarded from the plant. Activity associated with the former type is considered to be "available" in the sense that it could be leached out of the waste if it were contacted with water. The radioactivity associated with hulls and discarded equipment,

on the other hand, is not considered to be "available". In the case of the hulls, the radioactivity is induced in the hulls themselves which are either stainless steel or zirconium. Both of these metals are highly refractory and and would not be expected to corrode in the burial environment to any significant extent. They will have been carefully leached in boiling nitric acid prior to burial, inspected, and an aliquot analyzed to assure that significant quantities of fuel values are not being discarded with them. Equipment to be discarded will have been exhaustively decontaminated in place before bringing it out of the cells and it will then be further decontaminated in the Equipment Decontamination Room before it is buried. Hence, significant quantities of "available" activity is not expected to be associated with this type of waste either.

7.15 Burial of both types of solid waste will be done in the silty till described in paragraphs 2.17 through 2.25 and 2.41. We have now had considerable experience in working with this material in various excavations in the course of constructing the plant and in the operation of a low-level waste burial operation for wastes of the first type described in paragraph 7.14. From this experience it is possible to accept the very low permeability figures which were obtained during the subsurface investigations reported in Section II. Therein a calculated horizontal flow rate of 5×10^{-5} ft/day was reported. Since we expect to carry out no burial operations within 100 feet of any ravine, this calculates to something over 5000 years for any leached activity reach the ravine. Further this silty till has been shown to have good ion exchange capacity for the longer lived isotopes, Cs-137 and Sr-90. Thus, we expect the natural defenses of this material to contain completely the activity buried in it.

7.16 Silty till does not, however, act as a natural ion exchange material for ruthenium. This is a relatively short-lived isotope, however. For the sake of illustration assume that a curie of ruthenium were to escape from the burial site and begin to work its way toward on of the ravines. Further assume that discontinuities or chemical reaction of the waste with the soil should increase its velocity by a factor of 100. It would still take over 50 years for the activity to reach the stream. In this period of time the curie ruthenium would have decayed to 10^{-15} curie. The yearly flow in Cattaraugus Creek averages 3.5×10^{13} cc. Thus, for each curie/year which was leached from the burial ground, the concentration in Cattaraugus Creek would be 3×10^{-23} μc/cc. The MPC is 10^{-5} μc/cc.

7.17 We expect the release of activity to the environment from the operation of the waste burial ground--either from low-level trash containing "available" activity or from the high-level waste described above--to be completely inconsequential.

Egress of Personnel or Material

7.18 The control of release of activity into the environment by carrying it out on the persons or clothing of personnel or on material leaving the plant must be accomplished by administrative means. Personnel working with radioactivity in the plant will be provided with protective clothing which must be changed before they leave the plant. They will also be required to take a shower. Hand and foot counters will be provided for monitoring all persons--visitors included--who leave the working areas.

7.19 Similarly procedures will be set up whereby nothing may be sent off the plant without first having been surveyed and smeared by Health-Safety personnel. Guards will be instructed not to pass out any material which does not have Health-Safety certification.

7.20 While it is possible that occasionally barely detectable quantities of activity might slip through these procedures, it is essentially impossible for significant quantities of activity to get outside the plant in this manner. No difficulty in contamination of the environment is expected from this operation.

Product Shipment

7.21 Radioactive shipments are covered by AEC regulations in 10 CFR Part 71 and 72 primarily. All regulations in effect at the time of the shipment pertaining to such shipments are expected to be complied with by the shipper and the carrier. The only way in which radioactivity could enter the environment by way of product shipments is for the shipment to become involved in a serious accident. The regulations on product shipping containers are designed with that possibility in mind. The hazard thus involved is not one peculiar to this plant, its design, or its operation. There is a considerable body of experience on this aspect of the business and we expect in no way to increase the degree of risk above that which has already been accepted.

Conclusion

7.22 On the basis of the data and calculations presented in Paragraphs 7.6 through 7.21, in the normal operation of the chemical processing plant described herein, there will be no discharge of radioactivity to the environment in excess of the limits set forth in 10 CFR Part 20.

Abnormal Operations

7.23 In Paragraph 7.4 five abnormal events were hypothesized. These events range from the unlikely to the incredible but they delineate, we believe, the upper limit of any catastrophe which could occur in this plant and its related facilities. None of these accidents would result in levels of exposure to the general public exceeding the quide limits for gaseous emission suggested in Section 100.11 of 10 CFR Part 100; and further there is reasonable assurance that liqiud discharges at the site boundary could be kept within the concentrations for drinking water purpose specified in 10 CFR Part 20.

Loss from High-Level Waste Tanks

7.24 Careful measures have been taken to ensure the reliability of the high-level waste tanks, to provide multiple means of detecting any leakage in the unlikely event that any defects should develop and to minimize the effects on the environment of such leakage.

7.25 There are several methods of detecting leakage from the waste tanks barriers between the stored waste and the environment. The tanks have been equipped with liquid level measurement systems which are accurate to 1/4 inch or about 700 gallons. The tanks are located within saucers and each saucer is equipped with a liquid monitoring system. Each tank and saucer is contained within a reinforced concrete vault; the vault in turn is constructed upon four feet of graded gravel into which water is introduced for the primary purpose of maintaining the moisture content--and thus the bearing properties--of the underlying silty till. There are eight wells located within a foot of the vault which go down into the gravel area and through which the level of the water is measured and from which samples may be drawn to determine if there has been any leakage through the first three barriers. If there should have been any large penetration of the first three barriers, it would be possible to retrieve the activity with relatively little dilution by pumping out of the gravel area through any of the eight wells. This area thus represents the forth barrier to the escape of activity.

7.26 The local environment provides two additional barriers to the escape if radioactivity from the site. The tanks are located in the approximate center of a peninsula with a thick layer of silty till. It has been shown that the permeability of this silty till is so low that essentially complete containment would be expected of any waste that did escape the first four barriers. The till, then, is a fifth and most important barrier. The peninsula is bounded by Erdman Brook and Quarry Creek. USGS geologists who did the survey work on the site assure us that any radioactivity which escaped either onto or into the ground on this peninsula would eventually have to show up in one or the other of these creeks if it were not adsorbed on the soil by ion exchange. At the confluence of these two creeks there is established a sampling station to determine again that activity has not escaped from the site. The average yearly flow at this point is about 2 cfs. While it would be expensive, it would not be impossible to collect the total flow at this point and pump it back up to the plant site for additional processing if this should prove to be necessary. This represents the sixth barrier. There is still a final sampling of the discharge in Cattaraugus Creek at the point where the effluent leaves the plant property. This will provide the legal record of the plant discharges.

7.27 A spare tank identical to the working tank is provided so that in case the working tank begins to leak the contents may be transferred to the spare. Initially there will be a 1:1 sparing ratio. It is contemplated that during the first 15 years of operation of the plant two additional working tanks will be built and that the spare will serve all three. The eventual sparing ratio will be dictated by plant experience.

7.28 We believe that a waste tank could be ruptured only by sabotage or by a major earthquake. The former is outside the scope of the requirements of this review. The latter has been shown to be highly unlikely (see Paragraphs 2.46 through 2.48). In the event that a tank should rupture, however, the combination of the vault, the gravel area and wells, and the impermeability of the surrounding silty till can be expected to maintain the tank contents within the immediate area for a long period of time. There would be more than ample time to arrange a temporary piping system to permit pumping the waste solution from the tank, the saucer, or wells into the gravel, into the spare tank.

7.29 The multiplicity of methods for determining any leakage from the tank make it essentially impossible that such leakage could remain undetected. There are so many barriers between the waste and the environment that significant escape into the uncontrolled environment is also considered impossible. We even consider it possible to suffer a complete tank rupture--a most serious hypothetical and unlikely accident--and still maintain Cattaraugus Creek below the MPC levels of 10 CFR Part 20.

Criticality Incident Anywhere in the Plant

7.30 There have been eleven criticality incidents in solution systems.* Eight of these have resulted in a total number of fissions ranging from 4×10^{16} to 1.3×10^{18}. One, that at Idaho Chemical Processing Plant in October, 1959, resulted in 4×10^{19} fissions. Except in one case in which there was some warping of a tank bottom, none of these resulted in any physical damage. The assumption is made here that a criticality incident producing 10^{19} fissions in a single burst or 10^{20} fissions in a repeating incident is experienced anywhere in the plant and that the entire production of noble gaseous fission products plus 1/3 of the iodines (from 10^{20} fissions) are lost. The value of 10^{19} fissions is chosen to conform to calculations made at Savannah River suggesting this value as the upper limit of a single burst. These same calculations suggest 10^{20} fissions as the resultant of a maximum repeated burst. It will be shown that the limiting problem with this incident is not a public protection problem but rather the exposure of in-plant personnel to penetrating radiation at the time of the burst. For a repeating incident there would be time to evacuate personnel after the first burst and the exposure to penetrating radiation can be considered equivalent to that from a 10^{19} fission burst. This is considered in Paragraphs 8.26 and 8.27. Insofar as the general public is concerned there is no hazard from the immediate radiation at the time of the burst. It is well established that the limiting condition in an occurrence of this type is the thyroid dose from the iodine isotopes released. Therefore, this event is analyzed on the basis of thyroid dose to a person on the periphery of the site, at Springville, and at Buffalo. All three are calculated for the average and inversion conditions specified in Table 7.5. In the case of Springville and Buffalo the total population dose is calculated and expressed in man-rem.

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* Nuclear Safety, Quarterly Literature Review, Vol. 3, No. 2, Dec. 1961, Pages 34-37 plus a subsequent Hanford incident and one in Charlestown, Rholde Island in July, 1964.

7.31 Table 7.31 lists the peak activity of each of the iodine isotopes 131 through 135 and the time after the accident when the peak occurs. These have been calculated using NRDL-456, "Calculated Activities and Abundances of U-235 Fission Products". With one exception, the peak activities have been assumed in calculating the population dose. This procedure is conservative but by a relatively small amount over the time periods involved. The one exception is the activity of iodine-134 at the time it reaches Buffalo under inversion conditions. The transit time in this case is so large in relation to the half-life of iodine-134 and its precursors that its activity level was found to be negligible compared to the remaining iodine isotopes.

7.32 The off-site doses have been computed assuming that the iodine is released from the stack instantaneously. The total inhaled activity has been calculated using Equation 7.7a for short-term center-line concentrations. The calculations have been performed for average (slightly unstable) meteorological conditons and for inversion (moderately stable) conditions. The distances involved are:

Site periphery	1,500 meters
Springville	7,200 meters
Buffalo	51,000 meters

The use of Equation 7.7a is valid for the first two distances. Extrapolation to 50,000 meters is questionable, but gives a fair estimate. The results so obtained are given in Table 7.32a. Then using the approximations suggested in 10 CFR Part 100 for the thyroid dose from each of these isotopes, the total rem per person and the fraction of the 300 rem reference value are calculated for the three locations and for both types of meteorology. Total man-rem values have also been calculated. These date are presented in Table 7.32b. Calculations supporting the numbers shown in these three tables are given in Appendix 7.32. It can be seen that in no case is the reference value used for evaluaton of reactor sites exceeded or even closely approached. The highest value indicated, a 1.95-rem/person dose in Springville under inversion conditions, is not expected to be encountered since it is the opinion of meteorologists (see Paragraph 2.13) that an inversion aimed at Springville would be caught and held in the Buttermilk-Cattaraugus Valley systems. Even this value is only about 0.7 per cent of an emergency dose of 300 rem.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Table 7.31

Quantities of Iodine Isotopes Formed from 10^{20} Fissions[a]

Isotope	Time of Peak Activity	Peak Activity, Curies
I-131	5.2 hours	25
I-132	7.6 hours	80
I-133	3.5 hours	420
I-134	46.0 hours	5770
I-135	2.2 hours	1470
	Total	7765

a Assuming 1/3 of the iodines are lost from the stack.

Table 7.32a

Total Dose Due to Radioiodines, Rem/Person[a]

Location	Inversion (Moderately Stable)	Average (Slightly Unstable)
Site Boundary	0.09	0.63
Springville	1.95	0.06
Buffalo	0.33	2.8×10^{-3}

a From instantaneous release of 1/3 the iodines from 10^{20} fissions.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Table 7.32b

Individual and Population Doses at Several Points

In Event of a Criticality Incident

	Site Periphery		Springville		Buffalo	
	Inversion[a]	Average[a]	Inversion[a]	Average[a]	Inversion[a]	Average[a]
Individual Dose, Rem/ Person[c]	0.09	0.63	1.95[b]	0.06	0.33[b]	2.8×10^{-3}
Fraction of 300 Rem/ Person Dose	3.1×10^{-4}	2.6×10^{-3}	6.3×10^{-3}[b]	2.1×10^{-4}	1.1×10^{-3}[b]	9.3×10^{-6}
Exposed Population Dose, Rem Total	---	---	24[b]	0.8	583[b]	5

a See Table 7.5 for conditions assumed.

b It is not expected that, under inversion conditions, any activity will reach Springville or Buffalo; but rather will be trapped in Cattaraugus Valley.

c See Table 7.32a.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Criticality Incident in Fuel Pool

7.33 The fuel pool is designed to hold 1000 fuel elements in racks of such geometry that the establishment of a critical array is impossible even if the elements were all of the maximum reactivity of any fuel before it is irradiated. Allowance is made for moving through the storage array an element of the highest reactivity. This is discussed in Section VI and the occurrence of a criticality incident here is shown to be extremely unlikely. Despite the fact that a criticality incident in the fuel pool is extremely unlikely, the following event is hypothesized:

> It is assumed that an element is jammed into the interstice between four elements and that the five elements are involved in a critical event, that a 10-mwt boiling water reactor will be set up, and that it will operate 3 hours before it is possible to shut it down. It is further assumed that all five elements are defective and, thus, that some gaseous activity can escape from the element.

7.34 Calculations supporting this section are shown in Appendix 7.34. The heat released would raise the temperature of the water in the storage pool only about 16F even if the pool water coolers failed to operate. Therefore, there is no danger that the water level in the pool would drop significantly and consequently the shielding provided by the water would prevent any hazard from increased radiation levels from direct radiation. EBWR defect test studies have shown that the fraction of noble gaseous activity lost per second from a defective fuel element is about 4×10^{-8}. This same test showed that the iodine loss was at least an order of magnitude less than this. The total inventory of gaseous activity in the five fuel elements assumed to be involved in this incident and the amounts which may reasonably be lost from the fuel pool water are shown in Table 7.34. These quantities of iodine isotopes are much less than the amounts which have already been shown to be readily tolerated by this environment (see Paragraphs 7.30 to 7.32). Consequently the iodine releases result in less hazard than has already been shown to be acceptable. The releases of kryptons are also much less than those which have already been shown to be within MPC. Similarly, the xenon-133 discharge results in concentrations under the worst condtions of only 0.01 MPC. The only aspect of this hypothetical incident which has not already been calculated in the section is the xenon-138 release.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Table 7.34

Gaseous Activities Lost from Fuel Pool

During Assumed Criticality Incident

Isotope	Inventroy Curies	Fraction Lost In 3 Hours	Total Lost in 3 Hours, Curies
Kr^{85m}	6.1×10^4	4×10^{-4}	24
Kr^{85}	2.5×10^3	4×10^{-4}	1
Kr^{88}	2.4×10^5	4×10^{-4}	10
I^{131}	2.7×10^3	4×10^{-5}	0.1
I^{132}	3.2×10^5	4×10^{-5}	13
I^{133}	5.4×10^4	4×10^{-5}	2.2
I^{134}	1.6×10^6	4×10^{-5}	64
I^{135}	1.6×10^5	4×10^{-5}	6.4
Xe^{133m}	5.2×10^2	4×10^{-4}	0.21
Xe^{133}	9×10^3	4×10^{-4}	3.6
Xe^{135}	1.5×10^6	4×10^{-4}	600
Xe^{138}	3.7×10^6	4×10^{-4}	1500

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Using the method employed in Paragraph 7.7 (Equation 7.7a) the concentration of xenon-138 at the site boundary under inversion conditions is 1.4×10^{-6} μc/cc. No MPC for this isotope is given in either 10 CFR Part 20 or in NBS Handbook 69 but it would not appear that this would result in any hazard. This event can, therefore, be tolerated without exceeding published MPC's.

Chemical Explosion

7.35 The assumption is next made that a vessel containing one full day's charge of fuel in solution suffers an explosion which ruptures the vessel distributing the contents throughout a cell and putting some fraction of the contained solution into the ventilating system. The ventilating system will withstand the rupturing of a tank. However, there might be some plugs or windows loosened. So long as ventilation is maintained air flow should remain into the cell except for the instant of the explosion. In analyzing this event some assumptions contained in "Radiochemical Facility Hazard Evaluation", by E. D. Arnold, A. T. Gresky, and J. P. Nichols, ORNL-CF-61-7-39, July 10, 1961 are used. This is a very similar analysis of a completely analogous situation to that considered here. The assumptions are made therein that aerosols penetrating high-efficiency filters will contain 0.14 mg/M^3 of material with the same concentration as the original dispersed solution and that the MPC for mixed fission products is 6.6×10^{-9} μc/cc. The ventilating air passing through the filters of this plant amounts to 32,000 cfm or 900 M^3/min. Then 0.14 x 900 or 125 mg/min of the original solution may be assumed to pass through the filter. We further assume that the gaseous activity has already been released and that in twenty minutes the ventilating system will have picked up nearly all of the gross activity that it is going to. Under these conditions about 2.5 grams of solution will be released. The maximum activity to be expected in the plant is about 700 curies per liter or 0.45 curie per gram for a total discharge of 1.1 curies. Following the methods of Paragraph 7.7 the poorest value of X/Q (at a distance of about 5,000 meters) is 1×10^{-5}. Q is equal to 1.1/3600 = 3×10^{-4} curie/sec. The X = 3×10^{-9} μc/cc. This is less than the MPC for mixed fission products assumed by ORNL in the above report and it would appear possible to accept this particularly untoward accident. There would be, of course, a big cleanup job in the cells. This would be undertaken according to methods outlined in Paragraphs 8.8 and 6.54 through 6.56.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

Failure of Iodine Removal Equipment

7.36 Finally it is assumed that the silver reactors and other iodine removal equipment all fail and that this is not discovered for a period of one day. That this could remain undetected for 24 hours is extremely unlikely since the stack monitor would detect the iodine increase at once. If the entire charge of iodine-131, 1.7 curies, were to be lost during a day the value of Q is 2×10^{-5} curie/sec. Under worst conditions (at a distance of about 5,000 meters) the poorest value of X/Q is 1×10^{-5} and the concentration of iodine-131 at this point would then be 2×10^{-10} μc/cc. This is less than the MPC for continuous exposure off-site.

Conclusion

7.37 All of the abnormal incidents hypothesized in Paragraph 7.4 have been analyzed. It has been shown that in all cases except the 10^{20} fission criticality incident the limits prescribed in 10 CFR Part 20 for continuous exposure are met and that in this one case there is no dose at any point which exceeds, or even closely approaches, the guides suggested in 10 CFR Part 100 for emergency conditions. Since we expect the probability of these events to be very low, to the point of incredibility, and since they can be handled by the environment even if they should occur, we submit that the operation of this plant does not constitute an undue hazard to the general public beyond the site boundary.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

VII – PROTECTION OF PLANT PERSONNEL

Before the

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For Licenses for a Spent Fuel Processing Plant

Under Sections 53, 63, 81, 104 (b), and 185 of the

Atomic Energy Act

AEC Docket No. 50-201

Submission No. 16 - Final Safety Analysis Report

Revision of Section VIII - Protection of Plant Personnel

Revision 2, August 20, 1964

VIII PROTECTION OF PLANT PERSONNEL

Design Criteria

8.1 The design criteria and the operating rules of the NFS plant have been set up so that the plant will conform to the rules and regulations specified in 10 CFR Part 20, Standards for Protection Against Radiation.

8.2 The plant will have an across-the-board industrial safety program (see Section IX) aimed at reducing accidents of all types. It will maintain a constant program designed to increase the safety morale of all of its personnel, both in the area of normal industrial safety and in that of radiation safety.

8.3 The radiation safety program is designed to protect the plant personnel from:

a. external radiation,
b. inhalation,
c. ingestion.

All three have been taken into consideration in the design of the plant. They also dictate the conditions under which the plant will be operated. In subsequent paragraphs each of these areas is discussed in detail to demonstrate that the plant as designed can be operated in accordance with the provisions of 10 CFR Part 20. In addition, the accidents which were hypothesized in Section VII are reanalyzed from the standpoint of personnel in the plant; and some less serious but more probable events are discussed from the viewpoint of personnel protection.

Protection from External Radiation

8.4 The primary protection for the worker from penetrating radiation is to interpose sufficient shielding between him and the radioactivity at all times. The plant shielding has been described in some detail in Paragraphs 6.59 through 6.65. The shielding has been designed so that, when the most active unit which could be in any particular section of the plant is present, the radiation level on contact outside the shielding in a normal access area would be 1 mr/hr. In many cases the point of contact will not be readily accessible to personnel and the percentage of the time that the shielding wall is subjected to the maximum activity level is small. The shielding design has been based on a "design"

fuel having the following irradiation on history:

Burnup	30,000 MWD/T
Specific Power	35 MW/T
Cooling Time	150 Days

8.5 Fuel is brought into the plant in shielded casks which have had their design carefully checked to ensure that adequate protection is available. A shipment will be surveyed before it is sent out. It will be surveyed again upon arrival at the plant. Before the carrier is opened, it is placed under sufficient water (see Paragraph 3.7) so that, as a fuel element is removed, there will be at least 11 feet of water over the top of the longest type of fuel element. Movement of the elements in the storage pool and their storage are also conducted under at least this much water. Transfer to the PMC is done remotely under water and back of concrete shielding. The mechanical operations in the PMC and GPC are carried out remotely back of concrete shielding. The transfer to the CPC is handled remotely. All operations in the CPC are remote. Transfers to the remaining contact-maintained cells are fluid transfers carried out remotely. All operations in the entire process, therefore, are carried out behind shielding until product is decontaminated to the point where external radiation is no longer a problem. Plutonium products containing high concentrations of Pu-240 will be placed semi-remotely into containers with sufficient shielding so that they may be handled safety.

8.6 Sampling is an operation which can contribute significantly to exposure of personnel. The sampling systems, which were described in detail in Paragraphs 6.22 through 6.36, have been designed to permit most of the sampling to be carried out completely behind shielding and to provide working background of 1 mr/hr or less. Many dilutions will be made inside the shielding and only the diluted analytical sample will be brought out. This will reduce considerably the potential for spillage and also the resultant exposure in the event of spillage.

8.7 In order to maintain the background levels in the plant at design levels, it is necessary not only to have adequate shielding but also to maintain strict controls to prevent spillage. This is done first by keeping the activity back of the shielding--there are no planned withdrawals of activity except for the samples, many of which have been already diluted; second by a careful and continual radiation survey program to detect areas in which there may have been an inadvertent introduction of activity; and third by a prompt and immediate cleanup of such areas at the same time determining the cause of the event and correcting it.

8.8 Maintenance work, both routine and major, can be expected to contribute somewhat to the whole body radiation of the plant personnel. It is the intention of NFS to permit maintenance work only under such conditions that no worker will be exposed in excess of the limits defined in 10 CFR Part 20. The maintenance procedures, which are described in detail in Section IX have been set up to minimize the exposure of the personnel. However, it is clear that maintenance work will have to be done in high radiation areas (areas in which the background levels exceed 100 mr/hr). Such work will be controlled by a work permit system as described in the Health-Safety portion of Section IX and be authorized by the plant manager. In attacking any maintenance job, the area is carefully surveyed and the amount of time that may be permitted a worker in the are is calculated. Work in the radiation field is done under closely supervised conditions. Accurate time is kept from outside the field. Recording meters as well as film badges are worn during the operation and a log of the exposure is kept and this is added to each worker's permanent radiation record. The level to which an area will be decontaminated before maintenance is attempted will vary with the amount of time needed to carry out the job, but in no case will a worker be allowed to enter a radiation field exceeding 2 r/hr without special approval of the plant manager. It will be normal plant practice to limit the exposure of any individual for any single maintenance job to 0.2 rem. Subject to the maximum limitation specified above, the balance between time and activity level will be a decision to be made by plant supervision in each instance.

8.9 In the normal operation of the plant we expect that an operator will spend no more than two hours per day in the full 1 mr/hr permitted in a normal access area. It is expected that most of the normal access area will have a background much less than this. For planning purposes we have assumed that the additional six hours per day will be in an average background of 1/6 mr/hr. The total background radiation for the quarter would then amount to 0.2 rem. This would leave about 1 rem per quarter for maintenance operations without exceeding 1.25 rem/quarter. With exposure limited to 0.2 rem/maintenance jobs, a given individual could perform five such maintenance jobs per quarter. There will be about sixty men in the plant who can be called upon to carry out such jobs so that the plant can carry out a maximum of 300 such operations per quarter, about five per day.

Inhalation

8.10 The primary protection of the workers from inhalation lies in keeping the activity inside the process equipment itself. As a second line of defense, all of the equipment is contained in cells maintained by a separate ventilation system at a pressure negative to the working areas. As third line of defense, masks and supply-air equipment are available. These ventilation systems have been described in detail in Paragraphs 6.3 through 6.21. Under all normal operating conditions no process activity is expected to escape past the first two barriers and into the operating areas.

8.11 There will be a system of fixed air samplers backed up by a program of air monitoring with portable air monitors to assure that the air in the working area does, indeed, remain free of activity. This monitoring program has already been described (see Paragraphs 6.66 through 6.76). The monitors will have audible and visual alarms set to operate at the lowest practical level so that remedial action may be taken before any consideration of evacuation is necessary.

8.12 Consideration has also been given to the mechanism whereby activity could be brought into the plant by recycle into the building air intake of air discharged from the plant stack which is located on top of the building. In Appendix 8.12 there are shown calculations for average and inversion conditons which indicate that the amount of recycle to be expected is completely negligible in either case. There is, however, an infrequent condition whereby the discharge from a stack may come directly down upon the stack. Under these conditions the amount of dilution could be small. A calculation is shown in Appendix 8.12 for the normal iodine-131 discharge. This shows the concentration of Iodine-131 at the stack exit with no dilution at all except that afforded by the ventilation air in the stack itself. The concentration of iodine discharged from the stack would be 6.7×10^{-8} $\mu c/cc$ which is only a factor of 7.5 higher than the occupational MPC of 10 CFR Part 20. This particular meteorological condition is not expected to occur very frequently or to persist for any long period of time. Even with no dilution, and it would be expected that there would be some—perhaps a factor of ten, the concentration is such that under the provisions of Paragraph 20.103b the iodine-131 present in the building air could be tolerated for five hours. Such a condition would be picked up very quickly by one or more of the monitors. This iodine concentration would be attained only during the course of a dissolution; there would be ample time to shut down the dissolver or evacuate the building or both.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

8.13 There will be an ample supply of protective equipment such as Scott Air Packs available for use during emergency conditions or during maintenance work inside cells. The Health-Safety program (see Section IX) will include frequent training sessions and drills for all personnel in the use of this equipment so that in an emergency the equipment should be used promptly and properly.

Ingestion

8.14 The control of the problem of ingestion of radioactivity is largely one of developing within the workers good safety morale and habits of personal hygiene and of providing them with adequate protective clothing, devices, and monitors to facilitate the execution of the program. It is a problem which has been dealt with routinely at all of the presently operating chemical processing plants of the AEC without creating any serious hazards.

8.15 Protective clothing will be issued to all personnel working in the plant areas and must be worn therein. This clothing will not be worn outside the plant areas. It will be laundered at the plant and returned to service or discarded to solid waste depending on monitoring preceding and following the laundering process. An ample supply of other specialized protective clothing such as surgical, heavy rubber, and cotton gloves, caps, boots, and tape (for taping gloves to coveralls, for instance) also will be maintained.

8.16 Eating and smoking will be controlled throughout the plant. Eating will be done only in the designated lunch room. Protective clothing will not be worn into the lunch room. There will be hand and foot counters at the door. The area will be checked frequently by the Health-Safety survey. Smoking will be done only in designated areas.

8.17 At the conclusion of each shift each individual who works in the plant areas will be required to change clothes, and check his hands and shoes before leaving the premises. Two hand and foot counters are provided.

8.18 The above program of hygiene has proved to be satisfactory to maintain the ingestion of activity to negligible levels at other installations. The common practice to back up this program with a medical program will be followed at the NFS plant. The medical plans for the plant are as follows:

> The medical program will consist of a very thorough pre-employment medical history and physical examination for each prospective employee. The medical history will be aimed at not only past illnesses and injuries but particular attention will be paid to history of past radiation exposure, allergies, blood dyscrasias, tumors, and any evidence of emotional instability. The laboratory studies on all applicants will consist of a minimum of complete blood count, serology, urinalysis, chest X-ray, and vital capacity determinations. Each employee will have a complete physical examination yearly. A complete blood count will be done twice yearly; clinical urinalysis monthly. Bio-assays will be done on an "across-the-plant statistical survey" plan and follow-up examination, as this survey may indicate. The pre-employment physical examination and laboratory studies will be repeated on each individual leaving the employ of the company.
>
> A dispensary will be maintained for care of ordinary minor on-the-job injuries. There will be facilities for intensive first-aid care of severe injuries such as burns, fractures, and gross contamination with radioactive material. Immunization against tetanus will be routine for all employees.
>
> Close liaison with the Health and Safety Department will be maintained. The medical director will assist in health and safety training and indoctrination. He will review with the superintendent of health and safety all industrial radiation exposure records; air, water, and plant radiation survey records. He will cooperate with the superintendent of health and safety in plant inspections.

Detailed records of all the above will be maintained by the medical director.

8.19 As explained in the foregoing paragraphs we anticipate no difficulty in conducting the normal operation of this plant within the framework of permissible levels of exposure. It remains in the remainder of this section to analyze the consequence to employees of accidents.

Analysis of Accidents

8.20 In Paragraph 7.4 five highly abnormal hypothetical incidents were proposed and the effect of these upon the public was considered in Paragraphs 7.24 through 7.37. These same incidents are now considered with reference to the plant personnel.

Tank Rupture

8.21 It has been shown that the soil in which the tanks will be constructed in quite impervious and that the liquid from a ruptured tank would be held in the immediate vicinity for some period of time. It is proposed that the waste be transferred into one of the spare tanks as quickly as possible. The type of action envisioned would involve pumping the solution into a spare tank probably through temporary lines which might be laid overground with only a minimum of shielding. The laying of this pipe would not require personnel exposure except for the connection into the ruptured tank. This would be done by lowering a flexible hose into the tank through one of the spare nozzles on the tank. If necessary, such an operation could be accomplished from back of a temporary shield constructed outside the radiation field and pushed into place with a payloader or crane. If the earth shield remained intact, no additional shielding would be required. If the condenser system was inoperative, it might well be necessary to carry out this operation with the protection of supply-air masks. The transfer system would probably be set up with two pumps in the system in an effort to avoid any maintenance on the pumps during the transfer operation. The pumps would be operable from outside the radiation field.

8.22 Maintenance of the pumps, if required during this operation, would certainly entail operations in a high radiation. In the transfer set-up a tee would be inserted upstream from the pumps so that the lines could be flushed and decontaminated somewhat before such

maintenance would be attempted. It would have to be done from behind a portable shield. In the case of so serious a problem as this there would be no question that operations of the plant would be shut down and all available exposure time would be used in solving this problem. Supervisory personnel to the highest levels and individuals from other plants operated by the company—those who receive no radiation in the course of their work—could be brought in if necessary. No individual, however, need be exposed beyond permissible levels.

8.23 Most of the exposure associated with this incident would be in cleaning up afterward. Dismantling the highly contaminated lines, pumps, and valves and disposing of them would certainly require operations in high radiation fields. However, the need for speed would no longer be present and enough time and people would be used to assure that the task was accomplished within the permissible radiation exposures.

Criticality Incident in Plant

8.24 This incident, discussed in Paragraphs 7.30 through 7.32, assumed that the ventilation system remained in working order since that situation results in the most immediate and complete discharge of the gaseous isotopes to the environment. In that event the air inside the plant would be completely safe, as evidenced by the calculations shown in Appendix 8.12, in all cases except that of the unlikely recycle. In Appendix 8.25 it is shown that if such a downdraft occurs under average conditions the amount of dilution will amount to a factor of about 500. If it occurs under inversion conditions the dilution factor will be about 25. For the purpose of this calculation the dilution is taken as 10. It is further assumed that during the course of the entire discharge (assumed to be 10 minutes) the downdraft will be centered precisely on the air intake ten percent of the time. It is also assumed that some personnel are exposed to the resulting concentration for the entire ten minutes and that they are so preoccupied with rendering assistance to other personnel or are otherwise so upset that they did not make use of the available supply-air equipment. Under these circumstances they would be exposed to the concentrations and receive the thyroid doses shown in Table 8.24. These calculations are also shown in Appendix 8.25. The total thyroid dose is less than the dose suggested as an emergency guide in 10 CFR Part 100.

Table 8.24

Thyroid Dose During Recycle Coincident With a Criticality Incident

Isotope	Conc., μc/cc	$\frac{X}{Q}$ x Frequency	Dose Rate, Rem/ μc/(cc)(sec)	Time, Seconds	Dose, Rem
I-131	2.6×10^{-3}	$\frac{1}{100}$	110	600	5
I-132	6.3×10^{-3}	$\frac{1}{100}$	4	600	0.7
I-133	0.04	$\frac{1}{100}$	31	600	24
I-134	0.6	$\frac{1}{100}$	2	600	22
I-135	0.16	$\frac{1}{100}$	6	600	23
				Total	74.7

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

8.25 There is no reason to assume that the events which could lead to a criticality incident would also lead to shutting down the ventilation system. There may be, however, a small probability that the two events might occur simultaneously. The possibility is a difficult one to analyze since the amount of leakage of activity from the cell would be expected to vary considerably depending on the conditions. If the supply air remains on, the exterior of the cells would remain at a higher pressure than the cells and little, if any, leakage should occur. If both the supply air and the exhaust go out, there may still be a little negative pressure in the cells due to the natural draft of the 65-meter stack. If the static pressure in the cell becomes equal to that outside the cells, some leakage would occur but it should not be large since any leakage path would be tortuous. There would not seem to be a mechanism whereby the cells would reach a higher pressure than the surroundings. Failure of the ventilation system will activate an alarm which will require evacuation of the plant unless countermanded by plant supervision. We conclude, therefore, that if a criticality incident were to occur coincident with a failure of the ventilating system, the plant would be evacuated long before anyone could receive a significant dose from inhalation.

8.26 The most important aspect of protection of plant personnel in connection with a criticality incident is to assure that no one receives a serious dose of penetrating radiation at the time of the incident. The first line of defense is, of course, to prevent the occurrence of the incident. Great care is being exercised in the design of this plant and in the setting up of its operating procedures to ensure that a criticality incident does not take place. The whole subject of criticality control throughout the plant has been presented in detail in the final paragraphs of Section VI. We believe that we have reduced the probability of such an incident to an absolute minimum. However, there have been eleven such accidents in solution systems. Every major site save one has had one. There have been five incidents in metal-air systems at Los Alamos.

8.27 An Oak Ridge study* has calculated the prompt neutron and gamma dose at the outside of a normal concrete shield from a nuclear reactor of 10^{18} fissions and these data are shown in Table 8.27. They can be used for a 10^{19} fission event by direct ratio. The concrete shielding walls

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* ORNL-CF-61-7-39, "Radiochemical Facility Hazard Evaluation", E. D. Arnold, A. T. Gresky, and J. P. Nichols, July 10, 1961, Page 6.

Table 8.27

The Prompt Neutron and Gamma Dose at the Outside Of a Normal Concrete Shield From a Nuclear Reaction of 10^{18} Fissions[a], [b]

Ordinary[c] Concrete Shield Thickness, Ft	Dose at Outside of Shield, rem	
	Metal Nuclear Reaction	Nuclear Reaction in Aqueous Solution
1	88,000	5,200
3	317	23
4	17.0	1.9
5	0.960	0.14
6	0.059	0.012

a The dose rate may be calculated for any other number of fissions through the use of a direct proporation.

b ORNL-CF-61-7-39, "Radiochemical Facility Hazard Evaluation", E.D. Arnold, A.T. Gresky, and J.P. Nichols, July 10, 1961, Page 6.

c For high density concrete the gamma dose is reduced by a factor of 1.6 for a given concrete thickness.

for the GPC, PMC and CPC have openings for viewing windows which are equivilant in shielding value to the concrete walls for gamma radiation but offer less protection than the concrete for neutron radiation. Table 8.27 does not reflect the increased neutron dose to an employee who might be in front of one of the viewing windows. In the case of a 10^{19} burst from a criticality accident in the dissolver, the total prompt neutron plus gamma dose to an employee at the nearest viewing window would be about 300 rem, if the window were completely transparent to neutrons. The only place in the plant where a metal-air incident is at all possible is in the PMC - GPC. There we have four feet of high density concrete shielding and the resulting dose would be negligible. A solution system event could conceivably occur from the dissolver on. In the CPC there are six feet of concrete and the dose would be even less than in the PMC. In Cell #1 there are five feet of concrete shielding. The dose would still be negligible. In the remaining four cells there are three feet. At the lower end of the process there is no need for this much shielding from the fission product content. The minimum of three feet of concrete shielding has been carried down to the end of the process in order to assure that even if a 10^{19} fission critical incident should occur, and a worker should be standing right opposite the point in the cell at which the event occurred, he would still not receive a MLD of penetrating radiation.

8.28 When the product must be removed from the plant and put into storage and eventually onto a truck for shipment, contact with the product is required. Therefore, particular care has been exercised with product shipment plans. This is discussed in Paragraph 7.21.

Criticality Incident in the Fuel Pool

8.29 The hazard of a criticality incident in the fuel pool to the general public has been discussed in Paragraphs 7.33 and 7.34. It was shown therein that the amount of heat released is not enough to destroy the integrity of the water shielding which is enough to keep such an incident from irradiating anyone significantly from the prompt neutrons and gammas. It is necessary to consider the gaseous activity which is given off, however. The quantities of gaseous isotopes expected to be released during three hours was shown in Table 7.34. In Table 8.29 these quantities are shown as μc/cc and their concentrations in the fuel receiving and storage area air are shown assuming that it is diluted with the 11,000 cfm of ventilating air which is drawn through

Table 8.29

Gaseous Activities Lost into Fuel Receiving and Storage Area During Assumed Criticality Incident

Isotope	Activity Released μc/sec	Conc, μc/cc	MPC
Kr-85m	2300	4.4×10^{-4}	6×10^{-6}
Kr-85	93	1.8×10^{-5}	1×10^{-5}
Kr-88	930	1.8×10^{-4}	-
I-131	9.3	1.8×10^{-6}	9×10^{-9}
I-132	1200	2.3×10^{-4}	2×10^{-7}
I-133	200	3.8×10^{-5}	3×10^{-8}
I-134	6000	1.1×10^{-3}	5×10^{-7}
I-135	600	1.1×10^{-4}	1×10^{-7}
Xe-133m	20	3.9×10^{-6}	-
Xe-133	330	6.4×10^{-5}	1×10^{-5}
Xe-135m	5.5×10^{4}	1×10^{-2}	-
Xe-138	1.5×10^{5}	2.7×10^{-2}	-

that area. These concentrations range from twice the 40-hour MPC for Kr-85 to 2000 times the MPC for I-134. These MPC's are for continuous breathing and can be scaled up or down with time. Taking the I-134 as controlling, the room would have to be evacuated within 1/2000 of 40 hours or in just under one minute in order not to exceed one week's allowable inhalation. It will certainly be possible to evacuate this room in less than a minute and an event such as this would be immediately obvious to anyone in the room. Presumably there would be a visible flash. Monitors would trip and alarm, and the fuel pool itself would be visibly agitated. After evacuation, personnel could put on supply-air equipment in other parts of the building before re-entering to take remedial action.

Chemical Explosion

8.30 In Paragraph 7.35 a chemical explosion is assumed which ruptures a tank containing an entire day's charge of activity less the gaseous activity (since, in order to have the full day's charge in solution, it will have to have been through the dissolution step during which the gaseous activity is lost). The cell ventilation system has been designed to withstand the effects of such an explosion. So long as the ventilation system is maintained, no activity should get out of the cell in which the explosion took place except some that would be lost during the period of overpressure following the explosion. This period is estimated to be about one second. The calculations of Appendix 8.12 show that, for all conditions except a direct recycle of stack discharge into the air intake, there will be negligible concentration of activity in the building air. In the unlikely event that such a recycle and an explosion should coincide and using a calculative method analogous to that shown in Appendix 8.25, the concentration of unfiltered solid activity at thr throat of the stack would be:

$$\frac{0.125 \text{ g/min} \times 0.45 \text{ curie/gram} \times 10^6 \ \mu\text{c/cc}}{32{,}000 \text{ cfm} \times 28317 \text{ cc/cf}} = 6.2 \times 10^{-5} \ \mu\text{c/cc}$$

As in Appendix 8.25 a dilution factor of 1/10 from stack to intake and a frequency factor of 1/10 were used. The resulting concentration in the building would be 6.2×10^{-7} µc/cc. This is about 100 times the MPC assumed by Oak Ridge for mixed fission products as aerosols and it implies that about 25 minutes would be available for evacuating the plant. This is more than adequate.

Revision 1, Oct. 29, 1962
Revision 2, Aug. 20, 1964

8.31 Although we believe that the ventilation system will be maintained in operation if an event such as this should occur the possibility that it does not continue to function must be considered. As in the discussion of Paragraph 8.25 we find this situation difficult to analyze and for the same reasons. In this case it is certain that the cell would be pressurized for a short time, perhaps several seconds, and that during this time some activity would escape. If the building ventilation system were not functioning, the air in the building will be essentially stagnant. In the immediate vicinity of the cell quite high concentrations can be hypothesized depending on the assumptions chosen. However, it seems difficult to hypothesize a mechanism whereby this activity will spread very quickly from the immediate area if the building air is not moving. An explosion would alert anyone close to it and the immediate area would be evacuated in a matter of seconds. Reconnaissance and remedial action would then be carried out with supply-air masks.

Failure of Iodine Removal Equipment

8.32 As in all of the situations in which activity is put up the stack, here again there is no hazard inside the plant at all except in the unlikely case of direct recycle the concentration of iodine that might be found in the building air would be:

$$\frac{20\ \mu c/sec \times 0.1 \text{ dilution factor} \times 0.1 \text{ frequency}}{1.5 \times 10^{7}\ cc/sec} = 1.3 \times 10^{-8} \mu c/cc$$

The 40-hour MPC for iodine-131 is 9×10^{-9} $\mu c/cc$ which is lower than the above calculated concentration by only about 30 percent. Thus, this concentration could be permitted for over two days. It is unlikely that the recycle would be in just the right position for more than an hour during the day. In any event this concentration would trip all the building air monitors and the incident would be discovered more readily and the dissolver shut down.

Minor Accidents

8.33 We do not believe that the accidents which have been discussed in Sections VII and VIII will occur. It is a good deal more likely, however, that during the course of the operation of this plant there will be a number of much more minor occurrances which pose no hazard at all to the general public but which, if they were not handled properly, could lead to additional exposure of the plant personnel. Such

events might be illustrated by:

a. Spilling of activity, particularly analytical samples or special samples such as waste tank sample.

b. Tracking of spilled activity from one place to another.

c. Pulling activity into jet steam lines by improper venting.

d. Leaking waste lines in diversion boxes.

e. Spilling of product solution.

The problem with all of this type of event is the same. In one way or another they lead to an increase in the background radiation which the worker may receive. This is undesirable since in nuclear work one wishes to avoid any unnecessary radiation. It is also undesirable since it is important to keep the "operating background" as low as possible in order to leave a cushion with which to carry out the required maintenance work. There are several lines of defense against this sort of problem and they are the same for all of them:

a. First, the plant has been designed to eliminate, insofar as possible, the necessity for handling even small amounts of activity.

b. Second, the operating rules are designed to eliminate every possible exposure.

c. Third, the fixed monitoring system (see Paragraphs 6.66 through 6.76) is designed to detect increases in either background radiation or air concentrations.

d. The fixed monitoring system is backed up by a formal mobile monitoring and survey program (see Section IX).

e. Each employee will wear both meters and film badges. He will be trained to check his own exposure rate frequently and to use portable monitors himself so that he need not rely completely upon Health-Safety Coverage.

Thus, these minor accidents should not go undetected. None of these accidents could credibly result in exposure to plant personnel in excess of the limits set forth in 10 CFR Part 20.

Summary

8.34 In this section we have shown that we are able to operate the NFS plant under all normal conditions within the requirements of 10 CFR Part 20 as they pertain to the protection of the plant personnel. This includes protection from external radiation, inhalation, and ingestion. Both the operation of the process proper and all the necessary maintenance operations are included in this statement.

8.35 The same series of hypothesized major accidents that were discussed in Section VII have been considered from the standpoint of the protection of the workers in the plant. It is shown that each of these unlikely events could be sustained without undue risk of exposure to plant personnel.

8.36 Finally the problem of minor accidents that could lead to increases in the background radiation received by plant personnel is considered and the multiple series of defenses against these are shown.

8.37 We conclude that the NFS plant can be operated within the requirements of 10 CFR Part 20 as they apply to the protection of its own personnel.

IX – PLANT OPERATION

Before the

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For Licenses for a Spent Fuel Processing Plant
Under Sections 53, 63, 81, 104(b), and 185 of the
Atomic Energy Act

AEC Docket No. 50-201

Submission No. 21 - Final Safety Analysis Report

Paragraphs 9.97 through 9.117 of Section IX of the Safety Analysis,
Revision of Table of Contents for Chapter IX, and Paragraph 1.91

October 26, 1964

Before the

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For Licenses for a Spent Fuel Processing Plant
Under Sections 53, 63, 81, 104 (b), and 185 of the
Atomic Energy Act

AEC Docket No. 50-201

Submission No. 18 - Final Safety Analysis Report

Paragraphs 9.0 through 9.96 of Section IX of the Safety Analysis

IX PLANT OPERATION

9.1 Detailed in this section are the following items: The organizational make up of the Spent Fuels Reprocessing Plant; aspects of administrative control and procedures in various operations of the plant; Training Program, Health and Safety Program; Fire Safety Program and Emergency Procedures; the uses of the Operating Procedures and Letters of Authorization; a discussion of Maloperation; and the use of Maintenance Procedures.

Organization

Plant Manager

9.2 The Plant Manager is responsible for all activities at the plant and is, therefore, concerned with all aspects of plant operation. The more important areas include production, technical services, health and safety, and nuclear safety.

Production Manager

The Production Manager is responsible for carrying out production in accordance with approved procedures and accepted Health and Safety standards.

Health and Safety Director

The Health and Safety Director serves in a police and guidance capacity to assure conformance to approved Standard Operating Procedures and to advise in plant operations from a Health and Safety standpoint.

Technical Services Manager

The Technical Services Manager is concerned with the technical soundness of the operations proposed, the surveillance of material, and particularly, as a member of the criticality group, the maintenance of a critically safe system. He generates applications for license revisions. He reviews proposed SOP's and Letters of Authorization to confirm compliance with the license.

Plant Criticality Committee

9.3 The Plant Criticality Committee consists of the Plant Manager, the Technical Services Manager, the Health and Safety Director and the Production Manager. This committee sits in individual judgment on all SOP's and Letters of Authorization. Each member satisfies himself that the proposed procedure is in compliance with approved Health and Safety policies and that no criticality problem is involved. Each member gives particular attention to the function that he represents. The usual sequence for review is: 1. Production Manager 2. Technical Services Manager 3. Health and Safety Director and 4. Plant Manager.

Administration

9.4 The main function of management is to safely and economically administer all operations relative to the plant. The Plant Manager, who has overall responsibility for Plant Operations, has delegated certain responsibilities as enumerated in paragraph 9.2. In addition to the above, other delegation of responsibility is as shown on the Administrative Organization Chart, Figure 9.4.

Operating Procedures and Letters of Authorization

9.5 Processing of all special nuclear material handled under the license is done in accordance with the criteria set forth in the license. All operations in the Spent Fuel Reprocessing Plant are done in accordance with approved operating procedures which define the methods to be used and incorporate criteria contained in the license.

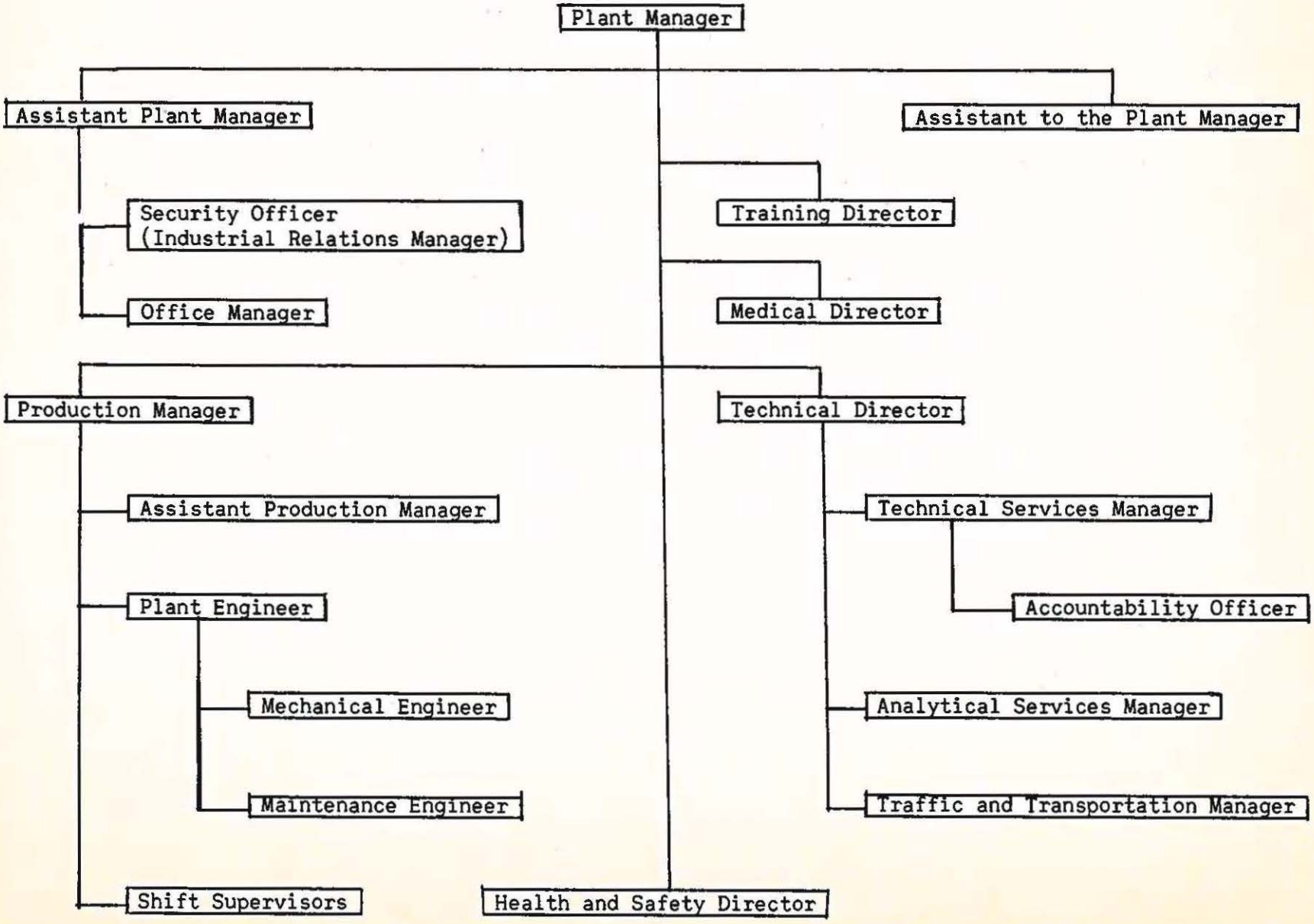
9.6 It will be the responsibility of each employee to read, understand and follow explicitly the directions contained in the Standard Operating Procedures for jobs which he is called upon to do. It is the responsibility of each supervisor to know the Standard Operating Procedures which apply in his area, to have copies of these SOP's available for employees to read and to be certain that individuals under his supervision read and understand each procedure. It is Management's responsibility to review and re-issue SOP's as necessary to reflect changes in the process and to insure that the instructions contained in SOP's represent a safe and efficient method for accomplishing the work.

9.7 Special nuclear material is received into the plant and processed by Approved Letters of Authorization which state the operating procedure(s) to be used, special handling where required, the customer for whom the processing is being done, the container(s) in which the material will be found and the material to be used as to type, enrichment and weight. Before a Letter of Authorization can be used it is independently reviewed by each member of the Plant Criticality Committee to assure its conformance with approved license criteria. It is the responsibility of each employee to read, understand and follow explicitly the directions contained in the Letter(s) of Authorization. It is the responsibility of each supervisor to have a copy of the Letter(s) of Authorization available for individuals under his supervision to read and to be certain that they read and understand the instructions contained in the Letter(s) of Authorization. It is Management's responsibility to issue Letter(s) of Authorization thereby scheduling work throughout the plant. Normally, process engineers under the supervision of the Production Manager or the Technical Services Manager draft the procedures or authorization. These engineers serve also in technical liaison and guidance in production and they conduct and supervise engineering development.

9.8 The general administrative philosophy will be to establish standard procedures for as many situations as possible and to control the effectiveness of these procedures by means of regularly maintained logs and check-off lists. These procedures, together with their

Figure 9.4

Plant Organization Chart



supporting logs and check-off lists, will be subject to regular, but random, inspection by higher levels of authority. For instance, certain routine examinations and measurements will be carried out daily according to approved check-off lists and duly logged. In these cases the next higher level of supervision will, once a week at a random time, follow through the specific procedure and determine that it is being properly carried out. Once every two months the next higher level of supervision will do likewise. Once a year these same procedures will be observed by the highest level of authority. Personal responsibility will be emphasized by having each one of these inspections recorded by signature and date; the logs will be kept as a permanent record. In addition, duty lists, addresses and telephone numbers will be maintained, and selected groups of off-duty personnel, at all levels of authority and skill, will be required to keep the plant informed on their whereabouts at all times for emergency call.

Training of Plant Personnel

9.9 The initial staff cadre will be largely made up of people with extensive experience in the handling, processing and monitoring of radioactive materials. This group, under the Training Director, will conduct the training courses for all additional employees. The curriculum (see Appendix 9.9) will be directed toward the education of certain plant personnel in the processes and related operations in such detail as to ensure complete familiarity with the equipment, its function and competence in its operation.

9.10 It is the intent of the training program to enable process operators to successfully satisfy AEC requirements for operators' licenses by test and examination. Approximately 75 operators will be so trained by permanent or temporary staff members. Initially, three types of operators will be trained for work in three different types of areas:

1. Manipulative Processing Operations

2. Chemical Processing Operations

3. Control Operations

The cadre, in addition to serving as the faculty, will take these and additional courses designed to satisfy AEC requirements for Senior Operators' Licenses. Certain employees such as watchmen, secretaries, etc. will be exempted from most of the more technical aspects of the curriculum but all employees will be exposed to a radiological familiarization curriculum. Written examinations, graded according to level of responsibility and work exposure, will be conducted.

9.11 The curriculum will include an introduction (comprising details of the background and descriptive material of the plant) details of the processes, health and safety, instrumentation, equipment description and usage, mechanical manipulation, process control, process maloperation, decontamination procedures, waste treatment, emergency measures, accountability, economic and criticality considerations and lay chemistry and

physics associated with reactor operations and chemical reprocessing. In addition, the curriculum for the cadre and others preparing for Senior Operators' licenses will include the conditions and limitations in the facility license, the design and operating limitations in technical specifications, the mechanism for any changes in the limitations in the license or specifications and more advanced study of chemistry and radioactivity. The training program for plant personnel will be a continuing one. Regular process operators will be given, periodically, a reorientation exposure to radiation safety and to processes and equipment involved in their particular plant specialty. New employees will be indoctrinated by training as are the initial employees and will be required to pass the same NFS examinations in addition to AEC license examinations.

Training of Outside Organizations

9.12 Partly as a matter of public relations but primarily to obtain effective and non-panicky assistance if an emergency requiring their cooperation should develop, local town, county, state police officers, fire departments of the area, civil defense organizations and elected officials will be invited to lectures at the plant. The subjects covered will be mainly those connected with protection of the public and will be designed to establish methods of liaison and cooperation if desirable and necessary under hypothetical emergency conditions so that assistance is most effective and radiological hazards to outsiders are minimized.

Health and Safety Program

9.13 The Health & Safety Department is charged with the responsibility for protecting plant personnel from all job hazards and the public from hazardous quantities of radiation and radioactive materials. Within the scope of this responsibility the Health & Safety Department will:

1. Monitor for radiation and contamination all plant areas and operations; (see Appendix 9.13 for equipment)

2. Monitor for radiation and contamination, areas external to the plant;

3. Approve procedures for work with radioactive materials;

4. Establish emergency procedures;

5. Establish liaison with all other departments and advise them in matters pertaining to health and safety;

6. Supervise the receipt and shipment of all hazardous materials;

7. Provide curriculum and teach aspects of the health and safety program;

8. Establish and maintain plant fire brigades trained to cope with radiation area fires;

9. Conduct a continuing safety training program for all employees;

10. Conduct inspections of all areas for fire and safety hazards and institute corrective action when necessary;

11. Maintain complete, accurate records of personnel exposure, radiation-contamination conditions in and around the plant, and perform radiation instrument calibration.

Health and Safety Organization

9.14 Specific responsibilities for members of the Health and Safety group are as follows:

Health and Safety Director

Plan, organize and supervise the work of the department. Maintain close liaison with the Medical Director advising and seeking advice concerning the employees' health and welfare. Maintain close liaison with other departments and advise them in matters pertaining to health and safety. Maintain complete, accurate records of plant, personnel and environmental radiation-contamination conditions. Administer health and safety aspects of training programs. Organize and train plant fire brigades. Inspect and maintain fire fighting and emergency equipment. Prepare material for use in safety meetings. Conduct fire and safety inspections.

Lead Technician

Perform routine and non routine monitoring tasks as directed. Write complete, accurate reports of conditions observed. Calibrate and check monitoring instruments. Obtain and count air samples. Perform safety inspections. Participate in shift safety training programs and safety meetings. Maintain exposure and survey record files. Check gamma dosimeters and record results. Prepare film badges for distribution and processing and record results.

Technician - Medical

Perform routine and non routine medical tests on employees including processing bio-assay specimens. Receive environmental and monitoring samples and prepare them for counting.

Technician - Shift

Perform routine and non routine monitoring and inspection tasks as directed. Participate in shift safety training programs and safety meetings. Write complete, accurate reports of activities.

Radiation Area Work Procedures

9.15 All radiation area work is governed by procedures approved by responsible persons in Production, Plant Engineering and Health and Safety. It is the intent of these procedures, in accordance with NFS policy, to incorporate sound industrial safety practice and to maintain exposure of employees to ionizing radiation and radioactive contamination at a level below the limits stated in 10 CFR 20.101 and appendix B, through the use of monitoring, decontamination and shielding techniques and through the use of protective clothing, respiratory protective devices and other safety equipment as required.

9.16 For the purpose of defining radiation areas, the following zones are established:

Zone I All areas beyond the site perimeter boundary;

Zone II All areas within the site perimeter boundary which are normally free of radiation--contamination in excess of 500 d/m alpha and 0.05 mrad/hr beta-gamma;

Zone III All areas within the site perimeter boundary which may have detectable radiation-contamination but in which the radiation level is normally less than 100 mrem/hr and the contamination level is not significant;

Zone IV All areas within the site in which the radiation level exceeds 100 mrem/hr or in which significant contamination exists.

Zoning of the plant and site will be the responsibility of Health and Safety.

9.17 The General Regulations for Radiation Area Work will apply to all work procedures. See Appendix 9.17 for listing of equipment.

General Regulations

The minimum requirements for protective clothing are:

a. For entry to Zones I and II, no protective clothing is required;

b. For entry to Zone III areas, for inspection only, the minimum protective clothing required shall be: Laboratory coat, shoe covers and gloves;

c. For entry to Zone III areas to perform work, the minimum protective clothing required shall be: Coveralls, shoe covers, gloves and cloth hat;

d. Protective clothing required for entry to Zone IV will be specified on a Special Work Procedure. No one will be permitted to enter a Zone IV area until a Special Work Procedure has been completed and signed and all provisions of that procedure have been implemented;

e. Respiratory protection requirements will be posted in the "hot" lobby.

The minimum requirements for personnel monitoring are:

a. For entry to the Plant, the minimum requirement for personnel monitoring shall be; --Badge

b. For entry to Zone III areas, the minimum requirement for personnel monitoring shall be;--Badge, dosimeters and dose rate type radiation survey meter.

c. For entry to Zone IV areas, the personnel monitoring requirements will be specified on the Special Work Procedure.

The exiting procedure is:

a. When leaving Zone IV areas the minimum requirement for personnel survey shall be A complete clothing and body survey by Health and Safety Personnel;

b. When leaving Zone III areas the minimum requirement for personnel survey shall be A complete self survey at the station monitors located at the Zone III - Zone II boundary;

c. When leaving Plant Zone II the minimum requirement for personnel survey shall be A hand and shoe check using the hand and shoe counters and station monitors in the building lobby. This survey shall also be made before entering the building lunch room.

The rules for radiation area conduct are:

a. No smoking, eating, drinking, or chewing shall be permitted in Zones III and IV. Zone II Plant areas in which smoking is not permitted will be so designated;

b. Every surface and every piece of equipment in Zones III and IV and every tool or article taken into these Zones shall be regarded as being contaminated until surveyed and released by a representative of Health and Safety;

c. All the provisions of applicable work procedures shall be read, understood and followed explicitly by the personnel performing the work;

d. Each employee is responsible for the care and treatment of equipment issued to him and for his conduct in the performance of assigned work. Careless or willful mishandling of equipment or misconduct on the job will not be tolerated and will constitute grounds for dismissal.

9.18 For work of a routine nature in areas normally free of significant radiation and/or contamination and where conditions are known and the work to be performed will not cause any significant change in these conditions, work is governed by Extended Work Procedures which may be modified or terminated at any time by Health and Safety personnel. Such Extended Work Procedures are given a date of termination not exceeding twelve months from the date of issue. On, or in advance of, the date of termination, the procedure is reviewed by responsible persons in Plant Engineering, Production, and Health and Safety, changed as necessary to reflect current working conditions, and re-issued with a new termination date.

9.19 For work of a special or unusual nature or work in areas or on equipment which does involve significant radiation-contamination, a Special Work Procedure is issued. Each Special Work Procedure is valid for one shift only. Approval of responsible persons in Plant Engineering, Production and Health and Safety is required prior to the start of any work and before work can continue on succeeding shifts.

Job Planning and Scheduling

9.20 Each day responsible representatives of Plant Engineering, Production and Health and Safety meet to plan and schedule work for the following day. A Work Schedule is prepared and distributed and Special Work Procedures are prepared and approved in advance of the work. The Work Schedule lists the personnel assigned to each task, the time and place to meet for each job, the estimated duration of each job, the applicable procedures governing the work and other information of general interest.

9.21 The Plant Engineer is responsible for:

a. Estimating the time and manpower required to accomplish each maintenance job;

b. Assigning maintenance personnel to each scheduled maintenance job;

c. Assuring that all maintenance personnel read and understand applicable work procedures and are thoroughly trained in radiation-contamination work;

d. Assuring that scheduled maintenance personnel understand what work is to be accomplished and that the proper tools and equipment, in good condition, are available in advance of the job;

e. Assuring that assigned maintenance personnel are available at the place and time indicated on the schedule.

9.22 The Production Manager is responsible for:

a. Establishing priority of maintenance in the plant;

b. Determining what the effect will be of scheduled maintenance work on plant operations;

c. Arranging for equipment or area shutdown as necessary to accomplish the scheduled work;

d. Arranging for pre-maintenance decontamination and/or shielding as required;

e. Assigning operating personnel to scheduled jobs as required;

f. Assuring that all operating personnel read and understand applicable work procedures and are thoroughly trained in radiation-contamination work;

g. Assuring that operating personnel understand what their duties will be for each scheduled job and that the necessary equipment, in good condition, is available in advance of the job;

h. Assuring that assigned operating personnel are available at the place and time indicated on the schedule;

i. Issuing the work schedule following each planning and scheduling meeting.

9.23 The Health and Safety Director is responsible for:

a. Determining what radiation-contamination conditions and/or other special hazards will be encountered in performing the scheduled work;

b. Determining whether or not a Special Work Procedure will be required for each scheduled job and if not, which Extended Work Procedure will apply;

c. Determining requirements for protective clothing and/or other safety equipment for scheduled work and assuring that such equipment, in good condition, is available in advance of the work;

d. Scheduling and leading a pre-job conference if required;

e. Assigning Health Physics personnel to scheduled jobs as required;

f. Assuring that all Health Physics personnel read and understand applicable work procedures, are thoroughly trained in all phases of radiation-contamination work and are trained and equipped to respond to unusual or emergency conditions;

g. Assuring that assigned Health Physics personnel are available at the place and time indicated on the schedule;

h. Initiating Special Work Procedures following each planning and scheduling meeting.

Unconditional Release

9.24 Release surveys of equipment are the responsibility of Health and Safety. Any item leaving Zone IV or Zone III to go to Zone II or Zone I or any item leaving the plant site from any Zone, must be accompanied by a completed Unconditional Release. The original of the release accompanies the equipment, and one copy (in the case of an item leaving the plant site) is presented to the Plant Security Guard who is responsible for enforcing this procedure. This procedure also applies to commercial vehicles and railway cars. The Unconditional Release states the radiation-contamination levels on the items described, and releases them with no conditions or restrictions as to their use.

Conditional Release

9.25 The use of a Conditional Release is normally restricted to equipment which is not to leave Zone III. For example, a process pump which is to be taken to the Equipment Decontamination Room or the Maintenance Shop for repair will require a Conditional Release. The Conditional Release describes the item released, lists the radiation-contamination status of the item and lists any special precautions which must be taken for handling, dismantling, and repairing the item.

Lock and Tag Procedure

9.26 The Lock and Tag Procedure is used to lock out valves, controls, and switches, the unauthorized or inadvertent use of which could cause process upset, damage to facilities and equipment or personal injury. Each department will have its own locks and will be responsible for applying

locks to equipment as required for employee protection even if this practice results in several locks on the same switch. The responsibility for removing locks will rest with the department head (or his delegated assistant) of the department responsible for applying the lock. Non compliance with this provision will not be tolerated. Maintenance locks are normally applied only during maintenance work on equipment and are removed when the work is completed. The tags are used to indicate the reason for the lock and to warn all personnel of the possible consequences of violating this procedure.

Safety Hazard Tag Procedure

9.27 Any NFS employee is responsible for tagging or posting any equipment or condition which represents a safety hazard and/or unsafe working condition. After taking such action he should notify his foreman or supervisor so that the condition may be corrected promptly. The Supervisor or Foreman shall notify the Director of Health and Safety.

Radiation and Contamination Protection

9.28 In this paragraph there are discussed a number of administrative limits of radiation exposure for the NFS Plant. It is expected that these limits may be modified as plant experience dictates. NFS employees may be exposed to radiation up to the limits stated in the following table with the approval of the employee's immediate supervisor:

Table 9.28

Rems Per Calendar Quarter

a. Whole body; head and trunk; active blood forming organs; lens of eyes; or gonads---------- 1-1/4

b. Hands and forearm; feet and ankles----------18-3/4

c. Skin of whole body---------- 7-1/2

Whole body exposure to penetrating radiation in any 24 hours period shall be limited to 0.1 rem or, if approved in advance by the Health and Safety Director, 0.2 rem. Planned single exposures in excess of 0.2 rem must be approved in advance by the Plant Manager.

In emergencies involving the life of personnel, it shall be the responsibility of the NFS Senior representative present to determine and authorize, if such be his decision, entry into higher fields of radiation.

9.29 The whole body dose and skin dose is available from badge readings. The dose to extremities is controlled in the field. If the dose rate to the hands and forearms or feet and ankles is more than 15 times the dose rate to the whole body, the time limit for the work is based on the dose rate to the extremities. With prior approval of the Plant Manager and the individual concerned, an employee of NFS may be

permitted to receive a dose to the whole body greater than that permitted under paragraph 9.28 provided that:

a. During any calendar quarter the total whole body dose shall not exceed 3 rems; and
b. The dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems where "N" equals the employee's age in years at his last birthday; and
c. The employees accumulated occupational dose to the whole body has been determined using Form AEC-4, in accordance with the instructions in paragraph 20.102 of 10 CFR-20.

9.30 The consequence for intentionally causing erroneous film badge or dosimeter readings is dismissal.

9.31 NFS employees, who have been certified in the use of radiation monitoring instruments by the Health and Safety Director, may in the course of their normal duties, self monitor in areas where the dose rate does not exceed 100 mr/hr, except Zone IV areas. In areas in which the dose rate exceeds 100 mr/hr or in all Zone IV areas monitoring for any entry shall be by Health and Safety Technicians. In no case shall employees enter an area in which the dose rate exceeds 2 r/hr unless prior approval of the Plant Manager has been obtained. (See 9.28)

Maximum Permissible Levels of Radioactivity

9.32 The maximum allowable surface contaminations for the West Valley Plant are shown in Table 9.32a. The Maximum Permissible Concentrations in air of some radionuclides expected to be encountered in the West Valley Plant are shown in Table 9.32b. The Maximum Permissible Concentration in on-site, nonpotable water in Buttermilk Creek of some radionuclides expected to be encountered at the West Valley Plant are shown in Table 9.32c.

Air Sampling

9.33 The air sampling program provides for the evaluation of alpha and beta-gamma air contamination in all building areas, the plant site and the site perimeter. Included in the program are 54 in-plant area particulate samplers, 19 remote in-cell particulate samplers, 7 in-plant continuous air monitors, 1 plant site sampler and 3 site perimeter air monitors. This equipment is described in Appendix 9.33, is located as per Figure 6.67, and discussed in Paragraphs 6.66 to 6.67.

9.34 The filter paper used for particulate sampling is Whatman #41 or equal, two inches in diameter. Whatman #41 filter paper has a collection efficiency of 98 per cent for 0.18 micron particulate or larger at a flow velocity of 50 centimeters per second. To obtain this flow velocity a minimum flow rate of 60 liters per minute is used for in-plant air samplers. Self absorption in Whatman #41 paper is zero for beta and about 0.3 for alpha.

Table 9.32a

Maximum Allowable Surface Contamination for West Valley Plant

	Smearable		Non-Smearable	
Surface	Alpha d/(m)/(100 cm^2)	Beta-Gamma As Shown/100 cm^2	Alpha d/(m)/(100 cm^2)	Beta-Gamma As Shown/100 cm^2
Skin	No Detectable		500	100 c/m
Personal Clothing	No Detectable		500	100 c/m
Plant Clothing	500	100 c/m	1,000	2,000 c/m
Plant Vehicles	500	100 c/m	1,000	5,000 c/m
Commercial Vehicles	500	100 c/m	500	0.4 mrad/hr
Zone I	Zone I limits are per 10 CFR - 20, Appendix B, Table II			
Zone II	500	100 c/m	500	100 c/m
Zone III	5,000	10 mrad/hr	5,000	100 mrad/hr
Zone IV *	50,000	2 r/hr	50,000	2 r/hr
* For personnel entry				
Conditional Release	1,000	5,000 c/m	5,000	10 mrad/hr
Unconditional Release	500	100 c/m	500	100 c/m

Table 9.32b

Maximum Permissible Concentration (μc/ml)

Mixed Fission Products	
No respiratory protection	1×10^{-9}
Full face filter mask	2×10^{-8}
Supplied air mask	Above 2×10^{-8}
Strontium-90	
No respiratory protection	3×10^{-10}
Full face filter mask	6×10^{-9}
Supplied air mask	Above 6×10^{-9}
Cesium-137	
No respiratory protection	1×10^{-8}
Full face filter mask	2×10^{-7}
Supplied air mask	Above 2×10^{-7}
Plutonium 239	
No respiratory protection	2×10^{-12}
Full face filter mask	4×10^{-11}
Supplied air mask	Above 4×10^{-11}
Natural Uranium	
No respiratory protection	6×10^{-11}
Full face filter mask	1×10^{-9}
Supplied air mask	Above 1×10^{-9}
High Enriched Uranium	
No respiratory protection	1×10^{-10}
Full face filter mask	2×10^{-9}
Supplied air mask	Above 2×10^{-9}
Iodine-131	
No respiratory protection	9×10^{-9}
Supplied air mask	Above 9×10^{-9}
Krypton-85	
No respiratory protection	1×10^{-5}
Supplied air mask	Above 1×10^{-5}

Footnotes to Table 9.32b

Maximum Permissible Concentrations for other radio-nuclides is as indicated in 10 CFR-20, Appendix B, Table I. When a mixture of radionuclides is encountered and the identity and concentration of each radionuclide in the mixture are known, the Maximum Concentration is derived as follows: If radionuclides A, B, V, are present in concentrations Ca, Cb, Cc and the applicable MPC's are MPCa, MPCb, and MPCc respectively, than the concentrations shall be limited so that the following relationship exists:

$$\frac{Ca}{MPCa} + \frac{Cb}{MPCb} + \frac{Cc}{MPCc} \leq 1$$

Table 9.32c

Maximum Permissible Concentration (μc/ml)

	On-Site-Buttermilk Creek	Off-Site
Cesium-137	4×10^{-4}	2×10^{-5}
Cobalt-60	1×10^{-3}	3×10^{-5}
Tritium	1×10^{-1}	3×10^{-3}
Iodine-131	6×10^{-5}	2×10^{-6}
Plutonium-239	1×10^{-4}	5×10^{-6}
Ruthenium-103	2×10^{-3}	8×10^{-5}
Ruthenium-106	3×10^{-4}	1×10^{-5}
Strontium-90	4×10^{-6}	1×10^{-7}
Natural Uranium	5×10^{-4}	2×10^{-5}
High Enriched Uranium	8×10^{-4}	3×10^{-5}

Footnotes to Table 9.32c

Maximum Permissible Concentrations for other radionuclides are as stated in 10 CFR-20, Appendix B. When mixtures of radionuclides are encountered and the identity and concentrations of each is known, the procedure stated in the footnote to Table 9.32b is used to determine the MPC.

9.35 Air samples are collected and analyzed for radioactive material according to the schedule shown in Table 9.35. This schedule is subject to revision as experience is gained in operating the plant. Continuous air monitors are used in some occupied areas to provide an immediate alarm should high air contamination exist. The other remote samplers will be used occasionally to obtain very short, spot samples of air contamination conditions in the cells. These remote samplers are:

Miniature Cell

General Purpose Cell

Chemical Process Cell

Mechanical Cell

X-Cell 1

X-Cell 2

X-Cell 3

Product Purification and Concentration

Chemical Process Cell Crane Decontamination Area

Process Mechanical Cell Crane Decontamination Area

9.36 As each sample is removed from the sample head it is placed in an envelope which is marked with the sampler location, date-time started, date-time changed and the flow rate. When all samples have been changed, according to the schedule, they are brought into the Health Physics Lab, removed from the envelopes, placed in planchets and surveyed with portable beta-gamma and alpha detection instruments (Appendix 9.36). Any samples which show unusually high activity are segregated for special handling and prompt attention in the counting room.

9.37 Alpha, Beta proportional counters (Appendix 9.37) are used to analyze in-plant air samples. All samples receive a one minute alpha, beta count as soon as possible after being delivered to the counting room. The beta/alpha ratio is determined based on this count. Since the beta/alpha ratio is constant for natural activity, it may be possible at this time to make a preliminary estimate of the amount of long-lived emitters on the sample. The concentration of beta emitters on the sample will be determined based on the initial count. This is accomplished as shown in Appendix 9.37a. All samples receive a five minute alpha count five to seven hours after sampling and a second five minute alpha-beta count 23 to 25 hours after sampling. These counts are used to calculate the alpha counts due to long-lived alpha activity (product) on the sample. This is accomplished as shown in Appendix 9.37b. Any samples which, on the 24 hour count, show less than 1 c/m alpha and less than 1800 c/m beta are

Table 9.35

Start Up Schedule for Air Sampling

Shiftwise	Daily	Weekly
Hot Lobby	GPC Operating Aisle-west	Third Floor Office
Mechanical Operating Aisle-west	GPC Operating Aisle-east	Second Floor Office
Ram Equipment Room	Lower Warm Aisle-west	Main Lobby
Chemical Viewing Aisle-north	Lower Warm Aisle-east	Maintenance Shop
Ventilation Wash Room	Acid Recovery Pump Aisle	Utility Room
Process Sample Enclosure-1	Scrap Removal	Manipulator Repair Area
Process Sample Enclosure-2	Mechanical Operating Aisle-east	Product Packaging CAM
Analytical Aisle	X-Cell entrance air lock	Fuel Storage CAM
Extraction Sample Aisle-west	U-Product cell	EDR Viewing Station
Extraction Sample Aisle-east	Product purification cell	MCR Air Lock
Ventilation Exhaust Cell	Product Packaging 1, 2 and 3	Ventilation Supply Room
Pulser Aisle	Fuel Storage 1 and 2	Hot Lobby CAM
	Chemical Viewing Aisle-south	Mechanical Operating Aisle CAM
	Equipment Decontamination Room	VSR Access Aisle
	Chemical Operating Aisle-north	Off Gas Cell-3
	Chemical Operating Aisle-south	Analytical Cell Decon. Area
	Lower Extraction Aisle-east	Alpha Lab CAM
	Upper Warm Aisle-west	Lab Access Aisle
	Upper Warm Aisle-east	Control Room
	Off Gas Cell-2	Extraction Chemical Room CAM
	OGC-ARC Aisle	Plant Area
	Chem Lab-east and west	Perimeter-1
	Product Lab	Perimeter-2
	Emission Spec. Lab	Perimeter-3
	Mass Spec. Lab	GPC Crane Room
	Alpha Lab	Mechanical Crane Room
	Stack Sampler	Chemical Crane Room
	Upper Extraction Aisle-west	
	Upper Extraction Aisle-east	
	Extraction Chemical Room-east	
	Laundry	

discarded. These counts at maximum counting error, represent about 1% of MAC for plutonium-239 and strontium-90 respectively. Samples which exceed either or both of the counting limits will be held for a final count. The final 30-minute count on in-plant air samples is taken a minimum of four days after sampling to allow the natural activity to decay essentially to zero. All of the alpha counts are assumed to be counts due to product and the concentrations are calculated as follows:

Alpha

$$\mu c/ml = \frac{c/m\ (1.31 \times 10^{-12})}{M^3}$$

Since the counting error for a 30-minute count at 95 percent confidence level is ± 10% at 10 c/m, the minimum detectable alpha concentration on a 24-hour sample is:

$$\frac{10\ (1.31 \times 10^{-12})}{86.4} = 1.5 \times 10^{-13} \text{ with } \pm 10\% \text{ accuracy}$$

Beta-Gamma

$$\mu c/ml = \frac{c/m\ (9.19 \times 10^{-13})}{M^3}$$

9.38 Some in-plant air monitors are moving-filter type and the filter tape is not normally analyzed in the counting room. Portions of the tape may be counted and/or gamma scanned if this information is needed.

9.39 The perimeter samples are changed weekly and are analyzed once as soon as practical after sampling and again four days after sampling. The samples are analyzed by counting for one hour in a low background alpha, beta proportional system (see Appendix 9.39a). The geometry of this system is 50% for beta and 35% for alpha. The background is about 1 c/m. The concentration of beta emitters is determined as shown in Appendix 9.39b and the concentration of alpha emitters is determined as shown in Appendix 9.39c. A log is kept of air sample results. These results become part of the permanent record of radiation-contamination conditions in and around the plant.

9.40 Radioiodine activated charcoal filters from the stack and perimeter stations are analyzed as follows: The filters are gamma scanned to determine if there are other gamma emitting isotopes present and in what proportion. Since the radioiodine filter is preceded by a particulate filter, there will normally be no interference from other isotopes. The radioiodine filter is then counted in one of the proportional counting systems and the concentration is calculated as shown in Appendix 9.40.

9.41 The numbers used in this section for geometry of counters and efficiency of and self absorption in filter paper are numbers furnished by the manufacturers. The method used to determine actual counter geometry is described in the calibration section (Appendix 9.37). The collection efficiency of Whatman #41 filter paper can be checked by using a membrane type filter behind the Whatman #41 filter to test the penetration under various conditions of use in the plant. The self absorption of alpha in Whatman #41 can be determined by counting a filter, dissolving the filter, evaporating the solution on a planchet and counting the planchet. The absorption correction then becomes; filter count/planchet count. These tests will be run on each batch of filter paper received.

Radiation - Contamination Survey Program

9.42 Beta-gamma film badges are supplied to each employee and all visitors to the plant through an arrangement with a commercial film badge processor. Badges are exchanged and read weekly for most personnel; monthly for administrative personnel. This schedule is subject to change as operating experience is gained. Immediate notification by phone or wire is given for badges which show a dose in excess of 100 mrem. Neutron monitoring is accomplished on an area basis. Neutron badges are placed in the product storage and product packaging and handling areas to establish and check the neutron dose rate in these areas. The neutron badges are changed monthly during startup but this may be changed to quarterly at a later date.

9.43 Each production employee and each visitor is issued a 0-200 mr gamma dosimeter which is read and the dose recorded during the shift following the shift on which it is used. The dose is recorded on the "Dosimeter Readings" form, and is transferred later to the "Exposure Record" card which is also used to record badge readings. The 5 x 8 inch card, designed to be used in a "Victor Visible" type file, contains all of the information required by AEC Form 5. Each card represents 13 weeks exposure data. See Appendix 9.43 for the "Exposure Record" card referred to above.

9.44 A limited number of self reading dosimeters are available for use during "hot" area decontamination and maintenance work. These dosimeters will be used as the second line of defense against overexposure. The primary control will be monitoring, by Health and Safety or by the individual performing the work, and timekeeping, by the individual or by a timekeeper assigned to the job.

9.45 Health and Safety responsibility for product shipments entails checking the shipping papers for Production signature approval, for product specifications, accountability certification and surveying the shipping containers to insure conformance with all applicable federal, state, and local regulations. The signature of an authorized Health and Safety representative on the shipping papers will constitute approval to ship.

9.46 With the exception of Zone IV egress, personnel surveys are the responsibility of each employee. Health and Safety will audit the frequency and adequacy of such surveys. Personnel found in a Zone II or Zone I area with contaminated clothing may be subject to dismissal.

9.47 A regular schedule of routine surveys will be performed by Health and Safety. The routine survey program is designed to supplement the reports of radiation contamination conditions which are encountered during maintenance and other work, and to insure that all plant areas are surveyed on a regular basis. Each routine survey is described in considerable detail on the "routine survey" form (See Appendix 9.47) which will serve as a guide for the Health and Safety personnel performing the survey. A list of routine surveys is shown in Table 9.47. A written record is made of every survey performed by Health and Safety personnel. This record which is executed on pre-numbered survey log sheets becomes part of the permanent record of radiation-contamination conditions in and around the plant.

Environmental Survey Program

9.48 The environmental survey program, pre-operational and post-operational is divided into three categories:

1. Atmospheric monitoring including air particulate monitoring;

2. Water monitoring including surface and ground water sampling;

3. Earth and biota monitoring including samples of silt, mud, plankton, fish and shellfish from Buttermilk Creek and Cattaraugus Creek; soil, vegetation and milk samples from the site and surrounding area and small game from the site.

9.49 The pre-operational program is divided into two phases; the first phase, started in the spring of 1963, to establish on site gross activity background with a few analyses for specific isotopes and the second phase, starting in the fall of 1964, to include more analyses for specific isotopes. Phase II will continue into the post-operational period. Both Phases are detailed in Appendix 9.49. A summary of the Environmental Monitoring Program is presented in Tables 9.49a and 9.49b.

Waste Disposal Control Program--Gaseous Waste

9.50 Gaseous waste control is accomplished by treatment of waste gases before release, continuous monitoring at the point of release and environmental monitoring to determine the effect, if any, of released activity in the environment. Waste gas treatment is discussed in some detail in Section VI, Paragraphs 6.66 to 6.70. Prefilters, air scrubbers, silver reactors and high efficiency filters are used to minimize the amount of radioactive gases and particulates released routinely from the plant. It is anticipated that the routine releases will be well below the maximum allowable under applicable

Table 9.47

Routine Surveys

Survey No.	Title	Shift Assigned
S-1	Check Dosimeters and Record Results	1,2,3
S-2	Pick up Air Samples	1,2,3
S-3	Check Charts on Gamma Alarm, Sample System and Weather Monitoring System	1,2,3
S-4	Count Samples	1,2,3
D-1	Check Station Monitors and Hand Counters	3
D-2	Calibrate Instruments	3
D-3	Spot Check Laboratories	2
D-4	Survey Hot Lobby	3
D-5	Transfer Dosimeter Readings to Exposure Record Cards	1
D-6	Spot Check Sample Aisle, Pulser Aisle and Warm Equipment Aisle	3
D-7	Survey Lunch Room	2
D-8	Survey Step-off Pads	3
D-9	Prepare Control Samples for Counting	1
D-10	Prepare Environmental Samples for Counting	1
D-11	Spot Check Product, Packaging and Handling	2
W-1	Survey Alpha Lab	1
W-2	Survey Chem Labs	1
W-3	Survey Spec Labs	1
W-4	Survey Product Lab	1
W-5	Survey Zone III offices	3
W-6	Survey Mens Locker Room	2

Table 9.47 con't

Survey No.	Title	Shift Assigned
W-7	Survey Five Personnel	1,2,3
W-8	Survey Ventilation Penthouse	3
W-9	Survey Upper Warm Equipment Aisle	3
W-10	Survey Access Aisle	2
W-11	Survey Operating Aisles	2
W-12	Survey Sample Aisle	3
W-13	Survey Fuel Receiving & Storage	3
W-14	Survey Product Packaging & Handling	2
W-15	Survey Decontamination Area	3
W-16	Survey Scrap Transfer Area	3
W-17	Survey Health Physics Lab	1
W-18	Survey Mechanical Cell Viewing Area	2
W-19	Survey Laundry	3
W-20	Obtain Environmental Samples	1
W-21	Survey Womens Locker Room	1
W-22	Survey Warm Equipment Aisle	3
W-23	Survey Mobile Equipment	1
W-24	Survey Ventilation Equipment Rooms	2
M-1	Survey Analytical Viewing Area	2
M-2	Survey Instrument Shop	2
M-3	Survey Main Lobby	1
M-4	Survey Cold Chemical Penthouse	2
M-5	Survey Chemical Process Cell Viewing Area	2

Survey No.	Title	Shift Assigned
M-6	Survey Maintenance Shop	2
M-7	Survey Guard House	1
M-8	Survey Tank Farm	1
M-9	Survey Burial Ground	1
M-10	Survey Remote Operating Station	2
M-11	Survey First Aid	1
M-12	Obtain Environmental Samples	1
M-13	Autoradiograph Environmental Air Samples	1
Q-1	Survey Utility Building	3
Q-2	Survey Roads, Walks, Parking lot and R. R. Spur	2
Q-3	Survey Storage Lagoon and Hardstand Areas	1
Q-4	Survey Dry Wells	1
Q-5	Survey Zone II offices	3
Q-6	Obtain Environmental Samples	1

S = Shiftwise
D = Daily
W = Weekly
M = Monthly
Q = Quarterly

Table 9.49a

Environmental Monitoring

Phase I - Type of Analysis

	Weekly	Monthly	Semi-Annually
Air Sampling 3 Perimeter 1 Plant Site	Gross Alpha Gross Beta- Gamma		
Rain & Snow 1 Plant Site		Gross Alpha Gross Beta- Gamma, Tritium	
Surface Water 1 Erdman Brook 1 Buttermilk Creek 1 Cattaraugus Creek		Gross Alpha Gross Beta- Gamma, Tritium	
Mud and Silt 1 Erdman Brook 1 Buttermilk Creek 1 Cattaraugus Creek		Gross Alpha Gross Beta- Gamma	
Well Water 1 Plant Site		Gross Alpha Gross Beta- Gamma, Tritium	
Vegetation 3 Perimeter			Gross Alpha Gross Beta- Gamma I-131 Sr-90
Milk Neighboring Farm	Gross Beta Gamma	I-131 Sr-90	
Small Game 1 Plant Site			Gross Alpha Gross Beta- Gamma I-131 Sr-90

Table 9.49b

Environmental Monitoring

Phase II - Type of Analysis

	Weekly	Monthly	Semi-Annually
Air Sampling 3 Perimeter 1 Plant Site	Gross Alpha Gross Beta- Gamma	Gamma Scan	
Rain and Snow 1 Plant Site		Gross Alpha Gross Beta- Gamma Tritium	Sr-90
Surface Water 1 Erdman Brook 1 Buttermilk Creek 1 Cattaraugus Creek	Gross Alpha Gross Beta- Gamma, Tritium		
Mud and Silt 1 Erdman Brook 1 Buttermilk Creek 1 Cattaraugus Creek		Gross Alpha Gross Beta- Gamma	Sr-90
Well Water 1 Plant Site		Gross Alpha Gross Beta- Gamma, Tritium	
Vegetation 3 Perimeter		Gross Alpha Gross Beta- Gamma I-131	Sr-90
Milk 1 Plant Site	Gross Alpha Gross Beta- Gamma I-131		Sr-90
Fish and Shellfish 1 Cattaraugus Creek			Gross Alpha Gross Beta- Gamma I-131
Small Game 1 Plant Site			Gross Alpha Gross Beta- Gamma I-131 Sr-90

federal and state regulations. Spare units and automatic controls are used as necessary to prevent the escape of high level bursts of activity caused by major equipment failure.

9.51 A continuous stack gas monitor, described in Appendix 9.51, is used to detect concentrations of 3×10^{-12} μc/ml or less of gross beta-gamma particulate activity and about the same concentration of I-131. A significant increase in concentration of either particulates or radioiodine will cause an alarm in the plant control room. The exact alarm positions will be field selected based on operating experience; they will be kept at the lowest practical level to provide the earliest possible warning of off-standard conditions.

9.52 Environmental monitoring to determine the effects on the environment of waste gas disposal is concentrated in air sampling and sampling of soil, vegetation, milk and rainout. Three site perimeter continuous air monitoring stations are established to determine concentrations of radioactive particulates and radioiodine at these stations. One station is located 3,100 meters south-east of the plant, the second station is located 2,100 meters north-east of the plant and the third station is located 4,000 meters north-north-west of the plant. This places a monitoring station at either end of and adjacent to Buttermilk Valley and, according to prevailing wind patterns, will place one of the three monitors down wind of the stack nearly 60 per cent of the time.

9.53 The routine soil, vegetation, milk and rainout sampling program is defined in Table 9.49 b. The entire sampling program is subject to change as operating experience is gained but it is expected that any changes will be minor in nature. Special samples will be analyzed if the stack monitor indicates an alarm condition. The weather monitoring stations, (see Appendix 9.53) will supply data which may be used to determine the direction of travel of stack fumes and the distance at which the maximum ground level concentration occurs. A mobile motor-generator sampler set will allow sampling down wind of the stack regardless of wind conditions.

Waste Disposal Control Program -- Liquid Waste

9.54 The primary control of high level liquid waste is in the facilities provided. The waste tank itself, the concrete saucer for secondary containment, the impervious "silty till" formation and the spare tank all contribute to a high degree of confidence in the system. See Paragraphs 5.50 - 5.56, 7.10, 7.14 -7.18, 7.25 - 7.37. Facilities are provided for monitoring or sampling in the annular space between the tank and the vault. Routine surveys will be performed in the wells located adjacent to the waste tanks. A continuous water sampler located near the confluence or Erdman Brook and Quarry Creek will serve as a third monitoring point of control of liquid waste.

9.55 Low level liquid waste will be discharged to Cattaraugus Creek via Erdman Brook and Buttermilk Creek. Waste water at a volume of about 40,000 gallons per operating day is received in the interceptor, batch neutralized if necessary, and discharged to a series of holding ponds. The interceptor volume is about 50,000 gallons and the ponds

provide holdup for 4,000,000 gallons or 100 operating days above the minimum overflow points. Overflow points between ponds are a valved line at two feet above the bottom to provide for solids collection, and an open overflow at one foot from the top. The discharge line to the creek is valved so the amount of waste discharged may be regulated.

9.56 Stream gauging and sampling stations are provided near the confluence of Quarry Creek and Erdman Brook and on Cattaraugus Creek. Gauging is performed in order to determine the rate at which waste solutions may be metered into Erdman Brook. Samples from these stations will be collected and analyzed weekly. Analyses will include gross alpha, beta and gamma, tritium and specific isotope analyses as required for control. (Appendix 9.56)

Waste Disposal Control Program--High Level Solid Waste

9.57 A burial area for waste generated in the plant will be maintained in an area north of the plant between the waste tank farm and the confluence of Quarry Creek and Erdman Brook. This area will be reserved for process scrap and discarded process equipment. Process scrap, fuel element end pieces and leached hulls, will be packaged in 30 gallon drums, loaded into a shielding cask on a carry-all type trailer and transported to the burial area. (See Paragraph 7.14.) At the burial area a truck mounted crane with remote controls, 100 feet away, will be used to lift the lid of the cask, remove the scrap drum and place it in the trench. At the end of each burial operation, which may require several trips, the crane clam attachment or front end loader, will be used to backfill where necessary to maintain an exposure rate at the security fence of 2 mrem/hr. The drums will be covered with sufficient dirt to reduce the exposure rate at the edge of the trench to 200 mr/hr. Final backfilling when the trench, or a portion of the trench, is full will be to a radiation level of 1 mr/hr or less. The minimum dirt covering will be four feet thick.

9.58 A similar procedure will be followed for burial of process equipment. The equipment, after decontamination, will be suitably packaged and loaded on the truck in the Equipment Decontamination Room, transported to the burial area, loaded into the trench with the crane and backfilled. Packaging techniques will vary depending on the equipment itself and the radiation-contamination conditions. Generally a sprayed-on coating or a covering of plastic film will be used.

Medical Program

9.59 The medical program, under the direction of the Medical Director, will consist of a very thorough pre-employment medical history and physical examination for each prospective employee. The medical history will be aimed at not only past illnesses and injuries but particular attention will be paid to history of past radiation exposure, allergies, blood dyscrasias, tumors and any evidence of emotional instability. The laboratory studies on all applicants will consist of a minimum of complete blood count, serology, urinalysis, chest x-ray and vital capacity determinations. Each employee will have a complete physical examination yearly.

A complete blood count will be done twice yearly; clinical urinalysis monthly. The pre-employment physical examination and laboratory studies will be repeated on each individual leaving the employ of the company.

9.60 Bio-assays will be scheduled for employees using an "across-the-plant statistical survey" plan. The number of times each employee is sampled each year and the type analyses performed will depend on his work location. Office employees annually, for total alpha and gross fission products; mechanical head end, extraction operators, Health & Safety technicians, maintenance and utility operators semi-annually for total alpha and gross fission products; product purification and packaging operators quarterly for plutonium and total uranium. Additional samples will be obtained to confirm any positive result and special samples will be obtained when inhalation or ingestion is suspected for any employee.

9.61 Thyroid monitoring of employees will be performed at least once each year in conjuction with the annual physical examination. Special monitoring will be performed as indicated by air sample counting results.

9.62 A dispensary will be maintained for care of ordinary minor on-the-job injuries. There will be facilities for intensive first-aid care of severe injuries such as burns, fractures and gross contamination with radioactive materials. Immunization against tetanus will be routine for all employees.

9.63 Close liaison with the Health and Safety Department will be maintained. The Medical Director will assist in health and safety training and indoctrination. He will review with the Health and Safety Director, all industrial radiation exposure records; air, water and plant radiation survey records. He will cooperate with the Health and Safety Director in plant inspections.
Radiation exposure data for each employee shall be kept on form AEC-5 as part of the permanent record of each employee.
A permanent check-off list shall be attached to each employee's permanent record covering all of the plants' requirements regarding physical examinations and personal radiation exposure recording and control as well as all requirements of 10CFR-20.

Emergency Procedure

Fire Protection Organization

9.64 The Health and Safety department has the primary responsibility for training personnel and auditing procedures and activities for fire prevention as well as for fire fighting. The fire fighting function will be carried out through shift fire brigades organized as indicated in Appendix 9.64.

Organization for Radiation Emergencies

9.65 There are a very large number of combinations of conditions which might constitute or cause an emergency. It is, therefore, not possible to prescribe inflexible procedures for emergency action. However, there are

broad categories of emergencies for which general procedures may be stated and certain general rules which apply in nearly all cases. In any radiation emergency, the Health and Safety Department has the primary responsibility to define the magnitude and extent of the problem and to recommend a course of action which will restore the affected areas promptly and safely.

9.66 In any radiation emergency the responsible group (Production or Analytical) in the area in which the emergency condition exists must take immediate steps to accomplish the following:

a. Protect plant personnel by evacuating affected areas and take action to confine the condition and eliminate or moderate the cause.

b. Notify the Health and Safety Director (or Technician on off shift) giving all possible details about the nature and location of the emergency.

c. If the emergency involves property damage, personal injury, significant radiation levels, production interruption, or possible off-site contamination, the following must be notified:

   Health and Safety Director
   Medical Director
   Laboratory or Production Manager
   Plant Manager & Assistant Plant Manager
   Assistant to the Plant Manager
   Security Officer
   Plant Engineer

d. Following the survey by Health and Safety, barricade and post the affected area to prevent inadvertent entry.

e. Devise a plan for restoring the area and assemble the required men and materials.

9.67 Generally, the following rules apply in handling an emergency condition:

a. If incident involves wreckage and a person is believed to be alive and trapped, make every possible effort to rescue him. The usual radiation rules may be abrogated upon the authority of the senior person present.

b. Segregate and detain for further examination those persons who have had possible contact with the radioactive material. Perform complete contamination surveys of such personnel and institute decontamination at once if significant exposure could result from a delay. Normally, it is best to leave skin decontamination to those persons with specific training in this function.

c. Remove injured persons from the scene with as little direct personal contact as possible. Limit first aid and medical procedures to

those that must be done promptly until the doctor is present.

d. Do only what is necessary to preserve life and property prior to the arrival of Health and Safety specialists.

e. Work within the framework of any applicable SOP's covering a specific type of emergency.

Plant Maintenance Program

9.68 The Nuclear Fuel Services maintenance program has been planned to insure continued safe operation of the plant, commensurate with Paragraphs 9.13 to 9.41, with a minimum of downtime consistent with economic considerations.

9.69 The routine inspection and maintenance program is similar to that for a normal chemical plant, except where modified to reflect more stringent requirements for the nuclear aspects of the plant. The maintenance program is based upon utilizing conventional methods and procedures for performing contact maintenance work. Special controls are incorporated to cover work within contamination and radiation zones. Work on contaminated equipment or systems is done under the surveillance of the Health & Safety Department which recommends required control measures. Careful planning, prewritten job procedures, and close coordination with Production and Technical Services Departments assure safe and efficient plant operation. Normal inspection contemplates periodic shutdowns to permit inspection and maintenance of those portions of the plant not readily accessible during routine operation.

9.70 Certain equipment is deemed vital to the safe and continuous operation of the plant. This equipment is defined as 1. equipment that could become critically unsafe from a nuclear standpoint, and 2. any malfunctioning piece of equipment which could reasonably require the shutdown of the plant.
A list, referred to as the Vital Equipment List initiated by the Production and Technical Services Departments and approved by the Plant Manager, is compiled and issued to the Production Department. (See 9.82) The list states the requirements to be met before the equipment is taken out of service, and what tests and requirements are to be met before the equipment is returned to service. All equipment not specifically designated on the Vital Equipment List is considered as non critical and may be taken out of service, repaired, and returned to service according to normal standard maintenance practice.

Organization

9.71 Maintenance work on Nuclear Fuel Services equipment and systems is performed by Plant Engineering. Plant Engineering is responsible for all mechanical, instrument and electrical maintenance work. Each of these categories is under the direction of a group leader. Close cooperation between these groups is maintained to facilitate scheduling, conserve manpower, and minimize downtime.

Under normal conditions, mechanical and electrical maintenance is accomplished on a day schedule, five days per week. Much of the routine instrument maintenance is carried out on a similar schedule; however, instrument technicians are normally on shift with operations personnel.

Plant Engineering Section

Personnel

9.72 Plant Engineering is composed of a plant engineer, mechanical engineers, maintenance mechanics, instrument technicians, and stenographer. The Plant Engineer is responsible for:

1. Planning, scheduling, and controlling personnel, materials, equipment and tools.

2. Initiating training and educational programs for maintenance personnel.

3. Establishing and supervising the maintenance of a readily accessible file of design and vendor information, parts data, preventive maintenance records, and historical records.

4. Supervision of all maintenance assignments, including instructions to cover safe working practices, radiation protection measures and approved maintenance repair procedures.

5. Making technical studies on maintenance of mechanical, instrument and electrical equipment, and making recommendations on design changes.

6. Preparing labor and material costs estimates for non routine work.

The Plant Engineer is primarily assisted by two mechanical engineers to whom any of the above responsibilities may be delegated. Technical support is available from the Technical Services and the Health and Safety Departments which will provide specialists as required.

Facilities

9.73 The Plant Engineering Section and shop facilities are organized primarily to perform field maintenance work. On site shop work consists basically of minor repairs, replacement of defective components and checkout of equipment. The bulk of the work is of short duration and minor complexity, and the shops are equipped accordingly. Machine, electric, instrument, pipe, carpentry and welding shops are provided. In cases where maintenance functions require facilities not provided at the site, privately operated shops in nearby Buffalo, New York will be utilized where possible.

Instrument Maintenance

Personnel

9.74 The maintenance of instrumentation and control systems is the responsibility of the Plant Engineer assisted by the Instrument Engineer. These responsibilities are as follows:

1. Adequacy of the maintenance facilities and the training of personnel to meet all requirements, both routine and emergency;

2. Planning and scheduling of all instrument maintenance in cooperation with mechanical maintenance personnel;

3. Establishment of a preventive maintenance program for all control systems and components, with particular emphasis on those involving the safety of the plant;

4. Planning and maintenance of a file system that contains the information necessary to analyze, design, order spare parts and components, apply preventive maintenance procedures and provide history of repairs on all equipment. This will be done in conjunction with mechanical maintenance.

Instrument Shop Facilities

9.75 The instrument shop is equipped with services, (water, air, electricity, tools and test equipment) necessary for the calibration and maintenance of either pneumatic or electronic instruments.

Maintenance Categories

9.76 Plant Engineering performs three categories of work; preventive maintenance and inspection, routine maintenance and non routine maintenance. Any of these categories of work may involve hazardous conditions due to radiation or contamination. The procedures used in performing this work depend on both the category of work and the degree of hazard involved due to direct radiation or contamination. These procedures will be subject to approval by the Health and Safety Director in those cases involving radiation hazards.

Preventive Maintenance and Inspection

9.77 The preventive maintenance program minimizes shutdowns and breakdowns by systematically inspecting equipment, making calibrations or adjustment, and scheduling repairs and overhauls before failure occurs. Each piece of equipment is studied thoroughly, and a schedule of routine inspections is determined and established under the following classifications:

a. A-Class: Major inspection (complete check of equipment;)
b. B-Class: A "middle-of-the-road" inspection. Usually made quarterly to semi-annually and, on occasions, monthly;
c. C-Class: A minor inspection (ordinarily visual and frequent.) Usually made monthly to quarterly and, on occasions, weekly.

As each piece of equipment is studied, a complete list of items to be checked on each inspection is made. A central control system indicates when inspections are due. If inspections do not interfere with normal plant operation, the inspections are scheduled and carried out in accordance with work loads in the section. Inspections that require shutdown of equipment or interfere with normal plant operations are coordinated with the Production Department. After an inspection is completed, information is transferred from the inspection sheet to a card as a continuing record. If any repairs are necessary, such repairs fall into the category of routine maintenance and are scheduled according to the urgency required.

Routine Maintenance

9.78 Routine maintenance includes all maintenance work on equipment or systems which is directed toward restoring the equipment or system to its normal functioning capability, without altering its basic design function. Routine maintenance is conducted during normal plant operation, as well as during scheduled shutdowns.
Normal routine maintenance work is either requested by the Production Department or results from the preventive maintenance program. Because there is generally a backlog of work, all work is given a level of priority to facilitate effective scheduling. Priority is based on safeguards considerations, production loss resulting from the equipment being shut down, or the probability of a breakdown if a repair is not made, with consequent damage to equipment.

Non Routine Maintenance

9.79 Non routine maintenance includes modifications or additions to systems or processes as differentiated from repair or replacement of faulty equipment. Depending upon the nature and extent of the work, maintenance or construction forces are used. In the latter case, Plant Engineering is responsible for maintaining close contact with the work to see that it is performed in accordance with specifications, within the cost estimate, and reporting on the progress of the job during the construction period.

Administrative Procedures for Carrying Out Program

9.80 All work performed in the various categories of the maintenance program, including those of the Plant Engineering Section both during normal plant operation and during plant shutdown, are in accordance with established administrative procedures described below. These administrative procedures deal with the conditions or requirements that must be satisfied to initiate and complete a maintenance operation rather than to exercise control over the actual repair work.

Non Vital Components

9.81 Administratively controlled maintenance procedures are not required on non vital components for safe operation of the facility. Therefore, preventive maintenance or routine maintenance operations on non vital components is carried out by the maintenance sections in accordance

with normal standard maintenance practice, except as noted in Section 9.83. The maintenance work on non vital components is coordinated with the Production Department to minimize downtime. Detailed maintenance procedures for most pieces of equipment are provided by the vendor or are written by maintenance personnel; for hazardous conditions the operation may be altered and is administratively controlled as described in Section 9.83. Non routine maintenance of a non vital component is discussed in Section 9.84.

Vital Components

9.82 Administratively controlled maintenance procedures are required on vital components for safe operation of the facility. Therefore, prior to performing preventive maintenance or routine maintenance, it is necessary to evaluate the effect of performing the maintenance work. Such an evaluation is made on all items listed as vital equipment. The Vital Equipment List is prepared by the Production Department and the Technical Services Department and approved by the Plant Manager. If the maintenance work does not involve a radiation or contamination hazard, the work is initiated after approval by the Production Manager. If a radiation or contamination hazard is associated with the maintenance job, it is necessary to alter the operation as described in Section 9.83. Non routine maintenance of vital components is discussed in Section 9.84.

Hazardous Maintenance

9.83 When hazardous conditions exist, it is necessary to alter normal maintenance procedures before maintenance is initiated. In all cases, a Special Work Procedure is required. This work procedure is obtained and is administered as described in Paragraph 9.17. The use of this permit provides maximum assurance that both the worker and management take adequate steps to minimize the consequences of radiation or contamination associated with the job.
In all cases involving hazardous maintenance, it is necessary to fulfill the requirements set forth in the Special Work Procedure. After this is done, the maintenance operation is performed in accordance with Sections 9.81 and 9.82.

Non routine Maintenance

9.84 Non routine maintenance involves changes in basic design or additions to equipment. When it is necessary to perform this type of maintenance, on either vital or non vital components, such maintenance is not carried out until a complete evaluation of such a change is conducted and approved by the Criticality Committee. After the procedure is approved, the maintenance operations are performed in accordance with Sections 9.81 9.82 or 9.83.

Work Completion

9.85 Representatives of Plant Engineering, Production and Technical Services (if involved) and Health and Safety Departments (if involved) observe the testing and return to operation of the components or system involved in maintenance.

Production Department

9.86 The Production Department is responsible for the operation and maintenance of the processing plant and its related process services. The organization and administration of the department has been planned to provide safety to the public and plant personnel and to effect operation and maintenance of the facility within the operating license limitations. In order to effectively operate the plant within the prescribed limitations, the Production Department has been broken down into groups to achieve effective control of the necessary operations. The group breakdowns are as follows:

a. Fuel and Mechanical Handling;
b. Chemical Processing;
c. Plant Engineering;
d. Utilities and Process Services.

9.87 The Fuel and Mechanical Handling group is responsible for the Fuel Receiving and Storage area including cask transport, handling fuel assemblies, transfer and storage; operation of the FRS water treatment facilities; Process Mechanical Cell operation including fuel assembly transfer, handling, disassembly by saw or mechanical means, fuel shearing, handling of scrap, utility services to the area and hot equipment repair or replacement; General Purpose Cell including the loading, handling, storage and transfer operations of fuel baskets, scrap material and equipment utility services; Chemical Process Cell-Equipment Decontamination Room including the charge of fuel into and discharge of leached hulls out of the dissolvers, replacement of equipment, and remote handling operations within the CPC and the EDR; Scrap removal including the handling and transfer operations of waste and materials into and out of the mechanical head end facilities. Accountability and material control coordination consistent with Production Department requirements.

9.88 Chemical Processing group responsibilities include feed dissolution, solvent extraction, solvent recovery systems, product purification and concentration, acid recovery, sampling, cold chemical make up, waste concentration and rework operations, process off gas systems, building ventilation and accountability in these areas consistent with Production Department requirements.

9.89 The Plant Engineering group is responsible for the maintenance of the facility as necessary to maintain continuity of operation as described in detail in Paragraphs 9.68 through 9.85.

9.90 The Utilities and Process Services group includes the operations of: all utility systems within the utility room, plant area and off-plant facilities; non radioactive systems for both solids and liquids; operation of the conventional low level burial and scrap removal from the plant; material handling including the transport, handling, warehouse and distribution of equipment and supplies as required for plant operations; decontamination of areas and facilities not included under other groups; material control including records of input, output and inprocess material necessary to effect control; and accountability of source material and

special nuclear material as necessary for Production Department requirements.

9.91 The basic plant operation and control is carried out physically by the process operators and shift supervisors; however, in a large processing complex such as the NFS plant, additional support including technical and analytical services, monitoring, accountability, maintenance and control is necessary to assure proper operations. The groups, listed in Paragraph 9.86 and staffed by production supervisors, have been established within the Production Department to provide the defined portions of this support. Their primary function is to maintain an up-to-date intimate knowledge of their respective areas of responsibility. These staff functions have functional responsibility for their areas, however, administrative control is maintained by the Production Manager or an Assistant Production Manager. This type of organization provides a decentralized type of functional responsibility, yet maintains centralized control over operations.

9.92 To the maximum practical extent all details of plant operation are controlled by written procedures. These include Standard Operating Procedures, Run Sheets (including administrative controls) and Letters of Authorization. These procedures are maintained in a current status as described in Paragraphs 9.5 through 9.7 and 9.94.

9.93 The Standard Operating Procedures include a detailed step-by-step procedure for functional operation of each piece of equipment and/or process function in the plant. The format for SOP together with a general listing of the major systems covered by SOP are shown in Appendix 9.93. Included in each SOP is the scope encompassed, a general description of the operation involved, cautions to be observed in operations, administrative controls required during the operation, references to related SOP or other procedures, detailed instructions for functional operation of the equipment and, insofar as possible, the mechanical limitations of the equipment. This last item may, in some instances, more appropriately be included in Run Sheets.

9.94 Run Sheets are another set of procedures used to maintain control of the plant operation. They list the operating conditions for the campaign of a particular fuel beginning with mechanical processing and continuing through the process to product storage. They include the upper and lower limits for each flow of plant processing. For example, maximum and minimum flow rates are listed for each influent stream to each solvent extraction column as well as a desired operating flow. Separate Run Sheets are used for each flowsheet authorized under the operating license. The published Run Sheets available to the shift supervisor and his operators are generally more restrictive than those permissible under the operating license. This practice allows more strict enforcement and control of the plant operation. The shift supervisor cannot operate outside the specified limits of the Run Sheet. However, extension of these limits may be made, within the limits of the operating license, by an approved Letter of Authorization. If the supervisor cannot maintain the operation within the limit specified by the Run Sheet the affected portion of the operation must be shut down until the condition is corrected or approval to modify the run sheet is received. Run Sheets are reviewed periodically and amended as deemed necessary. Under no conditions is the plant operated outside the technical specifications included in the operating license.

9.95 Letters of Authorization are an administrative procedure directing actual plant operation as described in Paragraph 9.7. They are used to authorize a specific Run Sheet and/or auxiliary procedures for a particular processing campaign and in addition, are used to modify any of the restrictive procedures established for plant control. All Letters of Authorization are approved as discussed in Paragraph 9.7.

9.96 The actual operation of the complete processing plant is performed by personnel licensed as described in Paragraph 9.9 through 9.11. The basic areas of operator responsibility are broken down into specific catagories or areas of the plant consistent with production plant operating techniques. The specific areas are manned by operators consistent with their group license. The specifically-assigned areas for each shift are as follows:

1. Central Control Room;
2. Process Mechanical Cell;
3. General Purpose Cell;
4. Fuel Receiving and Storage-Chemical Processing Cell;
5. Sampling;
6. Chemical Makeup;
7. Product Packaging and Handling;
8. Waste Handling.

In addition, non licensed personnel are assigned to the following areas:

1. Utility Room;
2. Yards and ground, etc.

A brief description of each of these areas outlining the basic operator responsibilities for the respective areas is as follows:

1. <u>Central Control Room</u>

The Chemical processing portion of the plant is controlled from a Central Control Room located on the fourth floor of the process building. Processing beginning with dissolver operations and continuing through feed adjustment, solvent extraction, product purification, concentration and storage are operated from this location. Complete control of the process is exercised from the control room with the exception of non routine operations such as manual block valves for the process service requirements which are located in the Upper and Lower Extraction Aisles. Manual valving in the Upper and Lower Extraction Aisles is performed by other individuals at the request of the control room operators or shift supervisors. The control room panel is a semi-graphic type for ease of identification and efficient operation. In addition to posting the Run Sheets in the control room, many of the instruments are individually posted showing the limits of operation.

2. <u>Process Mechanical Cell</u>

a. Fuel assembly transfer and handling.
b. Fuel assembly disassembly using saw or mechanical means.
c. Removal of extraneous hardware.
d. Make up of fuel modules and shearing.
e. Handling individual fuel elements.
f. Scrap handling, cell decontamination and in-cell remote maintenance.

The run sheets for the PMC are somewhat different than those for the rest of the processing complex. They are made up of detailed fuel handling procedures which, in effect, are similar to an SOP. Each different category of fuel requires specific instructions for handling throughout the PMC. The fuel handling procedures indicate the adapters and fixtures required for handling different fuels within the cell, detailed instructions for handling the fuel on the saw table, the disassembly, inspection and push out table, shear feed magazine, sequence of shearing and special precautions to be taken. Included also are throughput quantities so that they can be coordinated with chemical processing.

3. General Purpose Cell

a. Chopped fuel loading, handling, storage and transfer operations.
b. Leached hull sampling, handling and transfer operations.
c. Receiving and transferring scrap and other materials to the PMC.
d. Fuel basket handling including liners, capping and material control.
e. In-cell remote maintenance

4. Fuel Receiving and Storage

a. Accountability as applicable to Production Department responsibilities.
b. Cask receiving, unloading and transport.
c. Fuel assembly handling, transfer and storage.
d. Coordination with mechanical head end operation.
e. Operation of pool water treatment systems.

5. Chemical Process Cell--Mechanical Handling

a. Charging dissolver with fuel.
b. Dissolver discharge of leached hulls.
c. Equipment replacement by remote mechanical means.
d. Other remote operations within the CPC requiring use of the remote mechanical handling facilities.

6. Process Sampling

Sampling of the various process streams and vessels is conducted for accountability, process control, and waste loss determination throughout the chemical processing portion of the plant. The samples are taken at times prescribed by the Run Sheet; auxiliary samples may be taken as determined by the shift supervisor. Laboratory analytical data from the samples are transmitted to the accountability officer and also to the control room where the results are logged. The shift supervisor then makes process adjustments or transfers within the limits prescribed by the Run Sheets.

7. Chemical Makeup Areas

The chemical makeup area includes process solution makeup for the chemical processing portion of the plant. These include all cold chemical influent streams for the solvent extraction columns and other cold process solutions such as dissolution and regeneration solutions. Each process solution is made up from

a prescribed detailed form listing the constituent concentrations and total amounts of each solution. The solution is then sampled and held for certification. Following certification, and upon process demand, the solution is then transferred to a run tank for subsequent introduction to the process or, in some cases, directly into the process vessels.

8. Product Packaging and Handling

a. Load out of plutonium product into bird cages and interim storage in the process building.
b. Load out of high enriched uranium into bird cages and interim storage.
c. Load out of low enriched uranium product to transport vessel.

All operations are conducted on a batch basis following specific instructions by the shift supervisor.

9. Supporting Areas

a. Scrap Removal Areas, including the receiving and transfer to the burial area of leached hulls and other head-end scrap generated during processing, and transfer of new materials to the General Purpose Cell for head-end processing.
b. Equipment Decontamination Room, including the mechanical handling to and from the chemical process cell.
c. Process Laundry for decontamination of the anti-contamination clothing used in the facility.

10. Utility Room Operation--All Plant Services Contained Within the Utility Room Complex

a. Water--raw, filtered, process, demineralized, and potable.
b. Air--process, instrumentation.
c. Steam--equipment, process and heating.
d. Electrical--normal and emergency.

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APPENDICES

APPENDICES I

APPENDICES II

Appendix 2.36

Public Water Supply Data

Western New York Nuclear Service Center

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

ALLEGANY COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
M-6	A-1	0215	Cuba (V)	Wells & Springs (Open Storage)	Aeration Iron Removal Calgon & Chlorination Calgon & Chlorination Chlorination	Cuba (V)	1,949
L-7	A-2	0212	Houghton College	Drilled Well & Spring (Closed Storage)	Iron Removal	Houghton College & Houghton (H)	500
L-7	A-3	0220	Fillmore (V)	Springs	None	Fillmore (V)	522
L-7	A-4	0208	Belfast Water District	Wells (Open Storage)	None	Belfast (T)	1,265

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

CATTARAUGUS COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
L-4	CA-1 (also E-2)	0526 0527 0547	Gowanda (V)	Peter Point Brook, Springs, Well & Reservoir (Open Storage)	Coagulation, Rapid Sand Filters, Activated Carbon, Chlorination	Gowanda (V)	2,273
L-4	CA-2	0506	Gowanda Central School - Dayton Branch	Well	None	Gowanda Central School - Dayton Branch	
L-3	CA-3	0550	Gowanda Central School - Perrysburg Branch	Well	None	Gowanda Central School - Perrysburg Branch	
L-4	CA-4	0523 0542	Cattaraugus (V)	Springs & Wells (Open Storage)	Chlorination	Cattaraugus (V)	1,258
L-4	CA-5	0521 0523	Otto Water District	Springs (Open Storage)	None	Otto (T)	715
L-4 M-4	CA-6	0518 0522	Little Valley (V)	Wells & Springs (Open Storage)	Chlorination	Little Valley (V)	1,244
M-4	CA-6a	0518	County Building, Little Valley (V)	Spring	Chlorination	County Building, Little Valley (V)	
M-4	CA-7	0529	East Randolph (V)	Well (Open Storage)	None	East Randolph (V)	594

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

CATTARAUGUS COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
M-3	CA-8	0529	Randolph (V)	Wells (Open Storage)	None	Randolph (V)	1,414
L-3	CA-9	0506	South Dayton (V)	Well (Closed Storage)	Chlorination	South Dayton (V)	696
L-3	CA-9a	0506	Curtice Bros. Canning Co.	Wells	Chlorination	Curtice Bros. Canning Co.	
L-3	CA-9b	0506	Carnation Milk Co.	Wells	None	Carnation Milk Co.	
L-3	CA-10	0526	J. N. Adam State School	Wells (Open Storage)	Chlorination	J. N. Adam State School	
L-5	CA-11	0509 0545	Ellicottville (V)	Wells & Spring (Closed Storage)	Chlorination	Ellicottville (V)	1,150
L-5	CA-12	0502	Crystal Water Co.	Springs (Closed Storage)	None	West Valley (H)	600
L-5	CA-12a	0502	West Valley Central School	Drilled Well	None	West Valley Central School	
L-5 L-6	CA-13	0520	Pierce Water Co.	Springs (Closed Storage)	None	Machias (H)	650
L-5 L-6	CA-13a	0520	Cattaraugus Co. Home	Springs & Well (Closed Storage)	None	Cattaraugus Co. Home Machias	

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

CATTARAUGUS COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
L-5 L-6	CA-14	0533	Delevan Village	Springs (Open Storage)	Chlorination	Delevan (V)	777
L-5	CA-15	0502	E. D. Ford - Meat Processing	Well	None	Abattoir	
L-6	CA-16	0511	Franklinville Village	Wells (Closed Storage)	Chlorination	Franklinville (V)	2,124
M-5	CA-17	0531	Salamanca City	Wells, Spring & Stream (Open & Closed Storage)	Chlorination	Salamanca (C)	8,480
M-5	CA-18	0503	Limestone Village	Wells	None	Limestone (V)	539
M-5	CA-19	0513	N. Y. S. Forestry Camp #2	Wells	None	Forestry Camp	
M-5	CA-20	0530	Allegany State Park	Well	None	Administration Building - State Park	
M-6	CA-21	0501	Allegany Village	Wells (Closed Storage)	None	Allegany (V) Fairfax District Allegany District #2 St. Bonaventure Univ.	2,064 40 200
M-6	CA-22	0524	Olean City	Stream & Wells (Closed Storage)	Filtration, Chlorination, Fluoridation	Olean (C)	21,868

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

CATTARAUGUS COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-5	CA-23	0533	Camp Duffield Presbyterian Church	Well	None	Camp Duffield	
M-6	CA-24	0528	Portville Village	Wells (Closed Storage)	Chlorination, Corrosion Control	Portville (V)	1,336

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

CHAUTAUGUA COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-3	Ch-1	0714	Silver Creek (V) (2 Surface Reservoirs)	Silver Creek	Chlorination	Silver Creek (V)	3,310
						Hanford Bay (H)	
						Sunset Bay (H)	300
						Irving (H)	
L-3	Ch-2*	0714	Forestville (V)	Springs (Open Storage)	Chlorination	Forestville (V)	905
L-3	Ch-2a*	0701	Forestville (V)	Springs	Chlorination	Forestville (V)	905
L-3	Ch-3*	0706	Cherry Creek (V)	Springs (Open Storage)	None	Cherry Creek (V)	649
L-3	Ch-3a*	0741	Cherry Creek (V)	Well	None	Cherry Creek (V)	649
M-3	Ch-4*	0710	Jamestown (C)	Wells (Closed Storage)	Chlorination	Jamestown (C)	41,818
						Falconer (V)	3,343
						Celoron (V)	1,507
						West Ellicott (H)	1,782

* Supplies outside 25 mile limit.

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

ERIE COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-5	E-1	1555	Springville (V)	Wells	Aeration	Springville (V)	3,852
L-4	E-2 (see CA-1)	1509	Gowanda (V)	Springs	Coagulation, Rapid Sand Filtration, Chlorination, Activated Carbon	Gowanda (V) Collins (T)	1,079 6,984
L-4	E-4	1509	Gowanda State Hospital - Collins(T)	Surface Reservoir	Coagulation, Rapid Sand Filtration, Chlorination	Gowanda State Hospital	
K-3	E-6	1505	Farnham (V)	Wells (Closed Storage)	Chlorination	Farnham (V)	422
K-4	E-7	1505 1520	North Collins (T)	Wells (Closed Storage)	Chlorination	North Collins (V)	1,574
K-4	E-8	1520	Lawtons Water Co. - North Collins (T)	Springs	None	North Collins (T)	3,805

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

ERIE COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-6	E-9	1522	Chaffee Water Works Company - Sardinia(T)	Wells	None	Chaffee (V) Sardinia (T)	250 2,145
K-5	E-10	1516	Holland Town Board - Holland (T)	Wells (Closed Storage)	Chlorination	Holland (T)	2,304
K-4	E-11	Indian Reservation	Gowanda State Hospital (Iroquois Annex)	Well (Closed Storage)	Chlorination	Gowanda State Hospital (Iroquois Annex)	
K-3	E-12	1513	Angola (V)	Lake Erie (Closed Storage)	Coagulation, Rapid Sand Filter, & Chlorination	Evans (T) Angola (V) Evangola State Park	2,499
J-5	E-13	1546	East Aurora (V)	Wells (Closed Storage)	Pressure Sand Filters, Softening, Fluoridation	East Aurora (V)	6,791
K-4 K-5	E-14	1521	Chestnut Ridge Park Orchard Park (T)	Wells	None	Chestnut Ridge Park	120
K-5	E-15	1503	Emery Park - Aurora (T)	Wells	None	Emery Park	
K-5	E-16	1508	Orchard Park (V)	Reservoir in Town of Colden	Coagulation, Rapid Sand Filter, & Chlorination	Orchard Park (T) Aurora (T) Orchard Park (V)	2,236 3,278

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

ERIE COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-4	E-17	1515	Hamburg (V)	Eighteen Mile Creek (North Branch) & Wells	Coagulation, Rapid Sand Filter, & Chlorination	Hamburg (V) Orchard Park (T)	9,145
K-4	E-18	1515 1549	Wanakah - Hamburg(T)	Lake Erie (Closed Storage)	Coagulation, Rapid Sand Filter, & Chlorination	Hamburg (T)	250
J-4 J-5 K-3 K-4	E-19*	1591 1515	Erie County Water Authority (E.C.W.A.)	Lake Erie	Coagulation, Rapid Sand Filter, & Chlorination	Evangola State Park Farnham (V) Evans (T) Angola (V)	241,000
			Woodlawn Plant - Hamburg (T)	Lake Erie	Coagulation, Rapid Sand Filter, & Chlorination	Orchard Park (T) Hamburg (T) Hamburg (V)	
			Sturgeon Point - Plant Evans (T)	Lake Erie (Open & Closed Storage)		Lackawanna (C) Blasdell (V) West Seneca (T) Lancaster (T) Alden (T) Amherst (T) Clarence (T)	

* Sources outside of 25 mile limit.

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

ERIE COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
J-4	E-20*	1581	Buffalo (C)	Lake Erie (Closed Storage)	Coagulation, Rapid Sand Filter, Chlorination, Fluoridation	Buffalo (C) Lackawanna (C) West Seneca (T) Cheektowaga (T) Amherst (T)	532,759
J-4	E-21*	1523	Tonawanda (T)	Niagara River (Closed Storage)	Coagulation, Rapid Sand Filter, Chlorination, Fluoridation	Tonawanda (T)	105,032
J-4	E-22*	1514	Grand Island (T)	Niagara River (Closed Storage)	Coagulation, Rapid Sand Filter, & Chlorination	Grand Island (T)	9,607

* Sources outside of 25 mile limit.

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

WYOMING COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-7	W-1	6113	Pike (V)	Springs (Open Storage)	None	Pike (V)	345
K-7	W-1a	6106	Pike (V)	Springs (Open Storage)	None	Pike (V)	345
K-6	W-2	6109	Java Village (H)	Springs (Open Storage)	None	Java Village (H)	
K-6	W-3	6109	North Java Water District	Springs & Wells (Closed Storage)	None	North Java(H)	300
K-6	W-4	6141	Arcade (V)	Well (Closed Storage)	Chlorination	Arcade (V) Yorkshire (H) Sandusky (H)	1,930 250 400
K-6	W-4a	0512	Arcade (V)	Springs	Chlorination	Arcade (V) Yorkshire (H) Sandusky (H)	1,930 250 400
K-6	W-5	6106	Bliss Water Company	Springs (Covered Storage)	Chlorination	Bliss (H)	375
J-6	W-6*	6114	Varysburg Water District	Springs & Well (Covered Storage)	None	Varysburg (H)	225

* Supplies outside 25 mile limit.

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

WYOMING COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
J-6	W-7*	6103	Akron (V) (Erie County)	Open Reservoir	Coagulation, Rapid Sand Filters, Corrosion Control, & Chlorination	Akron (V)	2,841
J-6	W-8*	6102	Attica (V)	River (Closed Storage)	Coagulation, Rapid Sand Filters, Activated Carbon, Corrosion Control, & Chlorination	Attica (V) Attica State Prison Alexander (H)	2,758 365
K-7	W-9*	6107	Warsaw (V)	River (Open Storage)	Coagulation, Rapid Sand Filters, Corrosion Control, & Chlorination	Warsaw (V) Rock Glen (H) Newburg (H) South Warsaw (H) Martinville (H) Saltville Water District	3,653
K-7	W-9a*	6107	Warsaw (V)	River (Open Storage)	Chlorination	Warsaw (V) Martinville (H) Saltville Water District	

* Supplies outside 25 mile limit.

PUBLIC WATER SUPPLY DATA
WESTERN NEW YORK NUCLEAR SERVICE CENTER

WYOMING COUNTY

Map No.	Waterworks Map No.	Co. Town Code No.	Waterworks	Source	Treatment	Community Served	Population
K-7	W-10*	6104	Perry (V)	Silver Lake (Open Storage)	Coagulation, Rapid Sand Filters, Activated Carbon, Chlorination, Fluoridation	Perry (V) Perry Center Water District North End of Letchworth Park Silver Lake Assembly	4,629
K-7	W-11*	6104	Batavia YMCA	Silver Lake (Open Storage)	Chlorination	Camp	
K-7	W-12*	6104	Mt. Morris (V) - Livingston County	Silver Lake	Coagulation, Rapid Sand Filters, Activated Carbon, & Chlorination	Mt. Morris (V) - Livingston County	3,250
K-7	W-13*	6107	Silver Springs (V)	Springs & Wells (Open Storage)	Chlorination	Silver Springs (V)	726
K-7	W-14*	6107	Castile (V)	Springs & Wells (Open Storage)	Chlorination	Castile (V)	1,146
K-7	W-14a*	6108	Castile (V)	Springs (Open Storage)	Chlorination	Castile (V)	

* Supplies outside 25 mile limit.

APPENDICES III

APPENDICES IV

Appendix 4.1

Material Balance Flowsheets

Drawing 4139 15R-A-31	Commonwealth Edison Fuel
Drawing 4139 15R-A-32	Yankee Atomic Fuel
Drawing 4139 15R-A-33	Consolidated Edison Fuel
~~Drawing 4139 15R-A-34~~	~~Power Reactor Development Corp. - Blanket~~ rescinded
Drawing 4139 15R-A-35	Power Reactor Development Corp. - Core
Drawing 4139 15R-A-36	Zr-U Alloy Fuel
~~Drawing 4139 15R-A-37~~	~~UO_2-Stainless Steel Cermet Fuel~~ rescinded
Drawing 4139 15R-A-38	Northern States Power Company
~~Drawing 4139 15R-A-39~~	~~Consumers Public Power - Hallam~~ rescinded

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APPENDIX "4.2"

CASK ACCEPTANCE CRITERIA

1. Except as otherwise agreed, shipping containers to be used by the Commission shall be: (a) SCRUP container (Bureau of Explosives No. 622); (b) Westinghouse containers (Bureau of Explosives No. 1475); (c) GE spent fuel shipping containers (Bureau of Explosives No. 1472); (d) Stanray S-1 container (Bureau of Explosives No. 1400). It is agreed that the containers specified in this Section 1 meet all of the requirements of this Appendix "B" which are related to the design of the containers.

2. Three complete sets of design or "as built" drawings for each container to be used shall be submitted to the Contractor for approval with the following information:

   a. Details on fuel loading arrangement in container.
   b. Special tools or equipment required to unload fuel from container and open containers.
   c. Method of attachment of container to the shipping vehicle proposed for use in shipment;

3. The Contractor shall accommodate only the type of containers specified in Section 1 of this Appendix "B". If an alternate container design is required by the commission at a later date, it shall be the responsibility of the Commission to design such container so that it can be handled by the Contractor or to reimburse the Contractor for changes to the fuel receiving station to accommodate the alternate container design.

4. Shipment will be accepted from rail or motor carriers.

5. All containers must be approved by the Bureau of Explosives for shipment of specific materials involved.

6. All containers must meet the Interstate Commerce Commission's allowable radiation requirements.

7. The maximum radiation limits shall be not more than 200 mr/hr at the surface of the container.

8. Containers must be readily removable from the shipping vehicle. Maximum weight of a loaded container to be lifted from the vehicle is 100 tons. The Commission shall provide a set of container unloading instructions for each container which give reasonable assurance that under normal circumstances the container can be unloaded with a total exposure to personnel of 25 mrem.

9. Containers must have provision for free drainage of water from all external surfaces and attachments.

10. The use of cooling media other than water is not acceptable when such media can contaminate the storage water or fuel elements or cause other problems. Substitution of other cooling media for water or the use of additives such as antifreeze will be considered on an individual case basis.

11. The container must have a readily operable drain and all internal surfaces should drain freely to it. A siphon drain is preferred since a bottom drain is more susceptible to damage and can result in the loss of the cooling water. A siphon drain must be in the body of the container and not part of the cover.

12. Container design should recognize the possible use of impact wrenches in the removal of nuts, and bolts. Proper consideration should, therefore, be given to the clearances necessary.

13. Containers must be top opening.

14. All inner container surfaces which are likely to be contacted by the coolant must be of stainless steel. All external surfaces must be accessible for decontamination. These external surfaces including weldments, must be smooth, free of weld spatter, and free of non-draining crevices or pockets.

15. Units with exposed uranium or plutonium shall be canned by the Commission in containers whose material and design shall have been approved by the Contractor, unless the Contractor determines such canning is unnecessary.

16. Container design should permit withdrawal of a representative sample of the container coolant while the container is on the transport vehicle.

17. The activity limit for container cooling water is 10^{-5} beta-gamma curie/cc or 10^{-7} alpha curie/cc.

18. The internal pressure of the primary coolant system of a container shall not exceed 50 pounds per square inch gauge or 50% of the design pressure, whichever is less, under normal conditions of transport. In containers designed to operate at a pressure no greater than atmospheric, an automatic pressure release system must be incorporated. Methods of control or containment of the released fluid must be provided.

19. Surface contamination shall not exceed 4000 beta-gamma disintegration per minute per 100 square centimeters nor 500 alpha disintegrations per minute per 100 square centimeters. Containers contaminated to no more than 5 times these levels will be decontaminated without charge as part of the chemical processing service provided by the Contractor under Article II of this contract.

20. Containers must not leak grossly.

APPENDICES V

APPENDICES VI

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APPENDICES VII

Appendix 7.7

Atmospheric Dispersion Calculations

Appendix 7.7

Atmospheric Dispersion Calculations*

For short-term centerline concentrations:

$$\frac{X}{Q} = \frac{1}{\pi \sigma_y \sigma_z u} \exp\left[-\frac{h^2}{2\sigma_z^2}\right] \tag{7.7a}$$

For long-period average concentrations:

$$\frac{X}{Q} = \frac{2^{\frac{1}{2}}\, 0.01f}{\pi^{\frac{1}{2}} \sigma_z u x} \exp\left[-\frac{h^2}{2\sigma_z^2}\right] \tag{7.7b}$$

Where X = concentration in c/m^3

Q = release rate in c/sec

u = mean wind speed in meters/sec

h = source (stack) height in meters

σ_z, σ_y = dispersion coefficients in m

x = distance from source

f = wind frequency in per cent per radian

For inversion conditions: $h = 65m$, $u = 1m/sec$

$$\frac{X}{Q} = \frac{1}{\pi \sigma_y \sigma_z \times 1} \exp\left[-\frac{65^2}{2\sigma_z^2}\right] = \frac{1}{\pi \sigma_y \sigma_z} \exp\left[-\frac{2120}{\sigma_z^2}\right]$$

For average conditions:

$h = 65m$

$u = 4m/sec$

f (per cent per octant) $= \frac{2\pi}{8} f$ (per cent per radian) = 25 per cent

$$\frac{X}{Q} = \frac{2^{\frac{1}{2}}\, 0.01f}{\pi^{\frac{1}{2}} \sigma_z (4x)} \times \frac{2\pi}{8} \exp\left[-\frac{65^2}{2\sigma_z^2}\right] = \frac{3.92 \times 10^{-2}}{\sigma_z x} \exp\left[-\frac{2120}{\sigma_z^2}\right]$$

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* Nuclear Safety, Volume 2, No. 2, December 1960, pg. 56.

Appendix 7.7, continued

Summary of Calculations

Condition	Distance	u m/sec	σ_y	σ_z	X/Q
Slightly Unstable (Average)	1,500	4	-	90	2.24×10^{-7}
Slightly Unstable	7,200	4	-	350	1.32×10^{-8}
Slightly Unstable	51,000	4	-	1500	5.13×10^{-10}
Moderately Stable (Inversion)	1,500	1	50	18	4.95×10^{-7}
Moderately Stable	2,000	1	65	22	2.88×10^{-6}
Moderately Stable	4,000	1	130	32	9.8×10^{-6}
Moderately Stable	5,000	1	170	37	1.08×10^{-5}
Moderately Stable	6,000	1	180	39	1.13×10^{-5}
Moderately Stable	7,200	1	210	41	1.05×10^{-5}
Moderately Stable	8,000	1	230	42	9.9×10^{-6}
Moderately Stable	10,000	1	280	47	9.27×10^{-6}
Moderately Stable	51,000	1	1200	80	2.4×10^{-6}

Appendix 7.8

Iodine Deposition

Appendix 7.8

Iodine Deposition

$W = Xv_g$

where X = concentration in c/m^3

v_g = deposition velocity in m/sec

W = deposition rate $c/(m^2)(sec)$

$$I = \int W dt = \frac{Xv_g}{\lambda}(1 - e^{-\lambda t})$$

where I = ground concentration in c/m^2

λ = decay constant in sec^{-1}

t = time of deposition in sec

Assuming equilibrium conditions, $t = \infty$

$$I = \frac{Xv_g}{\lambda} \text{ for } I=131 \qquad I = \frac{Xv_g}{10^{-6}} = 10^6\, Xv_g$$

Using v_g = 0.01 m/sec*, average X values computed by the method shown in Appendix 7.7 and converting to $\mu\mu c/m^2$:

$$I = 10^{16}\, X\, \mu\mu c/m^2$$

Distance Meters	X c/m^3 or $\mu c/cc$	I $\mu\mu c/m^2$
1,500	2.2×10^{-14}	22
2,000	1.5×10^{-14}	15
5,000	3.1×10^{-15}	3.1
10,000	0.89×10^{-15}	0.89
20,000	2.6×10^{-16}	0.26

* A. C. Chamberlin, Quarterly Journal of the Meteorological Society, Volume 85, No. 336, pg. 358.

Appendix 7.32

Calculation of Iodine Dose to the Thyroid Resulting from 10^{20} Fissions

Appendix 7.32

Calculation of Iodine Dose to the Thyroid Resulting from 10^{20} Fissions

NRDL-456, "Calculated Activities and Abundances of U-235 Fission Products", provides complete tabulations of activities of fissions resulting from 104 fissions. Two sets of data are provided; Glendenin's data has been used for the calculations herein. The data from NRDL-456 was multiplied by:

$$10^{16}/3.7 \times 10^{10} = 2.7 \times 10^{5}$$

to give activity in curies resulting from 10^{20} fissions. The peak activity was used in all cases.* All iodine activity was assumed to be released instantaneously.

The atmospheric dilution factors presented in Appendix 7.7 were used for inversion conditions. The same method was used for calculation of dilution factors under average meteorological conditions except that dispersion coefficients corresponding to slightly unstable conditions were used. The following is a summary of these additional calculations:

$$X/Q = \frac{1}{\pi \sigma_z \sigma_y u} \exp\left[-\frac{h^2}{2\sigma_z^2}\right] = \frac{1}{4\pi \sigma_z \sigma_y} \exp\left[-\frac{65^2}{2\sigma_z^2}\right]$$

$$= \frac{7.96 \times 10^{-2}}{\sigma_z \sigma_y} \exp\left[\frac{2120}{\sigma_z^2}\right]$$

where u = 4 m/sec and h = 65m.

Distance Meters	σ_y Meters	σ_z Meters	X/Q sec/m^3
1500	150	90	4.55×10^{-6}
7200	650	350	3.44×10^{-7}
51000	3500	1500	1.51×10^{-8}

Using activities and dilution factors calculated as described above, the inhalation dose was computed using the following:

$$\text{Dose} = X/Q \times Q_o \times BR \times D$$

Where BR = breathing rate = 3.47×10^{-4} m^3/sec

D = dose in rem per curie inhaled

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* In one case - the I-134 dose at Buffalo under inversion conditions - the decay time was sufficient that the dose could be neglected when compared with the remaining isotopes.

Appendix 7.32, continued

The values for BR and D were taken from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactors". The following is a sample calculation for Springville (7200 meters) under inversion conditions:

Isotope	Q_o Curies	X/Q sec/m^3	BR m^3/sec	D rem/c	Dose rem
I-131	73	1.05×10^{-5}	3.47×10^{-4}	1.48×10^{6}	0.395
I-132	240	1.05×10^{-5}	3.47×10^{-4}	5.35×10^{4}	0.05
I-133	1,250	1.05×10^{-5}	3.47×10^{-4}	4.0×10^{5}	1.83
I-134	17,300	1.05×10^{-5}	3.47×10^{-4}	2.5×10^{4}	1.58
I-135	4,400	1.05×10^{-5}	3.47×10^{-4}	1.24×10^{5}	2.0
				Total (rem/person)	5.85

Appendix 7.34

Criticality Incident in Fuel Pool

Appendix 7.34

Criticality Incident in Fuel Pool

$$10 \text{ mwt} \times 3 \text{ hrs} \times \frac{1 \text{ day}}{24 \text{ hours}} = 1.25 \text{ mwd}$$

$$= 1.25 \text{ mwd} \times 8.2 \times 10^7 \frac{\text{Btu}}{\text{mwd}}$$

$$= 1 \times 10^8 \text{ Btu}$$

$$= 1.25 \text{ mwd} \times 2.7 \times 10^{21} \text{ fissions/mwd}$$

$$= 3.4 \times 10^{21} \text{ fissions.}$$

The fuel pool (see Paragraph 3.8) is 91' x 40' x 28'.

$$= 10^5 \text{ cu ft water}$$

$$= 7.5 \times 10^5 \text{ gallons water}$$

$$= 6.4 \times 10^6 \text{ pounds water}$$

Enough heat would be released to heat the pool

$$\frac{1 \times 10^8}{6.4 \times 10^6} = 16F$$

EBWR defect studies showed an increase in release rate of Kr-88 of 0.07 c/day when a defected pin containing 2.7 grams U-235 operating at 20 mwt was inserted. The total inventory of U-235 in EBWR is about 75 kg.

Kr-88 has a yield of 3.7% and a 2.77 hour half-life. The total inventory of Kr-88 in the whole reactor is

$$\text{Curies} = 8 \times 10^5 \text{ R y (at equilibrium)}$$

$$R = \text{grams U-235 fissioned/day} = 20 \times 1.052 = 21$$

$$y = \text{fission yield g/g}$$

$$\text{Curies Kr-88} = 8 \times 10^5 \times 21 \times 0.037 = 6.2 \times 10^5$$

The curies of Kr-88 in the defected EBWR pin were

$$\frac{2.7}{75 \times 10^3} \times 6.2 \times 10^5 = 22 \text{ curies}$$

Of this 0.07 was lost/day or $\frac{0.07}{22} = 3.2 \times 10^{-3}$ fraction/day $= \frac{3.2 \times 10^{-3}}{86{,}400}$ $= 3.7 \times 10^{-8}$ fraction/sec.

Inventory of gaseous activity in the five fuel elements. Taken as 3.4×10^{21} fissions times yield with an allowance for inventory of the long-lived components Kr-85 and I-131.

	y, %	$t_{\frac{1}{2}}$	sec^{-1}	Atoms	Curies
Kr-85m	1.5	4.36 h	4.4×10^{-5}	5.1×10^{19}	6.1×10^{4}
Kr-85	0.3	10.57 y	2.1×10^{-9}	1×10^{19}	$0.6(+2.5 \times 10^{3})$ [a]
Kr-88	3.7	2.77 h	7×10^{-5}	1.25×10^{20}	2.4×10^{5}
I-131	2.9	8.05 d	1×10^{-6}	1×10^{20}	2.7×10^{3}
I-132	4.4	2.4 h	8×10^{-5}	1.5×10^{20}	3.2×10^{5}
I-133	6.5	20.5 h	9×10^{-6}	2.2×10^{20}	5.4×10^{4}
I-134	7.6	52 m	2.2×10^{-4}	2.6×10^{20}	1.6×10^{6}
I-135	6.0	6.7 h	2.9×10^{-5}	2×10^{20}	1.6×10^{5}
Xe-133m	0.16	2.3 d	3.5×10^{-6}	5.5×10^{18}	5.2×10^{2}
Xe-133	6.5	5.27 d	1.5×10^{-6}	2.2×10^{20}	9×10^{3}
Xe-135m	2.1	15 m	7.7×10^{-4}	7.1×10^{19}	1.5×10^{6}
Xe-135	6.2	9.2 h	2.1×10^{-5}	2.1×10^{20}	1.2×10^{5}
Xe-138	5.8	17 m	6.8×10^{-4}	2×10^{20}	3.7×10^{6}

During 3 hours (10,800 seconds)

$$3.7 \times 10^{-8} \times 10{,}800 = 4 \times 10^{-4}$$

of the xenons and kryptons and 4×10^{-5} of the iodines may be expected to escape from the fuel pool. See Table 7.42.

Calculation of xenon-138 concentration at site boundary under inversion meteorological conditions:

$$X/Q = 1 \times 10^{-5}$$

$$Q = \frac{1500}{3 \times 3600} = 0.14 \text{ curie/sec}$$

$$X = 0.14 \times 1 \times 10^{-5} = 1.4 \times 10^{-6} \ \mu c/cc$$

[a] Allowance for inventory.

APPENDICES VIII

Appendix 8.12

Recycled Iodine Activity

Appendix 8.12

Recycled Gaseous Activity (Iodine)

Horizontal distance from stack base to intake is 30 meters.

$$\frac{X}{Q} = \frac{2}{\pi c_y c_z u x^{2-n}} \exp\left[-\frac{h^2}{c_z^2 x^{2-n}}\right]$$

For average conditions

$$\frac{X}{Q} = \frac{2}{\pi(0.4)(0.4)(4)(30)^{1.75}} \exp\left[-\frac{65^2}{(0.4)^2(30)^{1.75}}\right]$$

$$= 1.7 \times 10^{-30}$$

∴ negligible recycle under these conditions.

Assume inversion conditions

$$\frac{X}{Q} = \frac{2}{\pi(0.2)(0.07)(1)(30)^{1.5}} \exp\left[-\frac{65^2}{(0.07)^2(30)^{1.5}}\right]$$

$$\exp\left[-\frac{65^2}{(0.07)^2(166)}\right] = \exp\ (-5.2 \times 10^3)$$

Absolutely negligible.

Assume a condition which brings the plume right down on the stack. Even undiluted, the iodine concentration would be:

$$0.1\ \mu c/sec \text{ in } 32{,}000 \text{ cfm} = \frac{0.1\ \mu c/sec}{1.5 \times 10^7\ cc/sec}$$

$$= 6.7 \times 10^{-8}\ \mu c/cc$$

The 40-hour MPC is $9 \times 10^{-9}\ \mu c/cc$, so this would be exceeded only by a factor of 7.5 if there were absolutely no dilution.

Appendix 8.25

Iodine Thyroid Dose by Recycling
During Criticality Incident

Appendix 8.25

Quantities of iodine released (see Table 7.39)

Dilution by 32,000 cfm for 10 minutes.

Total volume = 320,000 x 28317

$= 9.4 \times 10^9$ cc

Then conc =

I-131 $\frac{73 \times 10^6}{9.4 \times 10^9} = 7.8 \times 10^{-3}$ μc/cc

I-132 $\frac{240 \times 10^6}{9.4 \times 10^9} = 2.5 \times 10^{-2}$ μc/cc

I-133 $\frac{1250 \times 10^6}{9.4 \times 10^9} = 0.13$ μc/cc

I-134 $\frac{17300 \times 10^6}{9.4 \times 10^9} = 1.8$ μc/cc

I-135 $\frac{4400 \times 10^6}{9.4 \times 10^9} = 0.47$ μc/cc

Dilution during recycle

$$\frac{X}{Q} = \frac{2}{C_y C_z u x^{2-n}}$$

Use for x = $\sqrt{h^2 + 30^2}$ x = 72 meters

Use n = 0.25

C_y = 0.25

C_z = 0.2

C_x = 0.25

$$\frac{X}{Q} = \frac{2}{\pi \times 0.2 \times 0.25 \times 4 \times 1800} = \frac{2}{1080} = \frac{1}{500}$$

Using inversion conditions

$$\frac{X}{Q} = \frac{2}{\pi \times 0.2 \times 0.07 \times 600} = \frac{1}{25}$$

Use a value of $\frac{X}{Q}$ of 1/10.

Use a frequency factor of 1/10.

Then

Thyroid Dose During Recycle Coincident With Criticality Incident

Isotope	Conc, μc/cc	$\frac{X}{Q}$ × Frequency	Dose Rate, Rem/ μc/(cc)(sec)	Time, Seconds	Dose, Rem
I-131	7.8×10^{-3}	$\frac{1}{100}$	330	600	15
I-132	2.5×10^{-2}	$\frac{1}{100}$	12	600	2
I-133	0.13	$\frac{1}{100}$	92	600	72
I-134	1.8	$\frac{1}{100}$	6	600	65
I-135	0.47	$\frac{1}{100}$	25	600	70
				Total	224

APPENDICES IX

Appendix 9.9

Curriculum for Chemical Process Operators and Senior Process Operators

Appendix 9.9

Curriculum for Process Operators and Senior Process Operators

A. Chemical Process Operators

1. Introduction-for all trainees

   a. History of plant
   b. Site description
   c. Protection of plant personnel
   d. Protection of public
   e. Licenses and permits required
   f. Purpose of plant
   g. Reason for training
   h. Requirements of trainees
   i. Type of training
   j. Results of training
   k. Industrial relations

2. Lay nuclear physics and chemistry-for all trainees

   a. General description of reactors
   b. Different types of reactors
   c. Nuclear reactors--broadly
   d. Results of reactions
   e. Physical description of various fuels
   f. Significance of fission products and their build up
   g. Reasons for recovery of Source and Fissionable material

3. Process description-for all trainees

   a. Pictures of plant
   b. Model inspection
   c. Input material--form and content
   d. Stepwise handling procedure through process
   e. End product
   f. Packaging and shipping
   g. Waste treatment

4. Reading-for all trainees

   a. Schematics
   b. Instruments
   c. Definition of terms
   d. Data recording

5. Health and Safety program-for all trainees

   a. Elementary radiation theory

      1. Types of radiation
      2. Radiation in perspective
      3. Permissible limits

Appendix 9.9

b. Sources of radiation

1. Natural radioactivity
2. Fall out
3. Man-made sources
4. Fuel elements
5. Normal distribution of radioactive materials in the plant
6. NFS zone designations
7. Potential for accidents involving radioactive materials

c. Criticality

d. Radiation control methods I

Administrative control

e. Radiation control methods II

1. External exposure control
2. Internal exposure control

f. Radiation control method III

Contamination control

g. Scope of the radiation monitoring program

1. The purpose of a fuel processing plant is to make a 3-way split of incoming fuel elements (plutonium, uranium, fission products)
2. Radiation goals to be met
3. General policies used in meeting these goals
4. Services provided by Health & Safety
5. Summary

h. Aids to a good radiation zone job

1. Before start of work
2. During and after the job

i. Use of monitoring instruments for self monitoring

1. Portable alpha counter
   Alpha station monitor
2. Portable beta-gamma counter
   Beta-gamma station monitor
3. Cutie Pie
4. Self reading dosimeters

j. Advanced timekeeping training

   1. Simulated maintenance work with operator keeping time
   2. Practice session with small groups

k. Radiation arithmetic

l. Plant controls and problems

   1. Control features
   2. Special problems

m. Medical program

   1. Physical examination
   2. First aid

n. Chemical safety

   1. Types of chemicals handled
   2. Special hazards
   3. Protective clothing and equipment

o. Fire safety

   1. Description of fire systems
   2. Fire brigade organization
   3. Fire prevention

p. Safe operations of cranes and hoists

   1. Inspection and preventative maintenance
   2. Controls and limit switches
   3. Safe operating techniques

q. Safe operation of vehicles

   1. Heavy equipment
   2. Automobiles and light trucks
   3. Snow removal equipment

6. Equipment descriptions and uses by major area--for all trainees

   a. Fuel Receiving and Storage (FRS)
   b. General Purpose Cell (GPC)
   c. Process Mechanical Cell (PMC)
   d. Equipment Decontamination Room (EDR)
   e. Chemical Process Cell (CPC)

f. Product Packaging and Handling (PPH)
g. Cold Chemical (CC)
h. Control Room (CR)

7. Mechanical manipulation-for selective trainees

a. Fuel Receiving and Storage (FRS)
b. General Purpose Cell (GPC)
c. Process Mechanical Cell (PMC)
d. Equipment Decontamination Room (EDR)
e. Chemical Processing Cell(CPC)
f. Scrap Removal (SR)

8. Chemical processing steps-for selective trainees

a. Sampling
b. Cold Chemical (CC)
c. Product Packaging and Handling (PPH)
Product Packaging and Shipping (PPS)
d. Acid Recovery (AR)
e. Waste evaporation
f. Waste tank farm operations

9. Control room operations-for selected trainees

To include all of item 8 plus control room operations

10. Process maloperation--generally broad--for all trainees

a. Utilities
b. Judgment
c. Other e.g. (fire)
d. Equipment malfunction

11. General decontamination procedures--for all trainees

a. Personnel
b. Equipment

12. Waste treatment-for all trainees except control room trainees

a. Equipment
b. Low level
c. High level

13. General emergency measures--for all trainees

a. Loss from tankage
b. Criticality emergencies

c. Chemical explosions
d. Equipment failure
e. Process emergency procedure

14. Accountability-for all trainees

a. Economic consideration
b. Criticality consideration

15. Ancillary service

a. Utilities
b. Maintenance and shops
c. Warehouses
d. Security

B. Senior chemical process operators. All of the above and in addition:

1. Conditions and limitations in facility license (or authorization)
2. Design and operating limitations in technical specifications
3. Procedures for any changes in (1) and (2) above
4. Somewhat more advanced chemistry and physics
5. Relations with utilities--AEC--ESADA--ASDA
6. Somewhat more advanced radioactivity
7. Somewhat more advanced criticality

Appendix 9.13

Area Radiation Alarm System

Appendix 9.13

Area Radiation Alarm System

The area radiation alarm system consists of fifteen channels with readout and alarm locally and at the control room panel and 5 channels with local readout and alarm. The system is supplied by Tracerlab, a division of Laboratory for Electronics, Inc., Richmond, California.

The fifteen-channel system with remote readout and alarm includes the following:

15-Model TA-6A Remote Detector Assemblies. The remote detectors consist of a Halogen-quenched GM counter with solenoid-operated check source of one microcurie Sr-90, all-transistor ratemeter with Model AX-2 six decade meter, (0.01 mr/hr. to 10 r/hr.) alarm and alarm indicator light, providing station readout and station audio and visual alarm;

15-Model AS-5 Detector and Wall Bracket Sets with combination bracket, handle and bench cradle for detector, and wall mounting plate;

15-Model TA-3 Station Indicators, mounted at the control room panel, with a meter for detector readout and an adjustable alarm and alarm indicator light, providing readout and visual alarm for each remote station and an audio alarm for the system;

3-Model TA-2 Auxiliary Chassis to hold the fifteen station indicators in the control room panel;

2-Model TA-1 Control and Power Units, each of which will power up to 10 remote channels. The TA-1 has a master meter which, by rotary switch, indicates the radiation intensity of any channel detector high voltage or alarm relay voltage and has a push button for check source control;

1-Minneapolis-Honeywell Series 153 24-point recorder to accept and controlthe output from the 15 remote stations and allow extra points for future expansion of the system. The Tracerlab Area Radiation Monitoring System is based on a circuit whereby a small halogen-quenched GM tube, corrected for energy response, is employed in two modes according to the radiation level being monitored. At lower levels approaching background readings, pulses from the detector tube are integrated in a small ratemeter circuit located within the remote detector itself and this ratemeter output current is monitored by the station indicator meter relay. At high radiation levels, the mean tube current becomes significant and is monitored directly by the meter relay. In this mode, there is essentially no circuitry in the readout device. This high current design frees the system

from effects due to cable capacitance, stray fields, temperature drifts, multiple circuit adjustments and other instabilities.

The alarm logic is programmed so that an alarm trip on any channel precipitates the following cycle of events:

a. The master alarm light on the TA-1 Power Supply is illuminated;

b. The alarm light on the appropriate TA-3 Station Indicator is illuminated;

c. The alarm light on the appropriate TA-6 Remote Detector is illuminated;

d. A horn alarm relay is energized, actuating audible alarms. The audible alarm may be immediately disabled, if desired, by a push button on the TA-1 Power Supply. After reset, the audible alarm will again be operable;

e. A motor-driven meter relay pulsing circuit is activated within the TA-1 Power Supply. This circuit allows the TA-3 Station Indicator meter relays to reset at six-second intervals should the radiation level fall below the trip point;

f. The actual radiation level above the alarm point is provided by the TA-1 master meter and the strip-chart recorder.

When the radiation level returns to below the trip point, the following occurs:

a. The master alarm light on the TA-1 Power Supply is extinguished;

b. The meter relays involved in the alarm are reset, to ready them for subsequent trips;

c. The alarm lights on both the TA-3 Station Indicator and the AX-2 remote indicator remain on and serve as a "memory" to identify the alarm tripping channel. These alarm lights must be manually extinguished by pushing a button on the TA-1 Power Supply.

A high or low voltage power failure in the TA-1 Power Supply is indicated by the illumination of:

a. The master alarm light on the TA-1;

b. One of two power failure indicating lights on the TA-1.

The audible alarm circuit is also energized. The failing portion of the circuit may be located by monitoring the high and low voltages on the TA-1 master meter. The five local monitoring channels each will consist of the following components:

1-Model TA-6A Remote Detector Assembly as described above;

1-Model TA-9 Single Channel Indicator and power supply mounted in a Model AX-9 Portable Cabinet, providing detector voltages and radiation level indication on a 2½-inch meter relay with adjustable alarm setting and manual alarm reset;

1-Model AX-5 Detector and Wall Bracket set as described above.

Calibration and Maintenance of Area Gamma Alarm System

Each of the 15 Tracerlab remote gamma detectors and the 5 local gamma detectors is equipped with a solenoid-operated check source of one microcurie strontium-90. The remote system sources are operated by a push button on the control and power unit at the Control Room Panel.

The push button operates all fifteen sources simultaneously. The response of the detector units to the sources will be checked weekly. Once each quarter the system alarm will be tested. Twice each year the sources will be smear checked for leaks.

Liquid In-Line Monitors

Four channels of liquid in-line monitoring are supplied; the first monitoring the waste water to the interceptor, the second monitoring the weak acid from pump 7G-1, the third monitoring the condensate return line, and the fourth monitoring the cooling water return line. The indicator, controls and alarm for the first, third and fourth units are located in the utility building control panel. The indicator and controls for the second unit are in the acid-recovery stairwell and the alarm is at the main control panel, since it is primarily a process monitor. The system is supplied by Tracerlab, a division of Laboratory for Electronics, Inc., Richmond, California. The following equipment is included for the system:

4-Model MD-5B Gamma Scintillation Detectors with 2 x 2-inch sodium iodide, thallium activated crystal, Du Mont Photomultiplier and a 100 gain preamplifier;

4-Detector shield assemblies are designed to position detectors flush with the outside surface of the pipe being monitored. There is three inches of lead shielding around the detector;

4-Model Rm-20B Precision Log Ratemeters as described above;

2-Model Rm-40B Dual Power Supplies as described above;

4-Model MX-9A Check Source and Mechanism with solenoid-operated 9 μc CS-137 source;

1-De Var Series R-300, 3 pen recorder with alarm switches.

Each of the liquid in-line monitors is expected to have a background of about 400 cpm and will produce a count rate of twice background at an in-line concentration of 5×10^{-6} μc fission products/cc.

Calibration and Maintenance of Liquid In-Line Monitor

Each of the liquid in-line monitors is equipped with a solenoid-operated check source of 9 microcuries of cesium-137. Once each week the response of the detector will be checked using the check source. The sources are activated by push buttons at the master control units.

Beta-Gamma Dose Rate Meters

The equipment provided for gamma dose rate monitoring includes eight C.P. meters, two high-range C.P. type meters, and one low-range, low-energy survey meter. The C. P. meters are supplied by Technical Associates, Burbank, California and the low-energy survey meter by Victoreen Instrument Company, Cleveland, Ohio. All are described below:

8-Technical Associates Model CP-4 Survey Meters including bakelite chamber with 6 mg/cm^2 beta end window and removable 432 mg/cm^2 beta absorber. Ranges are 0-50, 500, and 5,000 mr/hr;

2-Technical Associates Model CP-TP with two model 1B chambers and one model 1A chamber, one 40-inch aluminum extention with swivel connector at chamber end and one 15-foot long flexible cable. With 1A chamber, unit has ranges of 0.5, 5, and 50 r/hr. and with 1B chamber the ranges are 50,500, and 5,000 r/hr. The 1A chamber will measure beta plus gamma radiation or gamma only and the 1B chamber will measure gamma radiation;

1-Victoreen Model 440 Low Energy Survey Meter with an air ionization chamber. The current generated as radiation enters the detector is measured by means of a dynamic capacitor electrometer circuit. A single control knob turns the instrument on and provides five full scale ranges of 0-3, 0-10, 0-30, 0-100, 0-300, mr/hr. Energy dependence is ±15% from 6.5 Kev. to 1.2 Mev and ±5% from 80 Kev to 1.0 Mev.

Appendix 9.13

Calibration and Maintenance of CP Meter and Low-Energy Survey Meter and CPTP High-Range Survey Meters

CP Meter and Low-Energy Survey Meter

Before each use the batteries in the CP Meter are tested using three test positions. Once each month the Meters are calibrated on all scales using the 10 millicurie cobalt-60 source.

CPTP High-Range Survey Meters

The response of the high-range survey meters will be checked monthly using the 10-millicurie cobalt-60 source. Since calibration of the upper ranges cannot be accomplished with the available source, these units will be sent out once each year to a commercial calibration and test facility or to the factory.

Neutron Survey Meter

A portable neutron survey meter is provided for fast and thermal neutrons. The meter is supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois.

1-Model 2671 Neutron Survey Meter with Model DN-3 neutron proportional detector and moderator encased in 0.020-inch thick cadmium shield. Moderator wall thickness is one inch. Unit is insensitive to gamma radiation. Ranges of 0-150, 1500, 15,000, and 150,000 cpm are available.

Calibration and Maintenance of Neutron Survey Meter

Since there is no neutron source available for calibration of this instrument, it will be sent to a commercial calibration and test facility once each year.

Emergency Dosimeters

Emergency dosimeters for measurement of gamma and fast neutrons are available. The dosimeters utilize lithium fluoride phosphor and the readout system measures the thermoluminescence of the irradiated phosphor. The dosimeters and readout system are supplied by Controls for Radiation, Inc., Cambridge, Massachusetts and are described below:

1-Model 3100 TLD readout including a heating unit to de-excite the phosphor, a photomultiplier tube and a digital display to indicate the dose. Readout range is 0.1 rad to 10^5 rad;

30-Model Tl-7-T Dosimeter Sets each containing three teflon dosimeters; one with type 7 LiF (enriched to 99.91% Li-7), one with alcohol and type-7 phosphor, and one with type-N phosphor (natural abundance of Li isotopes). The 30 dosimeter sets will be placed around the plant

in operating and office areas to measure emergency radiation doses in these areas.

Calibration and Maintenance Equipment

Equipment is provided for calibration and maintenance of radiation monitors and monitoring systems. This equipment, used by plant forces, will be supplemented by first-year service contracts with the manufacturers of major components and by factory service as required.

Calibrated Radiation Sources

Calibrated radiation sources are provided for standardization of monitoring and sample counting equipment.

1-Model 1235 calibrated gamma disc sources of one microcurie each. Set includes cesium-137, manganese-54, sodium-22, cobalt-60, and barium-133.

1-Set simulated iodine-131 sources of 0.03 and 0.7 microcuries. The source contains a mixture of barium-133 and cesium-137. Sources supplied by New England Nuclear Corporation, Boston, Massachusetts.

1-LH-5(c) calibrated cesium-137 solution. ($CsCl$ in HCl) Approximately 1 μc in 3 ml.

1-R-31(c) 10 millicurie cobalt-60 calibrated source with 4-inch shield, 12-inch handle and wrench.

3-R-240 (c) strontium-90 calibrated source, two-inch diameter mount with 0.9 mg/cm^2 mylar cover. Approximately 0.02 microcuries total activity. Sources supplied by Tracerlab, a division of Laboratory for Electronics, Inc., Waltham, Massachusetts.

1-Model S94-2. Three calibrated plutonium standards with activities of about 1,000, 10,000, and 100,000 disintegrations per minute.

3-Model SD-1 calibrated plutonium standards with activities of about 10,000 disintegrations per minute each. Sources supplied by Eberline Instrument Corporation, Santa Fe, New Mexico.

Equipment for Calibration and Maintenance

Equipment provided to aid in the maintenance and calibration of equipment includes a Simpson Model 2610 wide-band oscilloscope, a Simpson Model 270 VOM with accessory battery tester, an Atomic Assessories Model PRG-159 pulse generator, soldering irons and assorted tools.

Appendix 9.17

Film Badge and Dosimeter Monitors

Protective Clothing and Safety Equipment

Station Monitors

Hand and Foot Counters

Appendix 9.17

Film Badge and Dosimeter Monitors

Film badges to detect exposure to beta-gamma and in some cases, neutron radiation will be issued to all plant employees. Beta-gamma film packets will be changed monthly for employees in the Plant Managers Office, Office Management, Personnel, Security, Sales and Service, Engineering, Technical Services, and for all secretarial and clerical personnel. Beta-gamma film packets will be changed weekly for employees in Production, Health and Safety, Analytical Laboratories and Plant Maintenance Departments. Neutron film packets will be included for area monitoring in Product Packaging and Shipping, Maintenance and Health and Safety. The film badge service will be supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois.

The Nuclear-Chicago badge holder is composed of plastic front and back sections which tightly sandwich filters of copper and cadmium on both surfaces of the film. The front filters are circular while the rear filters are rectangular. Darkening of the entire film indicates exposure to gamma radiation. If the cadmium filter is clear while the rest of the film is darkened, it is a sign of exposure to low energy gamma or X-ray radiation. If the area under both the cadmium and copper filters remains clear while the rest of the film is darkened, exposure to beta radiation is indicated. Because of the use of differently shaped filters, the exposure pattern on the film can also indicate if the wearer was exposed to radiation from the back rather than the front of the body. The film, Eastman Kodak 3N, has an effective range for X-rays and gamma rays from 0.02 to 10.0 mev. and for betas from 0.2 to 10.0 mev. Through each step of film processing-from pre-shipment refrigeration to the electronic measurement of film exposure-safeguards are provided which assure accuracy. The film is developed under carefully controlled conditions along with other film from the same batch which has been exposed to a known amount of radiation. The developed film is measured electronically and the findings are relayed to automatic data handling equipment where exposures are recorded on a report form which provides all necessary information for each individual monitored. The report form has all pertinent data presented on a single, letter-size page. The report lists the film number, film data, wearer's identification, current reading for X, gamma, and beta radiation, as well as quarterly summaries and total dosage for the year-to-date. The report form also shows the number of readings from which results were compiled. The accuracy of the reported dose is $\pm$ 10% over the range of 0.25 rem to 100 rem.

Dosimeters

Both direct reading and indirect reading gamma dosimeters are available for use by plant personnel. The dosimeters and auxiliary equipment are supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois.

Appendix 9.17

400-Model NC-405 Indirect Reading Dosimeters with 0-200 mr. range. Energy dependence is less than 15% of true dose at energies as low as 30 kev and units are dose rate independent from 1 mr/hr to 100 r/hr.

1-Model NC-404 Charger-Reader.

25-Model NC-402 Direct Reading Dosimeters with 0-200 mr. range. Energy dependence is ± 10% of true dose from 55 kev to 2 mev. The maximum leakage is 2% in 24 hours.

1-Model NC-403 Charger.

Appendix 9.17

Protective Clothing and Safety Equipment

Following is the startup supply of protective clothing and equipment available at the plant

a. 20 dozen work shirts
b. 20 dozen pair work trousers
c. 20 dozen pair coveralls
d. 5 dozen laboratory coats
e. 20 dozen pair cloth boots
f. 20 dozen surgical type cloth hats
g. 5 dozen cloth hoods
h. 24 MSA Ultra Filter Masks
i. 18 MSA Air Line Respirators
j. 12 MSA Air Masks
k. 12 MSA Face Shields
l. 144 MSA Softsides Goggles
m. 12 MSA Plastic Suits
n. 500 dozen pair 6 mil PVC gloves
o. 36 dozen pair lined latex gloves
p. 12 dozen pair dry box gloves
q. 12 dozen pair leather palm gloves
r. 2 Safety harness
s. 3 Stretchers
t. 3 Fire Blankets
u. 144 Hard Hats
v. 1 pair Safety shoes for each production employee
w. 20 dozen pair shoe covers

Maintenance and Inspection of Protective Clothing and Equipment

Coveralls, shoe covers, gloves and related items of apparel will be collected, monitored, sorted according to levels of contamination, and laundered after each days use. Any clothing contaminated to greater than 50 mrad/hr beta-gamma or 50,000 d/m alpha will be packaged for burial. No attempt will be made to launder these items. All clothing will be spot checked after laundering for residual contamination. Contamination in excess of 0.2 mrad/hr beta-gamma or 1000 d/m alpha will require that the clothing be re-laundered and resurveyed. Items which can not be cleaned below these levels will be discarded. After each use, masks will be surveyed and released if contamination levels are less than 500 d/m alpha and 100 c/m beta-gamma. If contamination exceeding these levels is detected, the masks will be set aside for special decontamination. The contaminated areas will be cleaned by hand, taking special care to prevent spread of contamination to the inside of the mask. When released, the masks will be washed in a solution of MSA cleaner-sanitizer, rinsed in clean water, dried, and packaged in plastic bags. Filter canisters will be handled separately. Canisters will be surveyed, cleaned if necessary and stored apart from the masks. Contamination limits for non smearable contamination on canisters are 100 d/m alpha and 0.2 mrad/hr beta-gamma.

Appendix 9.17

Station Monitors

Alpha and beta-gamma station monitors are provided at personnel check stations throughout the plant.

There are 18 alpha station monitors supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois. Each unit consists of a Model 8619 ratemeter with speaker and meter for audio and visual indication of counting rate, and a Model AP4 alpha air proportional probe with a 4 foot long cord and alpha check source.

There are 24 beta-gamma station monitors supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois. Each unit consists of a Model 6327 ratemeter with speaker and meter for audio and visual indication of counting rate and a side window beta-gamma probe with a 4 foot long cord and beta check source.

Calibration and Maintenance of Alpha Station Monitors and Beta-Gamma Station Monitor

Before each use the response of the alpha and beta-gamma station monitor is checked using the source supplied with each unit.

Appendix 9.17

Hand and Foot Counters

Two beta-gamma hand and foot counters are provided. They are to be located in the Main Entrance Lobby to serve as a final contamination check before entering the lunch room or before leaving the plant. The counters are supplied by Eberline Instrument Corporation, Santa Fe, New Mexico and are described below:

> 2-Model HFM-2 Beta-Gamma Hand and Foot Monitors with external probe for clothing survey. The system operates continuously using 4 Amperex 90NB GM tubes in each hand and foot cavity. Cavity shielding is equivalent to 1 inch of lead. The external probe is a halogen quenched GM tube mounted in a Model HP-177 side window hand probe. Four 100 μa relay type meters are used to accept the output from the nand and foot cavities. One four inch edge reading meter is used for the external probe. Meter ranges are 0-500, 0-2000, 0-5-000 and 0-20,000 cpm with scale selector switch mounted internally. A single speaker with variable volume control provides an audio indication of count rate. If the count rate exceeds a preset level, a buzzer alarm sounds and warning lights indicate the source of the contamination.

Testing of Hand and Foot Monitors

The detectors in the hand and foot counters will be checked for response daily by positioning a beta source over each.

Appendix 9.33

Air Sampling and Air Monitoring Equipment

Appendix 9.33

Air Sampling and Air Monitoring Equipment

Site Perimeter Air Monitor

Continuous Air Monitors are located at three points around the perimeter of the service center. The units are supplied by Tracerlab, a division of Laboratory for Electronics, Inc., Richmond, California and are described below:

3-Model MA-5B Fixed Filter Air Particulate Monitors with specially designed, heated, ventilated, enclosure and filter holder modified to hold one particulate and one charcoal filter in series.

3-MM-6B Log Ratemeter with 2½ inch meter indicating from 20 to 200,000 cpm and a switch selected scale for monitoring high voltage. Time constants vary with counting rate from 60 seconds at 20 cpm to 50 milliseconds at 200,000 cpm.

3-Model MD-1B End Window Beta-Gamma G. M. Detector, a 2½-inch O.D. cartridge containing an Amperex 100-NB halogen quenched GM tube. Following the GM tube is a trigger circuit that gives a 4 volt-2 microsecond pulse into a terminated 93 ohm output cable.

3-L and N model S Continuous Strip Chart Recorders.

Each monitoring unit is placed on a ten foot high platform to keep it above the maximum anticipated snow level.

Plant Site Air Sampler

An air sampler is available for sampling air around the plant site. The unit is supplied by Gelman Instrument Company, Chelsea, Michigan and is described below:

1-Model 26001 Nuclear Air Sampler capable of sampling continuously at a constant rate of 1 CFM. The flow is controlled by a limiting orifice installed in the sampling line between the filter bowl and intake of a vacuum pump. The amount of air sampled is recorded on a dry gas meter and a vacuum gauge is included to correct the indicated flow for error due to pressure drop across the filter. A running time meter indicates cumulative operating time in hours and tenths. Samples are collected on two 2-inch in-line type filter holders. The entire assembly is housed in a heavy gauge steel cabinet fitted with louvers for ventilation.

Appendix 9.33

Plant Air Particulate Sampling System

An in-plant air sampling system is available utilizing a central vacuum pump and vacuum headers to all building occupied areas. There are 54 area air sampling stations and 19 in-cell remote air sampling stations available for use.
Each area air sampling station consists of a line to the vacuum header with a valve, a Gelman Model 8224, 10-84 lpm air flow meter and a Gelman Model 1200-A, 2 inch diameter open filter holder.
Each remote air sampling station consists of a line to the vacuum header with a valve, a Gelman Model 8224, 10-84 lpm air flow meter, a Gelman Model 1200-C 2 inch diameter closed filter holder, another valve and an offset penetration to the cell or remote area.

Continuous Air Monitors

Seven continuous air monitors are provided. The units are supplied by Nuclear Measurements Corporation, Indianapolis, Indiana and are described below:

1-Model PAPM-1 Programmed Alpha Plutonium Monitor including two ASC-1 alpha scintillation detectors utilizing ZnS phosphor and one LCRM-55 dual logarithmic count-ratemeter with two 5-cycle meters range 10 to 1,000,000 cpm and power supply. Each ratemeter has a dual contact meter manually set at a chosen scale for alert or fail-safe and alarm condition. One continuous duty positive displacement industrial air pump driven by a belt coupled, sealed ball bearing motor with an automatic switching valve which shifts collection from one collector to the other. The time cycle is controlled by a programmer with 1 through 24 hour cycles available. The count-ratemeter output is recorded on a two pen continuous strip chart recorder. During the last hour of off-collection time, the activity remaining on the filter is counted and the total count is printed out on paper tape. Assuming a 10 cfm sampling rate and a concentration of plutonium in air of 10^{-12} μc/cc, the build-up activity on the filter paper would be 37.8 cpm per hour of which 37% or 13.8 cpm would be detected. At the end of 12 hours the detector would see 165 cpm above background, not enough to cause an alarm. The air pump would then cycle to the other collector and the natural activity on the first collector would be allowed to decay for 11 hours. Then, from the 23rd to the 24th hour following the initial collection, the total count on the first collector would be recorded. The natural activity background should be about 300 to 500 counts per hour and the plutonium count would be about 9,900 for the one hour count. The unit then would alarm after 24 hours in a

concentration approaching the 40 hour M.P.C. If the concentration was 10^{-11} μc/cc the first detector would see 1100 cpm above background after 8 hours of collection and this would probably cause an alarm in the counting ratemeter. The unit will detect either a low level build up or a sudden burst of plutonium contamination and will alarm before the exposure of personnel exceeds the limits specified in 10 CFR-20.

1-Model AM-2A Fixed Filter Air Particulate Monitor. One ASC-1 alpha scintillation detector with ZnS phosphor. One LCRM-2M count ratemeter with one 3 cycle logarithmic scale of 50-50,000 cpm. Detector is shielded by 2 inches of lead equivalent. The air pump is a continuous duty positive displacement industrial type driven by a belt coupled, sealed ball bearing electric motor. Manually set alarm points with alert and alarm settings. Count-ratemeter output is recorded on a continuous strip chart recorder. Alpha air contamination of 10^{-12} μc/cc and sampling rate of 5 cfm will result in 7 cpm build-up per hour.

5-Model AM-2A Fixed Filter Air Particulate monitor identical to the unit described above except that the detector is a DGM-2 end window GM and the filter holder is modified to accept two filters in series, one particulate and one activated charcoal for collection of iodine-131. A concentration of 10^{-10} μc/cc and a sampling rate of 5 cfm will result in 350 cpm build-up per hour.

Calibration and Maintenance of Continuous Air Monitors

All the continuous air monitors will be calibrated monthly by analyzing the filters in the counting room and comparing the results with the count rate observed at the air monitors. Response of each unit to radiation will be apparent because of the natural activity filtered out of the air.

Medical Monitoring Equipment

Thyroid Monitor

A thyroid monitoring system is available for detecting iodine-131 deposited in the thyroid. The system is supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois and is described below:

1-Model 612 Collimated scintillation detector with 3 inch diameter by 1½-inch thick sodium iodide, thallium activated crystal and DuMont 6363 photomultiplier;

1-Model 1720 Support Stand with arm. The arm can be automatically positioned at any height from 12 to 66 inches above floor level with a reversible electric motor which drives a

precision ball-screw inside the vertical column. The motor control switches are located at the end of a 30-inch coil cord.

1-Model 132-B Analyzer Computer. The 132-B combines a precision single channel pulse height analyzer, regulated high voltage supply, a binary scaler and a computing-circuit.

A plutonium gamma detector is available for detecting plutonium contamination in wounds. The detector-ratemeter system is supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois and consists of the following:

1-Model 644 (DS8-21) gamma scintillation detector with a ½-inch diameter by 2 mm thick sodium iodide crystal. The crystal is coupled to the photocathode of a ten stage photomultiplier tube through a short light pipe. The crystal projects through a tight flange and has a ½-inch diameter by 0.0005 inch thick beryllium window to allow detection of low energy radiation without appreciable loss. Efficiency is about 90 per cent for gamma rays of less than 35 kev. Unshielded background is 5 to 10 cpm.

1-Model 8619 Labitron Ratemeter with 4½-inch meter, speaker with volume control and ranges of 0-500; 2,000; 5,000; and 20,000 cpm.

Equipment for Detection of Gases and Vapors

1-Universal Testing Kit, Model 2. Kit includes a piston type pump with a turret head and four orifices sized for optimum sampling rates, a calibrated handle to permit sampling volumes of 25, 50, 75, or 100 cc. and a remote sampling attachment for hard-to-reach spots. Kit provides capability for sampling carbon monixide, hydrogen sulphide, chlorine, mercury vapor, nitrogen dioxide, carbon dioxide, unsaturated hydrocarbons, phosgene, hydrocyanic acid gas, aromatic hydrocarbons, sulphur dioxide, halogenated hydrocarbons, lead-in-air, chromic acid mist, hydrogen fluoride, arsine, boranes-in-air and unsymmetrical dimethyl hydrazine.

1-Model 53 Gascope for detection of natural gas in air. The instrument has a dual range with one scale graduated from 0-100% of the lower explosive limit of natural gas in air and the second scale graduated from 0-100% by volume natural gas.

Appendix 9.36

Portable Monitoring Equipment

Appendix 9.36

Portable Monitoring Equipment

Alpha Detectors

The equipment provided for the detection of surface alpha contamination includes four portable alpha counters and one alpha floor monitor. These instruments are supplied by Eberline Instrument Corporation, Santa Fe, New Mexico, and are as described below:

4-Model PAC-3G Portable Gas Proportional Alpha Counters. Instrument grade propane flows through the probe at 30 cc per minute. The probe has an active surface area of 61 square centimeters. The instrument has three ranges, 0-1000, 0-10,000 and 0-100,000 cpm. Phones for aural monitoring and a uranium oxide check source are included.

1-Model FM-3G Gas Proportional Alpha Floor Monitor. Active probe area of 68 square inches for faster surveying of large, open floor areas. Three ranges, 0-1000, 0-10,000 and 0-100,000 cpm. Speaker and phones supplied for aural monitoring. The unit is mounted on wheels and the probe height from the floor is adjustable for 1/8 to 1/4 inches with additional adjustment to 2 inches for safe transportation.

Calibration and Maintenance of Portable Alpha Counters

The bi-monthly calibration procedure for portable alpha counters is as follows:

a. Check each scale using the calibrated plutonium-239 sources provided and adjust to the proper response.

b. Check the response of the instrument to the uranium check source.

c. Hold the instrument probe against the radium source container and, using the gain adjustment, tune out any response to the gamma radiation.

d. Recheck each scale with the plutonium-239 sources if a gain adjustment was necessary.

Appendix 9.36

Calibration and Maintenance of Alpha Floor Monitor

Twice each month the response of the alpha floor monitor will be tested using the uranium check sources.

Beta-Gamma Detectors

Beta-Gamma detection equipment includes: four GM Meters supplied by Victoreen Instrument Company, Cleveland, Ohio; one deep hole monitor supplied by Nuclear Chicago Corporation, Des Plaines, Illinois; and one floor monitor supplied by Eberline Instrument Corporation, Santa Fe, New Mexico. This equipment is described below:

4-Victoreen Model 489 Thyac II GM Survey Meter with Model 489-4 probe. The detector has a sliding metal window for beta discrimination and a 360 degree window for maximum beta-gamma sensitivity. The meter has three ranges of 0-800, 0-8,000 and 0-80,000 cpm and a built-in check source and phone for aural monitoring.

1-Nuclear Chicago Gamma Radiation Monitor for deep holes. The unit consists of a gamma scintillation detector in a waterproof, shock resistant housing, 150 feet of cable, Model 8619 ratemeter and a strip chart recorder.

1-Eberline Model FM-1 beta-gamma floor monitor. The detectors, Amperex 912NB GM tubes, are mounted in a steel encased lead shielded housing with an effective monitoring width of 21 inches. The shield can be rotated 45 degrees to check base-boards and other vertical surfaces close to ground level. The electronic, Model E-112B-1, has three ranges, (0.2, 2.0 and 20.0 mr/hr. full scale) with ratemeter, hand probe and phones for aural monitoring.

Calibration and Maintenance of Portable GM Counters

Before each use the response of the GM meter will be tested using the source supplied with each unit. After any maintenance has been performed on a unit, it will be calibrated using the calibrated 10 millicurie cobalt-60 source.

Calibration and Maintenance of Deep Hole Monitor

Before each use the response of the deep hole monitor will be checked using the radium source.

Appendix 9.36

Calibration and Maintenance of Beta-Gamma Floor Monitor

Twice each month the response of the beta-gamma floor monitor will be tested using beta check sources.

Appendix 9.37

Counting Room Equipment

Appendix 9.37

Counting Room Equipment

Liquid Scintillation Counting System

A Liquid Scintillation counting system is provided for the detection of tritium in samples. The system is supplied by Packard Instrument Company, Inc., La Grange, Illinois. The Model 314-EX2, as supplied, includes the following:

1-Tri-Carb Spectrometer, a two channel unit with two scalers, red and green, and an electronic timer. All three units have glow tube decade readout. Each channel has discriminator controls providing separate channels of pulse height analysis. Preset time control is in 20 steps from 3 seconds to 100 minutes. Preset counts may be selected on either scaler in increments from 10^2 to 10^6. Preset time and both preset count settings interact so that whenever any limit is reached the count will stop;

1-Model 500-C Automatic Control Unit and 100 sample automatic changer, with two photomultiplier detectors, a monitor detector and an analyzer detector monitoring the sample well. The automatic changer and detectors are mounted in an eleven cubic foot freezer for controlled temperature counting. The automatic control unit programs operation of the sample changer. Controls may be set to count anywhere from 1 to 100 samples and the unit will recycle continuously if repeat data are required on a batch of samples. The unit may also be set for repeat counting of a single sample. If power failure should occur while a count is in progress, the control unit will clear and repeat the count automatically when voltage is restored. A manual over-ride button allows the operator to select any sample for a special count;

1-Model A Digital Printer provides a printed record of counting data on a strip of paper tape. For each sample, the printer records sample number, elapsed time and counts on both scalers.

In the operation of the liquid scintillation counting system, radioactive decay events occurring in the sample cause scintillations which are seen simultaneously by both photomultiplier tubes, giving rise to pulses at the phototube output. Pulses from the photomultipliers pass through preamplifiers and into three separate amplifiers. Pulses from the "Analyzer" phototube then go to the discriminator pairs A-B and C-D for pulse-height analysis. The "Monitor" phototube functions only to determine whether a pulse is the legitimate result of a decay event or whether it arises from photomultiplier tube noise. Pulses falling between A and B are fed to the red scaler and pulses falling between C and D are fed to the green scaler. Output pulses from all of the discriminators pass through the coincidence circuitry and only pulses occurring simultaneously in both photomultipliers are counted. This results in some loss of

efficiency but effectively eliminates phototube noise. The two channels may be used to estimate the amount of quenching in a sample so that appropriate correction factors may be applied to the count. The disintegration rate of an unknown sample may be determined by counting the sample with the two channels operating first separately and then in coincidence. Based on the approximation that coincidence counting efficiency is a product of the two single-channel efficiencies, the disintegration rate is found from the equation:

$$\text{dpm} = \frac{\text{Counts (red) x Counts (green)}}{\text{Counts (coincidence)}}$$

Calibration and Maintenance of Sample Counters

The gas proportional alpha and beta sample counters will be calibrated and source checked according to the following procedure after any maintenance has been performed on the units.

a. From a series of twenty-five minute counts of a calibrated alpha or beta source determine:

1. Chi-square

$$\text{Chi-square} = \frac{\sum (x - \bar{x})^2}{\sum x}$$

2. Geometry

$$G = \frac{\bar{X}}{\text{Source d/m}}$$

3. Standard Deviation

$$\text{S.D.} = \sqrt{\frac{(X - \bar{X})^2}{n-1}}$$

4. Error

$$E = \frac{k\ (\text{S.D.})}{\sqrt{P}} \qquad P = \frac{\text{Time of sample count}}{\text{Time of control count}}$$

b. From the data derived above, establish a maximum and minimum counting rate for the 90% and 95% confidence intervals.

c. Each day check the response of the sample counters to the calibrated source. If more than one count in ten exceeds the 90% limits or more than one count in twenty exceeds the 95% limits, the unit is removed from service until repaired and recalibrated.

Calibration and Maintenance of Liquid Scintillation System

The calibration and source check procedure for the liquid scintillation system will be identical to that outlined above using a calibrated tritium source.

Gamma Spectrometer

A continuous scan gamma energy analyzer is provided for analysis of the activity and gamma energy distribution of any gamma emitting sample. The system, supplied by Nuclear Measurements Corporation, Indianapolis, Indiana, is designated Model GSS-1B and consists of the following components:

1-Model WSC-3S Well Scintillation Detector with 3 x 3-inch sodium iodide, thallium activated crystal, spectrometer grade. The well has 100 cc volume.

1-Model US-11 Super Shield providing 4½ inches of lead shielding around the detector and a counter-balanced lid.

1-Model PHA-1B Pulse Height Analyzer with linear count-ratemeter auto scanner and binary scale factor selector. Standard energy range is 30 kev to 3 mev. Count-ratemeter ranges are 0-300, 1000, 3,000, 10,000, 30,000, and 100,000 cpm. Time constants are 0.3, 1, 3, 10, and 30 seconds. Spectrometer window width is variable in ten steps from 0 to 30%. Three position scan speed selector, 10, 25, and 60 minutes.

1-Model GR-5 X-Y Spectrometer Graphic Recorder. Chart size is 8½ inches x 11 inches. Maximum pen speed is 7.5 inches per second.

1-Model SDS-1B Slave Decade Scaler with timer.

The NMC Model GSS-1B is an automatic scan pulse height analyzer system which provides a graphic record of the activity and energy distribution of any gamma emitting sample. A constant percentage of each gamma energy peak is analyzed. The fixed window counts only those pulses brought to it by the amplifier. The system uses a sliding pulse amplification technique and is capable of scanning the gamma spectrum in a range of 0.1 kev to 6 mev. Special recorder paper is available for nonstandard ranges. Both automatic and manual scan control are provided. Individual peak monitoring may be accomplished directly on the graph paper using the slave scaler to integrate the total count under the peak.

Calibration and Maintenance of Gamma Spectrometer

The response of the gamma spectrometer to a calibrated source will be checked daily as outlined above. Pulses will be fed into the X-Y recorder from a pulse generator. The recorder will be adjusted to the exact pulse height and input rate. This will be performed when the daily source checks indicate that the instrument response has shifted.

Appendix 9.37a

Determination of Beta Emitters in In-Plant Air Samples

Appendix 9.37a

Determination of Beta Emitters in In-Plant Air Samples

The concentration of beta emitters in in-plant air samples is determined as follows:

$$\mu c/ml = \frac{c/m\ (Factor)}{M^3}$$

$\mu c/ml$ = Microcuries per milliliter

c/m = Beta counts per minute on sample corrected for background.

M^3 = Total cubic meters of air sampled

$$Factor = \frac{1}{c\ g\ a\ K_1\ K_2}$$

c = Collection efficiency = 98%

g = Geometry of counter = 50%

a = Absorption correction = (not applicable -- see paragraph 9.34)

K_1 = $d/(m)/(\mu c) = 2.22 \times 10^6$

K_2 = Milliliters per cubic meter = 10^6

$$Factor = \frac{1}{(.98)\ (.50)\ (2.22 \times 10^6)}$$

$$Factor = 9.2 \times 10^{-13}$$

$$\mu c/ml = \frac{\mu c/m\ (9.2 \times 10^{-13})}{M^3}$$

Since, at 60 l/m a 24-hour sample represents $86.4M^3$ sampled and the counting error for a one-minute count at 95% confidence level is ± 10% at 400 c/m, the minimum detectable concentration is:

$$\frac{(400)\ (9.2 \times 10^{-13})}{86.4} = 4.3 \times 10^{-12}\ \mu c/ml$$

with ± 10% accuracy.

Appendix 9.37b

Determination of Long-Lived Alpha Emitters in In-Plant Air Samples

Appendix 9.37b

Determination of Long-Lived Alpha Emitters in In-Plant Air Samples

The concentration of long-lived alpha emitters in in-plant air samples is determined as follows:

$$\mu c/ml = \frac{C_p \text{ (Factor)}}{M^3}$$

C_p = Calculated c/m due to product

$$= \frac{C_{24} - C_6 e^{-\lambda \Delta t}}{1 - e^{-\lambda \Delta t}}$$

C_{24} = 24-hour count

C_6 = 6-hour count

Δt = Time of 24-hour count minus time of six-hour count

M^3 = Total cubic meters sampled

$$\text{Factor} = \frac{1}{c\ g\ a\ K_1\ K_2}$$

c = Collection efficiency = 98%

g = Geometry of counter = 50%

a = Absorption correction = 70%

K_1 = d/(m)/(μc) = 2.22×10^6

K_2 = Milliliters per cubic meter = 10^6

$$\text{Factor} = \frac{1}{(.98)\ (.50)\ (.70)\ (2.22 \times 10^6)\ (10^6)}$$

$$\text{Factor} = 1.3 \times 10^{-12}$$

$$\mu c/ml = \frac{C_p\ (1.3 \times 10^{-12})}{M^3}$$

Since the counting error for a five-minute count at 95% confidence level is ± 10% at 75 c/m, the minimum detectable concentration on a 24-hour sample is:

$$\frac{(75)\ (1.3 \times 10^{-12})}{86.4} = 1.1 \times 10^{-12}\ \mu c/ml$$

with ± 10% accuracy.

Appendix 9.39a

Low Background Counting System

Appendix 9.39a

Low Background Counting Systems

A low background sample counting system is provided for automatic counting of two inch diameter planchets and air samples. The counting system is supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois and is described below.

1 - Model 1105 sample changer with sample and guard detectors and shielding.

The detector system consists of a 2¼-inch diameter plastic gas flow detector with a 0.75 inch thick oxygen free high conductivity copper shield. Effective window diameter is therefore in excess of 2 inches.

The copper shield and gas detector assembly is recessed into one face of a cylinder of Pilot B plastic scintillator. The opposite face is optically coupled to a 3-inch multiplier phototube, forming the scintillation detector anti-coincidence guard. The entire detector system is housed in a stainless steel enclosure and, when in counting position on the C210 high lift changer, is shielded in all directions by an average of 3 inches of virgin lead.

The detector system will operate in the alpha or beta proportional mode (with PR gas 90 percent argon-10 percent methane), and will provide simultaneous alpha/beta proportional operation. The gas flow detector can be operated window or windowless.

The scintillation detector is operated in anti-coincidence with the gas flow detector in all modes of operation, thus it will suppress all gas detector outputs when a simultaneous event is detected in the scintillation channel.

The electronic circuitry consists of the following:

1. Scintillation detector amplifier system and threshold discriminator;

2. Beta amplifier system and threshold discriminator;

3. Alpha amplifier system and threshold discriminator.

4. Logic circuitry to provide the necessary output information, + 10 V or -2.0 V, switch selected;

5. A 500 to 1000 volt HV supply for the scintillation detector;

6. A low voltage operating power supply;

7. C210 sample changer control circuitry including a gas flow system for window or windowless operation. The gas flow detector operating voltage is provided by one of the output scalers.

Front Panel Controls

1. Scaler 1 Output - Provides "Gross Alpha", "Net Alpha", "gross Beta", "Net Beta", "Guard" or "Net (Alpha + Beta)".

2. Scaler 2 Output - Provides "Gross Alpha", "Net Alpha", "Gross Beta", "Net Beta", "Guard", or "Cancelled Background". Note that Cancelled Background refers to those counts occurring approximately coincidently in the "Guard" and "Sample Counter".

3. Mode Control - Selects Geiger or Proportional operation and internally controls dead time and modifies logic circuitry depending on operating mode.

4. Guard Gain - Controls high voltage to scintillation detector.

5. Alpha Gain - Controls gain of alpha amplifiers to adjust discriminations between alpha and beta pulses.

6. Guard HV - Switches HV on-off to scintillation detector.

Front Panel Controls - Sample Changer Controller

1. Program - Selects Manual, One Cycle Stop, One Cycle Background or Auto Recycle mode.

2. Flush - Selects flush time for "Window" or "Windowless" operation with an "Off" position. A flush position in Window operation is provided to flush radon from the air over the sample in window operation thereby contributing to the low background.

3. Print - Orders Clary printing timer to print.

4. Index Reset - Resets index number in Clary.

5. Reject - Reject sample in counting position, commands Clary to print, and advances sample numbers in Model 8274 scaler.

6. Restack - Restacks samples from left column to right column in sample changer.

7. Start - Initiates sample changer operation.

8. Flushing/Counting Light - The counting light will flash on and off during the flush time and will remain on during counting time.

The information available at the two scaler output connectors is identical with the exception of the last switch position on the sample changer controller. Scaler 1 can display a net (Alpha + Beta) and scaler 2 can display cancelled background. This redundancy permits the greatest flexibility in the readout format and is particularly useful in the Alpha/Beta mode where the 8274 scaler is employed. The following ratios can be conveniently obtained: Alpha/Beta; Beta/Alpha: Beta/(Alpha + Beta); Alpha/(Alpha + Beta). In addition, for purposes of selfchecking, the following information is very useful.

Cancelled Background - Those counts in the Alpha or Beta channels which were blanked as a result of a coincidence with counts in the guard channel.

Guard - All events exceeding the guard channel threshold.

Gross Alpha or Gross Beta - True counts plus actual background.

1 - Model 8274 Dual Scaler/Timer with high voltage supply and printing calculator.

Scaler/Timer

Complete monitoring and count control functions for detectors and sample changers are handled by two fast-decade scalers and an electronic timer. Each scaler accumulates the count from the detector

connected to it, while the timer supplies time control for each scaler simultaneously. The display offers continuous in-line numerical readout of the count from each scaler and the counting time in minutes and hundredths of a minute. A front panel switch selects the display to be presented on the numerical indicators.

Manual, preset count, preset time, or preset time/count operating modes are provided, with a wide selection of preset counts for each scaler and preset time intervals for the timer. Electronic gating is used to control the scaling and timing, thus eliminating start and stop timing errors usually inherent with mechanical relays and switches.

Ultrascaler II offers the following operating modes:

(1) Manual: The count is controlled by start and stop pushbuttons.

(2) Preset Time: The count terminates when a selected time interval elapses.

(3) Preset Count: The count terminates when either scaler reaches a selected count.

(4) Time/Count: The count terminates when a preset time or preset count, whichever occurs first, is reached.

The scaler/timer module also incorporates the program controls for the automatic calculator of Model 8274. The lister or calculator can be programmed for fully coordinated operation with automatic sample changers. All necessary start, stop, reset, print, and sample reject commands are performed in proper sequence. For manual sample changing the lister or calculator operation can be manually controlled at the scaler. The power supply for the scaler/timer is housed separately in a force-ventilated compartment at the top of the control console. No high-temperature producing components are located on the scaler/timer main chassis.

The Ultrascaler II Model 8274 offers an automatic calculator that provides, in addition to the data listing features of the printing lister, automatic calculation of counts per minute on each scaler and the ratio between two scaler counts. This latter feature is very useful for applications such as determining the ratio of alpha-beta concentration in a sample, or the ratio of counts in two separate parts of a single spectrum. Five different sequences of data readout programming can be selected for the automatic calculator.

(1) Listing only-sample number, time, and counts accumulated on each scaler.

(2) Listing and counts per minute for each scaler.

(3) Listing, counts per minute, and ratio of scaler B/scaler A.

(4) Listing and ratio B/A.

(5) Fast print-out of counts accumulated on scaler A only.

This program is suitable for radiochromatography applications such as continuous flow counting. Print-out and reset time is one (1) second total.

All data are clearly printed on easily replaceable paper tape. The sample index number, time, and count data are keyed to their respective positions on the tape and are always recorded in proper sequence. Ultrascaler II provides the sample number indexing facilities for the sample changers-from 1 to 100 samples in sequence. These index numbers are premanently keyed to the index number print-out of lister and calculator for positive sample identification throughout the sample-changing cycle. Control circuits are incorporated for manually advancing the index number in increments of 1 or 10 for situations where the sample changer is not starting with sample number 1. The sample index number may also be displayed by the numerical indicators if desired.

Two sample counting systems are provided for automatic counting of two inch diameter planchets and air samples. The counting systems are supplied by Nuclear-Chicago Corporation, Des Plaines, Illinois

and are described below:

2 - Model 1105 sample changers as described above.
2 - Model 8710 Scaler/Timers with high voltage supply and automatic printing lister that furnishes a record of sample numbers, time and count.

Complete monitoring and count control functions for detectors and sample changers are handled by a fast-decade scaler and an electronic timer. A continuous, numerical display shows both the count accumulating in the scaler and counting time in minutes and hundredths of a minute. The display consists of six vertical neon decades for the scaler and five for the timer. Manual and preset time/reset count operating modes are provided with a selection of preset counts and preset times more than sufficient for practically any application. Both preset time and preset count are set by means of unique sliding switches on the scaler/timer decades. This design allows a selection of preset times over the complete range of the timer and a choice of preset counts from 10 to 999,990 in increments of one count. In the manual operating mode, the count is controlled by convenient stop, reset, and start pushbuttons. In the time/count mode the count terminates when a preset time or preset count--whichever occurs first--is reached, or when the stop pushbutton is pressed. Electronic gating is used to control the scaling and timing, thus precluding start and stop timing errors. The wide range of the scaler-timer also effectively allows preset time or preset count operation for most counting situations. For preset time the scaler selector switches are simply set to maximum. For preset count the timer selector switches are set to maximum.

When in the preset time/preset count mode (automatic operation) Model 8710 is fully coordinated with the operation of the automatic sample changer and the printing lister. All necessary start, stop, reset, print, and sample reject commands are exchanged in proper sequence.

Model 8710 is equipped with an automatic printing lister that furnishes a permanent record of sample data on easily replaceable paper tape. The lister supplied with these models is capable of printing sample index number, counting time, and sample count in that order. The sample number, time and count are keyed to their respective positions on the tape, and are always recorded in proper sequence.

The 8710 will supply all necessary commands, in proper sequence for fully coordinated automatic operation of the changer. For manual sample changing, the printing lister

operation can be manually controlled at the scaler. The scaler also provides the sample number indexing facilities--from 1 to 100 samples in sequence--for the automatic sample changer. These index numbers are keyed to the index number print-out of the lister for positive sample identification throughout the sample-changing cycle.

Appendix 9.39b

Determination of Beta Emitters in Perimeter Samples

Appendix 9.39b

Determination of Beta Emitters in Perimeter Samples

The concentration of beta emitters in perimeter samples is determined as follows:

$$\mu c/ml = \frac{(\text{Net Count})\ (\text{Factor})}{M^3}$$

Net Count	=	Total count for 60 minutes less 60-minute background count.
M^3	=	Total cubic meters of air sampled
Factor	=	$\frac{1}{60\ c\ g\ a\ K_1\ K_2}$
60		Converts counts per 60 minutes to counts per minute
c	=	Collection efficiency = 98%
g	=	Geometry of counter = 50%
a	=	Absorption correction (not applicable, see Paragraph 9.34)
K_1	=	$d/(m)/(\mu c) = 2.22 \times 10^6$
K_2	=	Milliliters per cubic meter = 10^6
Factor	=	$\frac{1}{(60)\ (.98)\ (.50)\ (2.22 \times 10^6)\ (10^6)}$
Factor	=	1.5×10^{-14}

$$\mu c/ml = \frac{(\text{Net Count})\ (1.5 \times 10^{-14})}{M^3}$$

Since the counting error for a 60-minute count at 95% confidence level is ± 10% at 6.5 net counts per minute, the minimum detectable concentration of beta emitters on a weekly sample is:

$$\mu c/ml = \frac{(390)\ (1.5 \times 10^{-14})}{604.8}$$

$$\mu c/ml = 9.9 \times 10^{-15}$$

With ± 10% accuracy.

Appendix 9.39c

Determination of Alpha Emitters in Perimeter Samples

Appendix 9.39c

Determination of Alpha Emitters in Perimeter Samples

The concentration of alpha emitters is determined as follows:

$$\mu c/ml = \frac{(\text{Net Count})\ (\text{Factor})}{M^3}$$

Net Count = Total count for 60 minutes less 60-minute background count.

M^3 = Total cubic meters of air sampled

$$\text{Factor} = \frac{1}{60\ c\ g\ a\ K_1\ K_2}$$

60 Converts counts per 60 minutes to counts per minute.

c = Collection efficiency - 98%

g = Geometry of counter = 35%

a = Absorption correction - 70%

K_1 = d/(m)/(μc) = 2.22×10^6

K_2 = Milliliters per cubic meters = 10^6

$$\text{Factor} = \frac{1}{(60)\ (.98)\ (.35)\ (.70)\ (2.22 \times 10^6)\ (10^6)}$$

Factor = 3.1×10^{-14}

$$\mu c/ml = \frac{(\text{Net Count})\ (3.1 \times 10^{-14})}{M^3}$$

Since the counting error for a 60-minute count at 95% confidence level is ± 10% at 6.5 net counts per minute, the minimum detectable concentration of alpha emitters on a weekly sample is:

$$\mu c/ml = \frac{(390)\ (3.1 \times 10^{-14})}{604.8}$$

$$\mu c/ml = 2.0 \times 10^{-14}$$

With ± 10% accuracy

Appendix 9.40

Determination of Radioiodine

Appendix 9.40

Determination of Radioiodine

The concentration of radioiodine in stack and perimeter samples is determined as follows:

$$\mu c/ml = \frac{c/m \text{ (Factor)}}{M^3}$$

c/m = Net counts per minute on filter

M^3 = Total cubic meters samples

$$\text{Factor} = \frac{1}{c\ g\ a\ K_1\ K_2}$$

c = Collection efficiency = 97%

g = Geometry of counter = 50%

a = Absorption correction = (not applicable - see paragraph 9.34)

$K_1 = d/(m)/(\mu c) = 2.22 \times 10^6$

K_2 = Milliliters per cubic meter = 10^6

$$\text{Factor} = \frac{1}{(.97)\ (.50)\ (2.22 \times 10^6)\ (10^6)}$$

$$\text{Factor} = 9.3 \times 10^{-13}$$

The length of time the sample is counted depends on the activity present. The stack sampler is operated at about 300 l/m or 144 M^3/8 hours. For a one-minute count, the minimum detectable I-131 with $\pm$ 10% accuracy is:

$$\frac{(400)\ (9.3 \times 10^{-13})}{144} = 2.6 \times 10^{-12}\ \mu c/ml$$

Appendix 9.43

Exposure Record Card

Appendix 9.43

Exposure Record Card

Exposed From - To	Dosimeter Readings								Badge Readings				Total
	M	T	W	T	F	S	S	Total	G	B	N	Total	

Recorded For	Body Status	Prev. Total	This Card	Acc. Dose	5 (n-18)	Unused Dose

Badge	Name (Last, First, Middle)	S. S. Number	Birth Date	Nuclear Fuel Services, Inc. West Valley, New York

Appendix 9.47

Routine Survey Form

Appendix 9.47

Routine Survey Form

ROUTINE SURVEY LOG
NUCLEAR FUEL SERVICES, INC.
SPENT FUEL REPROCESSING PLANT

Survey Number: Shift Assigned:

Frequency: Time Allotted:

Title:

MATERIALS AND EQUIPMENT REQUIRED:

DESCRIPTION OF SURVEY:

SPECIAL SAFETY INSTRUCTIONS:

PREPARED BY: REVISED BY: DATE:

Appendix 9.49

Environmental Monitoring Program

Appendix 9.49

Environmental Monitoring Program

Phase I - Atmospheric Monitoring

Three air-sampling stations have been established at the site perimeter. These stations consist of a vacuum pump drawing air through a filter, a beta-gamma detector to measure the activity deposited on the filter, a log-count ratemeter and strip-chart recorder to provide a permanent record of activity at each location. These air monitors are serviced weekly. One air-sampling station has been established near the plant site. This station will consist of a vacuum pump pulling air through a filter paper to collect particulates. The filter will be changed and monitored weekly for gross alpha and gross beta activity.

Water Monitoring

A rain and snow collection station has been established at the plant site. Samples of water from this station are collected and monitored as available for gross alpha and gross beta activity, iodine-131, and strontium-90. Surface water samples and mud and silt samples are collected monthly and monitored for gross alpha and gross beta-gamma activity. Samples will be collected from the following locations:

1. Erdman Brook near Buttermilk Creek;
2. Buttermilk Creek near the Emerson Road crossing;
3. Cattaraugus Creek near the Nagel Road crossing.

A well water sample is obtained from the site monthly and monitored for gross alpha and gross beta-gamma activity.

Earth and Biota Monitoring

Vegetation samples are collected near the three perimeter monitoring stations in the spring and fall and will be monitored for gross alpha, gross beta-gamma, iodine-131, and strontium-90.

A milk sample is collected from a neighboring farm weekly and is monitored for gross beta-gamma. Samples collected once each month are analyzed for iodine-131 and strontium-90.

In the spring and fall a rabbit or other small game from the site will be analyzed for gross beta, gross alpha and iodine-131 activity.

Phase II - Atmospheric Monitoring

The three air-monitoring stations at the site perimeter, as described in Phase I, will be used and serviced weekly.

Appendix 9.49

The Plant site air-sampling station, as described in Phase I, will be used and serviced weekly.

Samples may be autoradiographed on x-ray film to determine number and relative intensity of particulates collected.

Water Monitoring

A rain and snow collection station is established near the plant site. Samples of water from this station will be collected and monitored as available for gross alpha, gross beta and tritium. A strontium-90 determination will be made twice each year.

Surface water, mud, and silt samples will be collected monthly from the locations specified in Phase I. These samples will be monitored for gross alpha, gross beta-gamma, and tritium. Each month one or more of the samples will be gamma scanned if sufficient activity is present. Well water samples will be collected monthly from the site. These samples will be analyzed for gross alpha, gross beta-gamma, and tritium activity.

Earth and Biota Monitoring

Vegetation samples will be collected in the spring and fall near the three perimeter stations. These samples will be analyzed for gross alpha, gross beta-gamma, and iodine-131. One or more of the samples collected in the spring and fall will be analyzed for strontium-90.

Milk samples will be collected weekly from the plant site and analyzed for gross alpha, gross beta-gamma, and iodine-131. In the spring and fall strontium-90 determinations will be made.

Twice each year, in the spring and fall, fish and shellfish specimens will be collected from Buttermilk Creek and Cattaraugus Creek. These specimens will be analyzed for gross alpha, gross beta-gamma, and iodine-131.

In the spring and fall a rabbit or other small game from the site will be analyzed for gross alpha, gross beta-gamma, and iodine-131 activity.

Appendix 9.51

Stack Monitoring System

Appendix 9.51

Stack Monitoring System

The stack monitoring system consists of two channels of monitoring; the first channel for beta-gamma emitting particulates, and the second channel for iodine-131 with readout and alarm locally and at the Control Room Panel. The system is supplied by Tracerlab, a division of Laboratory for Electronics, Inc., Richmond, California. The following equipment is included:

1-Isokinetic nozzle;

1-Model MX-14C Pumping System with a 10 cfm sliding dry vane pump driven by an electric motor through double V-belts, a control valve, a calibrated fixed orifice, flow gauge, and necessary piping;

1-Model MA-1B Filter Tape Transport Mechanism, an advanced version of the Brookhaven design. A solid capstan with milled slots rides on a Teflon shear valve which limits the air bypass around the filter paper to less than 2 percent. The filter paper is held against the rotating capstan by the pressure differential across the paper and is moved by the rotating capstan. Two filter tape speeds are provided; one inch per hour for normal operation, and 28 inches per minute for fast advance to clear contaminated tape from the detector areas;

1-Model MD-1B Beta-Gamma GM Detector, a halogen-quenched end window detector, $2\frac{1}{2}$ inches in diameter. The window is mica -- less than 4 mg/cm^2 thickness. The detector is shielded by two inches of lead;

1-Model RM-20B Precision Log Ratemeter with a $4\frac{1}{2}$-inch panel meter indicating the counting rate directly in counts per minute on a switch selected three decade (10 to 10^4 cpm) or five decade (10 to 10^6 cpm) scale. One additional scale indicates high voltage. The ratemeter has an adjustable alarm point and a manual reset to provide "memory". The meter relay automatically resets to permit meter to read cpm below the alarm point;

1-Model RM-40B Dual Power Supply with main power switch, high voltage switch, two high voltage adjustment screws and alarm reset. Unit supplies high voltage for the system detectors;

1-Model MI-5B Iodine Sampler and Shield Assembly, with a holder for a two-inch diameter activated charcoal filter and a three-inch thick lead shield for the detector;

1-Model MD-5B Gamma Scintillation Detector, with a $1\frac{1}{2}$-inch diameter x 1-inch sodium iodide, thallium-activated crystal, a Dumont 6292 photomultiplier and a 100-gain preamplifier.

Appendix 9.51

1-Model RM-20BS Precision Log Ratemeter, identical to the RM-20B described above plus a spectrometer input circuit for pulse height discrimination. Window width adjustable from 2 percent to 100 percent and threshold adjustable from 5 percent to 100 percent of full scale.

1-De Var Series R 300 two-pen recorder, Control Room Panel mounted to record output from particulate and radioiodine stack monitors and to alarm if count rate of either unit exceeds the preset level.

The beta-gamma particulate monitor, with two inches of lead shielding around the detector, has a background of about 25 cpm. A counting rate of twice background is obtained at a concentration in the stack of about 1×10^{-11} $\mu c/cc$ of mixed fission products.

The Gamma Scintillation Detector, with three inches of lead shielding around the detector, has a background of about 175 cpm for a counting threshold of 100 kev. The detector will show 100 net cpm after an exposure of 30 minutes to a concentration of 10^{-10} $\mu c/cc$ of iodine-131.

Calibration and Maintenance of Stack Monitor

The response of the stack particulate monitor to radioactivity will be apparent from natural activity as well as activity from the stack air stream which will be trapped by the filter paper. Calibration of the particulate monitor, relating detector counts per minute to microcuries per milliliter in the air stream, will be accomplished by analyzing a section of the filter in the counting room and comparing the results to the count rate indicated by the stack sampler. After the initial calibration, this check will be made only as the need is indicated but not less than twice annually.

The iodine monitor will be calibrated quarterly by analyzing the activated charcoal filter in the counting room and comparing the results with the count rate indicated by the stack sampler. The collection efficiency of the charcoal filter will be checked by collecting a caustic scrubber sample independent of the stack monitor and comparing the results with the activity of the charcoal filter. This check will be made on each batch of filter paper received.

Appendix 9.53

Weather Monitoring Station

Appendix 9.53

Weather Monitoring Station

Two weather-monitoring stations are provided, one at 60 feet and the second at 200 feet above ground level. Each station continuously records wind direction, velocity, and ambient temperature. The weather-monitoring stations are supplied by Science Associates, Princeton, New Jersey and each station consists of the following:

1-No. 4-120 Aerovane transmitter, a combined anemometer and wind vane in one unit. A three-bladed plastic rotor with a starting speed of 2.5 mph drives a magneto which generates a voltage directly proportional to wind speed. The streamlined vane houses a synchro whose rotor position is determined by the vane. The transmitter includes a filter to prevent radio interference and is permanently lubricated.

1-No. 4-141-5 Aerovane Recorder provides instantaneous readout of wind direction and velocity on the same chart. Direction data from 0 to 360 degrees is recorded on one side of the chart and speed data from 0 to 100 mph is recorded on the other side of the chart. Each recording area is 4½ inches wide.

1-No. 170 Stainless Steel sheathed temperature bulb and temperature recorder. Recorder is two pen with range of-50 to 110 F in one degree divisions. The temperatures from both stations record on one chart.

1-No. 174 Aspirated Solar Radiation Shield to house the temperature bulb. Heat from the sun and from surrounding objects is excluded by the dome-shaped shield, by an inner and outer shield and by a surface oriented baffle. A motor and blower, located at the remote end of the mounting arm, induce a forced ventilation.

The recording charts for the two weather stations are located in the Control Room Panel.

Appendix 9.56

Stream Gauging and Sampling

Appendix 9.56

Stream Gauging and Sampling

Stations are provided for gauging and sampling the flow of Franks Creek and Cattaraugus Creek.

The gauging stations each consist of a calibrated level detector and continuous recorder.

The sampling stations each consist of a proportioning pump to take a maximum size sample of 10 gallons per week. The samplers are each housed in an electrically heated, weatherproof enclosure.

Appendix 9.64

Fire Brigade Organization

Appendix 9.64

Fire Brigade Organization

Purpose

It is the purpose of the shift fire brigades to serve as the primary fire fighting organization for the West Valley Plant.

General Organization

The Health and Safety Director is responsible for organizing and maintaining the shift fire brigades and will serve a fire chief for the plant. His alternates will be the Assistant Production Manager and the Production Manager. The Health and Safety Director, or his designated representative, will schedule and direct fire drills and classes in fire fighting to maintain a high degree of proficiency in the fire brigades.

The four Operating Shift Supervisors are assigned as fire captains and, as such, are to direct fire fighting activities in the absence of the Health and Safety Director. The Shift Fire Brigades shall be as follows:

a. Operations Shift Supervisor;

b. Health and Safety Technician;

c. Chemical Make-Up Operator;

d. Sampling Operator;

e. General Purpose Cell Operator;

f. Yard Maintenance Operator (2).

Procedure

When a fire is discovered, the fire alarm shall be sounded at once, before any attempt at extinguishing the fire is made by personnel in the immediate area. Personnel reporting fires to the Shift Supervisor shall give specific information relating to the location and nature of the fire. If the fire involves radioactive material, the area is to be evacuated at once and, if possible, all doors leading to the area should be closed. Entry to such an area will require full protective clothing and self-contained breathing apparatus. The fire brigade members will report to the control room for instruction and will proceed to the scene of the fire only after receiving such instruction. The Health and Safety technician will perform radiation monitoring for the fire brigade. Generally, the first member to reach the scene will attempt to control the fire with portable fire extinguishers while the later arrivals will lay hose from the nearest hose stations. The hoses will be made ready even if the fire appears small and easily controllable. This general statement of procedure is, of course, subject to change depending on the location and nature of the fire. In some cases it may be advisable to let the fire burn itself out and in other cases the use of

portable fire extinguishers may appear to be useless or hazardous. The evaluation of the fire and the determination of the best method for fighting the fire is the responsibility of the senior fire brigade officer in the area. In addition to the above, the Health and Safety Director will work cooperatively with local firefighting agencies.

Notification

The following persons shall be notified immediately when a fire is discovered:

a. Health and Safety Director;

b. Medical Director;

c. Plant Manager;

d. Production Manager;

e. Assistant Plant Manager;

f. Plant Engineer;

g. Assistant to the Plant Manager;

h. Security Officer.

Reports

It will be the responsibility of the Operations Shift Supervisor to prepare a full report of the incident for internal distribution. The Health and Safety Director will be responsible for reporting the incident to local, state, or federal authorities and insurance companies as required.

Public Relations

In the event of a fire or other serious accident, all public statements concerning the incident will be made by the Plant Manager or his duly authorized representative. Employees shall not make such statements and must refer all questions concerning the accident to the Plant Manager. In no case will persons not employed by the company be permitted to enter the plant area to view the scene of the accident or to question plant employees unless they are so authorized by the Plant Manager or his representative.

Appendix 9.64

Fire Detection and Extinguishing Equipment

Yard Fire Protection System

The supply of firewater will be from the filtered water storage tank with 300,000 gallons reserved for fire fighting. One of two 1000 gpm pumps, one electric motor driven and the second a diesel powered pump for emergency use, will supply firewater at 70 psig to the firewater loop. Four hydrants with hose houses are spaced around an 8-inch diameter firewater loop and one is placed near the warehouse. Each hose house contains 100 feet of hose, fog nozzle, and tools. Hydrants are 6-inch, UL-listed, frostproof with two 2½-inch valved outlets.

Sprinkler Systems

Dry sprinkler systems installed per National Board of Fire Underwriters Standard No. 13 are provided for the warehouse and cooling tower. The Chemical Process Cell and the Extraction Cells are equipped with spray-down systems using fog nozzles. While the primary purpose of these systems is decontamination, they would also prove effective for fighting in-cell fires.

Dry Chemical Systems

The fire-fighting systems for the Process Mechanical Cell and the General Purpose Cell are dry chemical systems because of the possibility of metal fires. Fixed spray nozzles are directed at the shear in the PMC and at the fuel basket loading station and fuel basket storage area in the GPC. Hose stations, to be handled by manipulators, are installed to reach all cell areas. The extinguishing agent will be "METL-X" powder powered by commercial argon. The "METL-X" canisters and argon bottles are located outside the cell. An argon purge line is tied into the "METL-X" header so that fire lines can be blown down after test or after emergency use. Alarm systems are provided to indicate a fire in an unattended cell.

Wet Standpipe System

A wet standpipe system is provided in the occupied areas of the Process Building. About every 75 feet in these areas a hose rack containing 75 feet of hose and fog nozzle is located.

Fire Extinguishers

Fire extinguishers are provided as shown below:

Appendix 9.64

Area	Pressurized Water	Foam	CO_2	Enclosure I	Enclosure II	Enclosure III
Process Building Office	4		2	4	1	1
Laboratory	1		8	9		
Utility Building		2				2
Warehouse	2		2			4
Cold Chemical		1				1
Control Room	2		3	5		
Gate House			1	1		
Maintenance Shop		1		1		

Enclosure Type I - Elkhart Model C-950

Enclosure Type II - Elkhart Model A-950

Enclosure Type III - Wood, painted red, type of extinguisher stenciled on door

Maintenance of Fire Protection Equipment

Fire protection equipment will be checked and tested in accordance with procedures recommended in applicable NBFU Standards.

Appendix 9.93

Format for Standard Operating Procedures
and
General Index of Standard Operating Procedures

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Export Controlled Information

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Export Controlled Information

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Before The

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.
For Construction Permit and Licenses for
a Spent Fuel Processing Plant
Under Sections 53, 63, 81, 104 (b), and 185
of the Atomic Energy Act

AEC Docket No. 50-201

Part B -- Safety Analysis

Amendment No. I

October 12, 1962

Part B -- Safety Analysis
Amendment No. I
October 12, 1962

On July 26, 1962, Nuclear Fuel Services filed with the Division of Licensing and Regulation an application to build and operate a fuel reprocessing plant for spent reactor fuel. A two-volume Safety Analysis was submitted as Part B of this application. During the past two months there have been several design changes. These will be reflected in a series of amended pages to the Safety Analysis. The amended pages will be submitted as soon as they can be prepared and the required copies printed. These changes are summarized as follows:

1) Elimination of Thorex Equipment

The expected load of ThO_2-UO_2 fuel is less than originally contemplated so that it does not now appear appropriate to include the expensive facilities needed to provide a processing capability for this type of fuel equivalent to that of the UO_2 fuels. Therefore, the plant capacity for ThO_2-UO_2 fuels will be 500 kg/day in place of the 1000 kg/day described in the Safety Analysis. Also, the facilities for decontamination of the recovered thorium have been eliminated. Thorium will be permitted to go into the high level waste stream and will be stored in stainless steel tankage along with the fission products from this particular fuel. Stainless steel tankage will be provided for this purpose as required.

2) Reduced Capacity for Stainless Steel-Cermet Fuels

The Safety Analysis indicates the inclusion of a Darex facility capable of handling 225 kg/day of stainless steel. The capacity of this unit has been reduced to 125 kg/day and the capital allowances include the cost of this capability. If development work on electrolytic dissolution at SRP continues to look favorable prior to freezing of design, NFS may change the design to provide for electrolytic dissolution at a processing rate of 125 kg/day

of stainless steel. Allowance has been made in cost estimates to permit the inclusion of either (not both) sets of equipment.

3) Waste Storage Facilities

Facilities for storing neutralized wastes which will be installed at the outset will include two 750,000-gallon carbon steel and concrete tanks. One of the tanks to be installed will be used for storage of the neutralized wastes from the entire processing sequence. The second will be held as a spare. Other waste storage shown in the Safety Analysis will not be built at the outset. Stainless steel tankage will be provided by NFS as required, but funds will not be committed until receipt of firm commitments for the processing of fuels whose wastes require this type of storage (e.g., stainless steel-UO_2 cermets, ThO_2-UO_2, depleted uranium-molybdenum, uranium-aluminum, or uranium-zirconium alloys). The depreciation schedules for waste storage are sufficient to provide a revolving fund for new mild steel waste storage as required and contractual commitments will permit additional capital as needed.

4) Removal of One Dissolver

As a result of the reduced requirements on the plant for the ThO_2-UO_2 fuels, one of the three dissolvers was removed.

The remaining plant facilities including fuel receiving and storage, mechanical cell, extraction equipment, acid recovery, solvent handling, prod-

uct packaging and handling, utilities, maintenance, and analytical facilities remain essentially unchanged as described in the Safety Analysis.

These changes have already been communicated to the USAEC in a letter to Mr. R. C. Blair, dated September 21, 1962.

By letter dated September 5, 1962, the Division of Licensing and Regulation of the USAEC raised eleven questions formal answers to which are hereby submitted.

1) Question: The application does not give the types and maximum quantities of radioactive material that will be present in each processing step and in the storage areas during normal operation of the plant. This information is necessary to determine the probable releases that could be anticipated in the event of an accident.

Answer: In Table A-I-1 there is given a tabulation of the quantities of radioactive materials expected in each process stream. Data are given for fissionable materials, fertile materials, total fission products and specific fission products. The data presented are representative of the most radioactive fuel which we expect to process in the plant. These data are for a fuel burned to 20,000 mwd/ton, 27.5 mw/ton, irradiated two years at 85% load factor and cooled 150 days.

The maximum quantity of fuel which can be stored in the storage pool is about 1000 fuel elements. It is expected that normally no more than 25% of the pool will be full. The amount of activity stored therein can only be estimated since it will depend upon the past history of the particular fuels delivered to the plant. An estimate of the amount of fuel normally in storage is given in Table A-I-2.

The amount and type of activity in waste storage will change with time, increasing as the tank is filled and at the same time decreasing due

to decay. The maximum amount of activity in storage in a given tank will be present just at the completion of the filling period. An estimate of the total quantity of waste and the major specific fission product contributors at the time the first waste tank is filled is given in Table A-I-3.

2) Question: It does not appear that a "mock-up" shop will be included in the plant. Existing Commission facilities which utilize "remote maintenance" have "mock-up" shops and these shops have reduced exposure considerably during maintenance. How does NFS plan to minimize employee exposure during maintenance operations without such a shop?

Answer: The NFS plant does not have a full-scale "mock-up" shop complete with crane of the same type as that used at Hanford and Savannah River. However, the plant does have the capability for carrying out the function of the "mock-up" shop, viz, to check out the exact dimensions of a piece of equipment to be installed in the chemical processing cell. Jigs are provided in the maintenance shop to set up an equipment piece going into the CPC and to check all of the pertinent dimensions against the known requirements.

The need for this type of facility is less in the NFS plant than in the AEC production plants since there are only 15 pieces of equipment in the CPC which are handled in this manner and the proposed procedures will adequately allow the checking of such a small number of equipment pieces.

Further, it is felt that this is not a safety problem at all but rather an economic one. The installation of equipment in the CPC is done remotely and does not involve the exposure of personnel in excess of the normal plant background. The penalty for making an error in the measurement of dimensions (resulting in failure to be able to install the equipment) is not taken in increased personnel radiation exposure but rather in lost production

time. If it were not possible to install the equipment piece, it would be necessary to move it out into the decontamination area, clean it up to the point where it could be worked on again, and make the necessary corrections.

3) Question: The chopping technique for the variety of fuels that will be processed has not been employed before in spent fuel processing plants and the probability of widespread contamination in the cell, as anticipated by NFS, does necessitate "remote maintenance." How is this remote maintenance going to be accomplished on large items such as cell windows and the "push-out ram"?

Answer: The chopping technique has not been in routine plant operation, but it has been operated on full plant scale at Oak Ridge National Laboratory by the Chemical Technology Division. The amount of "dusting" is estimated to be about 1 per cent. To confine this and any fines produced, the chopper will be separately sealed and washed down at the end of a cycle. Nonetheless, the cell is expected to be highly contaminated and the specifications for every equipment piece installed in the cell require that it be capable of remote maintenance for most repairs, and that it be removable either in pieces or in toto if that is required. The windows are removable from the outside of the cell.

Since the push-out ram was questioned specifically, a step-by-step method for its removal is given below:

1) The drive system, control, and power connections are located outside the cell and are not contaminated. These are first removed.

2) All the supporting connections outside the cell are then removed after which the supporting connections inside the cell are removed or loosened by use of the manipulator.

3) A wall section of the operating aisle opposite to the PMC must be removed to allow complete removal of the unit.

4) Around the ram drive mechanism there is some unit shielding. This is removed from outside the cell. A partial radiation check can then be made of the unit. Decontamination of the unit will undoubtedly be required. This is done using the remote manipulators inside the cell. Additional radiation checks are made during the course of the decontamination.

5) Roller units are set up in the operating aisle to allow the unit to be brought out horizontally. A special dolly is brought up outside the building to support the unit as it is being removed and to carry it to the maintenance shop.

6) A sling is attached to the crane inside the cell and the sling is secured to the end of the ram housing. The housing is raised sufficiently by the crane to take the weight off the supports.

7) The rollers are also raised until the ram housing no longer contacts the supports. The ram unit is now free to move.

8) A cable is attached to the ram in the operating aisle, threaded through the opening in the outside wall and attached to a winch located outside so that a straight line pull on the ram unit is possible.

9) The winch is actuated and the ram slowly pulled out of the cell. The crane inside the cell must be moved in parallel with the ram movement. Constant radiation monitoring is carried out during this operation to ascertain that the decontamination has been sufficient. It may be necessary to stop and do additional decontamination. Additional decontamination may be done either in the operating aisle, with or without the use of portable shielding, or it may be done inside the cell by partially reversing the removal operation.

10) As the unit emerges it is placed on the dolly and then transported

to the maintenance shop. The opening in the PMC wall is temporarily shielded.

4) Question: In paragraph 4.11 the return of the fuel transfer basket from the Process Mechanical Cell (PMC) to the Fuel Storage Pool Complex is described. The transfer of contamination to the fuel pool is not discussed during this step. What evaluation has been made to determine the quantity of contamination that will be transferred to the pool? How well will the ion exchange equipment (see paragraph 5.4) remove the contamination? Since only a single circuit of cooler, filter and ion exchange equipment (see Figure 5.4) is going to be used, how will cooling be accomplished throughout the pool while decontamination of only one compartment of the pool (see paragraph 5.5) is in progress?

Answer: Revaluation of the contamination problem has led NFS to decide on a slightly different approach to the problem of transferring the fuel elements into the PMC from the FRS. The fuel storage baskets will be sized so that the elements will protrude a short distance above the top of the basket. The basket will be affixed to the underwater transfer conveyor as indicated in paragraph 4.8 and brought into position below the hatch. The crane will then be used to pick up the fuel element from the basket leaving the basket in place in the underwater transfer conveyor. The basket is returned to the fuel storage pool never having gone into the PMC where it might become contaminated. Paragraphs 4.8 and 4.10 are being rewritten to reflect these changes and new pages will be submitted.

It would appear that the remainder of the questions in this section were prompted by the contamination possibility created by the return to the pool of baskets which had been in the PMC and, therefore, they may be considered to be answered by the above change. There is, of course, some

chance of pool water contamination from other sources such as the storage of the elements. This is not a problem which is unique to this plant, however, and the proposed method for handling the cooling and the decontamination of the pool water has proved effective in many previous installations. There is no justification for duplicating the ion exchange equipment. No contamination problem in a fuel pool is so acute or so sudden that the cleanup facility has to be in constant operation. The same is true of the cooling system. The fuel pool is a tremendous heat sink. At the design rate of heat release, 600,000 Btu/hr--a number which we believe to be conservative, equivalent to the heat from the 200-day-cooled output of 6000 mwt of reactors--the temperature of the fuel pool would increase only 0.1 F per hour. Thus it does not appear to be necessary to provide a spare cooling loop.

5) Question: In reference to the proposed ventilation system what evaluations indicate that (1) a manifolded and butterfly valve controlled system on the exhaust of the blowers (see Figure 6.3) will prevent blow back in the event of a blower failure and (2) that a "minimum" 100 fpm face velocity thru openings will prevent backmixing from active to less active areas. (see paragraph 6.9)

Answer: 1) A detailed answer is being prepared and will be submitted.

2) The 100 ft/min figure has been generally used throughout AEC installations for design purposes.

6) Question: According to Table 2.11a and 2.11c freezing conditions and snowfall can be expected from October to April. Under these conditions, will operational and contamination problems occur in outdoor areas, such as the washdown area (see paragraph 3.6 and 4.3), the general purpose evaporator (see paragraph 4.97 and 5.43), the concrete lined burial bins for high level solid waste (see paragraph 4.98), the low level trash burial area (see paragraph 4.98), the water seals on the thorium and depleted uranium storage

tanks (see Figure 4.96) and the diversion box and outdoor sampling points (see Figure 6.23e)?

Answer: (1) Operational problems can be expected to occur in an outdoor washdown area. Therefore, this concept has been abandoned. All washing and decontamination will now be done inside the building.

(2) The general purpose evaporator has also been moved into the building.

(3) It is no longer planned to use the concrete-lined burial vaults. These were for the storage of metal hulls which have been leached in boiling nitric acid and thoroughly rinsed with water. Activity which is not removed by this treatment is not likely to be removed by water at a pH of near neutrality. We propose to bury these directly in the silty till, a formation which is not a aquifer and in which the rate of movement of any water that should get into it is essentially zero.

(4) Low level trash burial is an operation which can be halted in extremely inclement weather. Burial ditches will be dug somewhat ahead of time and sloped to one end so that rainwater collecting in the open trench will collect away from the point at which burial is taking place. If water does collect in the end of the trench, it will either be pumped out prior to the use of the end of the trench or the end will be back-filled without making use of it for low level burial.

(5) The seals called for are liquid seals. During winter months, the liquid will be of a non-freezing type.

(6) The diversion box function has also been taken back into the building.

(7) The only outdoor sampling points are on the waste tanks themselves and into the annular space around the tanks. These will be a

simple thief-type sampler which will operate in inclement weather.

Appropriate changes have been made in the Safety Analysis to reflect the above and the revised pages will be submitted.

7) Question: Where and in what quantity will cell penetrations be provided for the future anticipated requirements as indicated by Figures 3.19(a), 3.19(b), 3.22, 4.21(a), 4.33(c), and 4.39(a)?

Answer: There are provided in the contact cells 10% spares for all penetrations and there are 20% spares in the remote cells. These penetrations take the form of stainless steel pipe or tubing so arranged that there is no leakage of radiation in excess of the design shielding for the wall in which they are installed. Both ends are sealed by welding. Consequently, they represent no safety hazard. Their exact locations will be shown on the final drawings.

8) Question: Clothing, monitoring and change facilities are discussed in paragraphs 8.14 to 8.17 but this discussion does not define the boundaries of the controlled zones in the plant, the traffic pattern in the controlled zones or the type and size of facilities provided at each boundary to prevent carry over of contamination from one zone to the next. A discussion of these points for the proposed plant layout will be necessary.

Answer: Drawings are being prepared, coded to show the five kinds of plant areas from a contamination control standpoint. These are:

a) unrestricted access

b) access when wearing plant clothing and shoes

c) access when wearing plant clothing, shoes, and special shoe covers

d) no access at all except after thorough decontamination, health physics surveys, special clothing, and shoe change

e) a few limited areas in which either (a) or (b) is permitted. Persons will enter the plant only through the main entrance.

They will have free access to the (a) areas without changing clothing or shoes and may also go into (e) areas. In the case of (e) areas visitors will not be permitted unless accompanied by plant personnel. The workers will change clothes and shoes in the locker rooms after which they will have access to the (b) areas. At the interface between all (b) and (c) areas there will be shoe cover racks and there will be a change of footgear at every crossing of these interfaces. The (d) areas will not be entered at all except under full health safety coverage and there will be clothes and shoe change areas set up at the point of entry.

9) Question: What are the average and maximum discharge concentrations and flow rates of I_{129} and I_{131} that will be exhausted through the stack? What total quantity of these materials will be released per year?

Answer: In view of the implications of question 11 (see below) the requested data are provided not only for the iodines but also for krypton 85. In Table A-I-4 there are presented data representative of the discharges which are expected from average fuel during the first few years of operation.

The discharges of these same isotopes from a fuel representative of the highest burnups which we contemplate processing in the NFS plant were shown as a part of Table A-I-1 (see answer to question 1) and are repeated here for convenience:

Kr	85	8.25×10^3 curies
I	129	0.017 curies
I	131	1.7 curies
Xe	131m	1.1 curies
Xe	133	5×10^{-3} curies

It had been our intention to discuss this type of fuel when the time came to develop technical specifications. In view of the interest shown by the ACRS at the meeting held on October 5, 1962, we have decided to redo the calculations of Sections VII and VIII using the above maximum numbers. Revised pages for these sections will be submitted as soon as possible.

10) Question: It is suggested in paragraph 7.12 that the storage lagoon will be used as an emergency holdup area for the overheads from the general purpose evaporator. What maximum concentration of activity in these overheads would be discharged to the storage lagoon? Do any other streams feed this lagoon? If so, what type and concentrations of activities will be in these streams? How operable will the lagoon be during winter weather conditions?

Answer: There appears to be some misunderstanding concerning the function of the storage lagoon. It is not our intention to operate this as a seepage basin. That is, it is not the intention to routinely percolate wastes out into the stream through the ground using the ion exchange capacity thereof for additional decontamination.

All the liquid discharges will pass through the storage lagoon. They will be monitored and a record kept of the volume and activity which has been discarded. We expect to discard liquid at this point such that 10 CFR Part 20 will be met in Buttermilk Creek on a gross count basis assuming the absence of radium (1×10^{-7} uc/cc). The average dilution factor available in Buttermilk Creek has been calculated to be 2.7×10^3. Using a dilution of 10^3 would imply that the discharge from the storage lagoon could be as high as 10^{-4} uc/cc.

If the storage lagoon discharge were to become higher than that required to meet 10 CFR Part 20 in Buttermilk Creek the overflow would be

stopped and the discharge held up until the condition causing the higher activity level had been corrected or until specific fission product analyses could show that the waste could be satisfactorily dischared within the limits of 10 CFR Part 20. It is then the function of the lagoon to hold up the activity. In dry warm weather some seepage into the ground can be expected and for the amount which does so seep advantage would be taken of the ion exchange capacity of the ground. This would be an abnormal and not a routine operation, however. In cold weather it is conceivable that the volume held up would freeze. Since the ice could be expected to remain in place the storage lagoon could then be said to be carrying out its function very well.

The process streams which go to the storage lagoon and their expected activity levels are as follows:

1) Overheads from acid fractionation----10^{-5} uc/cc

2) Overheads from General Purpose Evap----10^{-6} uc/cc

3) Floor drains from non-contaminated areas of the plant----------zero to 10^{-6} uc/cc

Miscellaneous wastes such as laundry wastes and laboratory wastes can go to the storage lagoon but they do so by way of the General Purpose Evap Feed Tank. If they prove to be low enough so that evaporation is not necessary they can be then routed to the storage lagoon. We expect to set an operating limit for any waste discharged to the storage lagoon at about 10^{-2} uc/cc. *

We expect not to discharge anything from the storage lagoon so that Buttermilk Creek could become in excess of 10 CFR Part 20.

* This is approximately the limit used at Savannah River also.

11) Question: Since significant quantities of Kr 85 and I_{131} (see paragraph 7.6 and Table 7.7) will be discharged to the atmosphere, with what degree of certainty can we be assured that the generalized parameters used are consistent with the actual site metereology?

Answer: Before operation has started we expect to have collected on-site meteorological data for at least a year and to have some evidential support for the meteorological parameters used. In the meantime the data used in making all calculations have been deduced by Dr. Maynard Smith from a study of the site. He was asked to supply a "conservative" set of data. At a meeting of the ACRS subcommittee held in Buffalo on September 12, 1962, Dr. Smith defined his concept of the degree of conservativism as follows: The parameters were selected to reflect conditions which are perhaps as much as three times better than the worst possible condition and about 1000 times worse than the best conditions.

Respectfully submitted,

Nuclear Fuel Services, Inc.

By T Chinn
President

Subscribed and sworn to before me this 16 day of October 1962. Christine A. Kotarski

My Commission expires 8/1/67

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Table A-I-2

Storage of Activity in Fuel Pool[a]

	Maximum Possible Loading	More Probable Loading
Total Elements	1000	500
Total Tons	250	125
Average Cooling, Days	150	200
Total Activity,	5.5×10^8	2×10^8

[a] Assumptions:

1) 4 elements/ton

2) No element is received with less than 100 days cooling.

3) Elements can be received at the rate of 60/week.

Table A-I-3

Quantities of Major Fission Products in Waste Storage[a]

Volume Stored, gallons		600,000
Time to Fill Tank, days		1,500
Curies in Storage When Full:		
Tc	99	1.4×10^4
Sm	151	2.7×10^6
Sr	90	1×10^8
Cs	137	1×10^8
Nb	93m	1.5×10^8
Pm	147	2×10^8
Ru	106	2.5×10^7
Ce	144	3.5×10^8
Zr	95	2.5×10^7
Y	91	2×10^7
Sr	89	1×10^7
Ru	103	3×10^6
Ce	141	3×10^6

[a] Assumptions:

1) Waste stored at 400 gallons/ton

2) Processing rate 365 tons/year. Although this rate is not expected to be achieved.

3) Tank size 750,000 gallons filled to 80% of capacity.

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Before the
UNITED STATES ATOMIC ENERGY COMMISSION
Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For licenses for a Spent Fuel Processing Plant
Under Sections 53, 63, 81, 104 (b), and 185 of the

Atomic Energy Act

AEC Docket No. 50-201

Submission No. 6 Final Safety Analysis Report

Report on Status of Completion of Construction and Research and Development Aspects

for the period ending

April 1, 1964

April 20, 1964

Submission No. 6--Final Safety Analysis Report

NFS Operating License Application

AEC Docket No. 50-201

Report on Status of Completion of Construction and Research and Development Aspects

for the period ending

April 1, 1964

The Provisional Construction Permit issued by the AEC to Nuclear Fuel Services, Inc. requires that NFS submit semi-annual reports setting forth the status of completion of construction of the facility and of the research and development program designed to establish the safety of the fuel segmentation process. This submission is the second of these reports.

Construction Progress

The status of construction as of March 1, 1964, is as follows:

1. Engineering is 85% complete and construction is 15% complete.
2. Five hundred forty-seven drawings have been issued for approval and/or quotation and four hundred and eighty-one for construction.
3. Detailed engineering work is proceeding in all areas with major effort being expended on completion

of all models, preparation of piping isometrics for completed models, and structural design of cell slabs, walls, and roofs.

4. Construction work continued on the yard facilities. Work continues on stripping the waste vault walls and on waterproofing and backfilling. The switch station was completed as well as the electrical installation in the warehouse. The cooling tower foundation was completed. The three off-site radiation monitors were installed.

5. The structural construction work included: The General Purpose (GPC) walls were completed; the interior H/D wall of the Liquid Waste Cell (LWC) was poured; the footings were started on the Uranium Product Cell (UPC), the Lower Warm Aisle (LWA), and the Chemical Process Cell (CPC); the interior walls of the water treating station of the Fuel Receiving and Storage Cell (FRS) were continued; and the crane rails were set in the FRS structure.

6. Construction continued on the underground piping and on drain lines and conduit in the FRS, LWA, UPC, and GPC areas. The shielded window-frames were installed in the GPC wall.

Safety of the Fuel Segmentation Process

In the first of these semi-annual reports the splitting of the mechanical operations into two cells, the Process Mechanical Cell (PMC) and the General Purpose Cell (GPC) was discussed. On December 9, 1963 there was submitted to you, as Submission No. 3, a complete revision of Section III of the Safety Analysis. This section describes the plant essentially in its final design and includes a detailed discussion of the PMC and the GPC.

In February, 1964, Bechtel arranged for two qualified consultants from Oak Ridge National Laboratory, Mr. Clyde Watson and Mr. Bruce Finney, to carry out a thorough review of all of the mechanical head and operations of the plant. They spent three weeks at this task at the end of which their conclusions were presented to representatives of Bechtel, American Machine and Foundry (AMF), and Nuclear Fuel Services. They presented a generally very favorable report. They also made many helpful suggestions which have been incorporated wherever possible into the design.

The responsibility for the PMC equipment has been subcontracted by Bechtel to the Atomics Division of AMF who are having the main pieces of equipment built by fabricators to AMF specifications and design. The equipment must withstand the special conditions of radioactivity, remote operations and

maintenance, and decontamination. The two major pieces of equipment, the Disassembly Saw and the Bundle Shear, are now being fabricated with delivery expected July 8 and August 4, 1964, respectively; the main auxiliary castings have been made and are in various stages of machining. Most of the remaining equipment is also in the process of being fabricated and are to be delivered on: Push-Out Ram - July 28; Pin-Shear - about June 15; Cranes - June 24; Maintenance Table - June 5; Television Set - June 1; Power Manipulators - July 1; and Hatch Covers - June 1. The Tilting Fixture, mirrors, adapters, and fixtures are currently being designed with delivery expected by July.

NFS is following closely the work being done at ORNL as a continuation of their development work in mechanical handling. The mechanical development work at ORNL has been concentrated on shearing of second generation fuel. Such fuel incorporates spacer assemblies of meshing metal bands or bent wires located at several positions along the length of the fuel bundles; this design compares with tube sheets or ferrule spacers used in the first generation fuels. ORNL made two attempts to shear intact an Indian Point 'B' fuel bundle which is a prototype design for several future reactor cores. Both attempts were unsuccessful. All the sheared sections did not separate into the small

pieces required for efficient leaching; in fact, some pieces were too large to fall through the exit throat of the shear. It has been tentatively concluded that the elements will have to be pulled out the ends of the spacer assemblies (which are integral with the perforated shroud enclosing the entire bundle); efforts will be made to pull out several rods simultaneously to reduce the processing time.

Some definitive progress has been made in obtaining dummy fuel elements for use in the cold start-up testing of the plant. Most of the major fuel fabricators have been contacted and requested to quote prices for providing one to three prototypes (with depleted UO_2) for each of the typical fuel designs which they have manufactured or plan to manufacture. Consolidated Edison is sending to NFS a prototype bundle of the Indian Point 'A'. Atomics International has sent eight prototype ESADA bundles and several hundred kilograms of UC slugs.

Plant Model

There is being constructed a series of models which, when completed, will detail the entire plant. The models will be used in the field for installation of piping and equipment. They will also be used for the training of operating personnel. They will be useful in establishing operating safety and in obtaining the necessary operating licenses.

The models for most of the plant areas are well on their way to completion. The percent completion for those under construction is listed below:

Plant Area	Percent Completion
Liquid Waste Cell	100
Extraction Cell Number 1	100
Extraction Cell Number 2	100
Extraction Cell Number 3	100
Product Purification Cell	80
Uranium Product Cell	95
Chemical Process Cell	100
General Purpose Cell	93
Lower Warm Equipment Area	100
Upper Warm Equipment Area	100
Process Mechanical Cell	10
Chemical Operating Aisle	45
Lower Extraction Operating Aisle	45
Off-Gass Cell	45
Solvent Storage Tank	100
Utility Room	90

Respectfully submitted,

NUCLEAR FUEL SERVICES, INC.

By S. L. Reese

S. L. Reese
Assistant to President

Subscribed and sworn to before me this 20th day of April, 1964.

_____
Notary Public

My commission expires

_____

REPLACEMENT SECTION – 07-01-1963 – SUBMISSION No. 1
Final Safety Analysis Report

Before the

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For Licenses for a Spent Fuel Processing Plant Under Sections 53, 63, 81, 104 (b), and 185 of the Atomic Energy Act

AEC Docket No. 50-201

Submission No. 1 Final Safety Analysis Report

July 1, 1963

Submission No. 1 Final Safety Analysis Report

NFS Operating License Application

AEC Docket No. 50-201

July 1, 1963

In the Safety Analysis submitted by Nuclear Fuel Services, Inc., in support of its application for a construction permit and operating licenses and in the Public Hearing held at Olean, New York, on March 4-5, 1963, it was indicated that it was the intention to locate the high level wastes completely within a geologic formation described as "silty till" (See paragraph 2.15-2.32 in Safety Analysis). Additional studies of sub-surface conditions have now been completed in the exact location chosen for the waste tanks. These studies were conducted by an independent earth science group, Dames & Moore, 340 Market Street, San Francisco, Calif. Their studies, which are summarized herein, show that the waste tanks are indeed completely located within the silty till.

Attached hereto (Table S2-A1-1) is a statement of Mr. Herbert Stewart, Geologist, USGS, Albany Office, indicating his agreement with the conclusion that the tanks are properly located in the silty till. Mr. Stewart did the original geologic study on the site and offered testimony thereto at the Olean hearing.

The final location of the plant and all of the ancillary equipment including the waste tanks is shown in Figure S2-A1-1. This figure replaces Figure 3.1

in the Safety Analysis. The plant area is shown in more detail in Figure S2-A1-2, a Dames & Moore drawing. This figure indicates the location of additional drill holes, 24 in total, run by Dames & Moore. The most pertinent drill hole is number 23, the drilling log of which is shown in Figure S2-A1-3. A geologic cross section through this area is shown in Figure S2-A1-4. The location of the waste tanks has been superimposed upon both of the latter two figures.

Details of the construction of the waste tank vaults are given in Figures S2-A1-5 and S2-A1-6 and of the tank itself in Figure S2-A1-7. These latter three figures replace Figure 5.47 in the Safety Analysis and when incorporated in the Safety Analysis should be number Figures 5.47a, 5.47b, and 5.47c respectively.

It can be seen from these exhibits that the tank vault will extend from the surface completely to the bottom of the tank. In the course of the construction it will be necessary to excavate a hole somewhat larger than the tank vault in order to permit the construction. In the installation of the tanks a four-foot layer of graded gravel will be put down on the undisturbed silty till in the bottom of the excavation. The vault-tank complex will be built upon this. When the installation is complete the excavation will be carefully refilled with silty till which was removed from the excavation.

The soils consultants, Dames & Moore, have indicated that the silty till underneath the tanks should be kept moist in order to maintain even load bearing characteristics. To assure this they have suggested the arrangement shown in Figure S2-A1-8 (this figure should be added to the Safety Analysis as Figure 5.51). It will be noted that we plan to inject water into the gravel layer underneath the

tank. There will be a series of five 8-inch standpipes located just outside the vault and the water level will be carried in these standpipes partway up the side of the tank.

The tank itself will be completely buried in silty till backfill. In addition it is located so that the liquid level in the tank will be below the original interface between the silty till and the surficial glacial till. Any waste which succeeds in penetrating the first three barriers: the tank, the pan, and the vault, would be expected to mix with the water in the gravel layer under the tank. The specific gravity of the waste would be much higher than that of the water so that it would be expected to remain near the bottom of the system. The standpipes represent test wells which can be used as an additional means to determine very quickly if any waste has escaped from the tank-vault complex. They also represent a method whereby any escaped waste could be rapidly and easily reclaimed with a minimum of dilution and spread into the environment. In essence, then, this system which has been incorporated into the design for structural reasons, adds a fourth barrier to the escape of activity into the environment. There still remains the fifth barrier, the silty till, which can be expected to contain any escaped waste for very long periods.

Thus, the attached design is believed to represent a satisfactory meeting of the intent that the wastes will be enclosed within the geologic formation described as the silty till.

Attached hereto are revisions to paragraphs 5.47 through 5.51 of the Safety Analysis and to Table 5.48.

Respectfully submitted,

Nuclear Fuel Services, Inc.

By_____
Technical Director

Subscribed and sworn to

before me this ____day

of _____, 1963.

My Commission expires_____

TABLE S2-A1-1

Statement of Mr. Herbert Stewart

WUGO 94 PD

Albany NY 21 249P EDT

Charles W. Taylor, Nuclear Fuel Services, Inc.

Bechtel Corp 220 Bush St. SFran.

Silty Clay in Boring 23 appears to be same as our silty clay till.

Herbert Stewart US Geological Survey Ground Water

Branch POB 948 Albany, NY

Silty Silty Till 23 POB 948

138P

FIGURE S2-A1-1

Overall Plot Plan

Bechtel Drawing 4413-15A-A-101

This Figure replaces Figure 3.1

in the Safety Analysis

FIGURE S2-A1-2

Plot Plan

Dames & Moore Drawing Plate 2

FIGURE S2-A1-3

Log of Boring 23

Dames & Moore Drawing Plate AIR

Figure S2-A1-4

Geologic Cross Section Through Tank Area

Dames & Moore Drawing Plate 3C

FIGURE S2-A1-5

Plans--Vault for 8D-1 and 8D-2

Bechtel Drawing 4413-8A-Q-1

This Figure replaces Figure 5.47 in the Safety Analysis and should be incorporated therein as Figure 5.47a

FIGURE S2-A1-6

Section & Details, Sht 1--Vault for 8D-1 and 8D-2

Bechtel Drawing 4413-8A-Q-2

This Figure replaces Figure 5.47 in the Safety Analysis and should be incorporated therein as Figure 5.47b

FIGURE S2-A1-7

Radioactive Waste Tanks 8D-1 and 8D-2

Bechtel Drawing 4413-8A-D-3

This Figure replaces Figure 5.47 in the Safety Analysis and should be incorporated therein as Figure 5.47c

Figure S2-A1-8

Section Through Radioactive Waste Storage Tanks

Bechtel Drawing 4413-8B-Q-8

This Figure should be incorporated

in the Safety Analysis as Figure 5.51.

Replacement Pages for

Paragraphs 5.47 through 5.51 of Safety Analysis

Replacement for Table 5.48

Before the
UNITED STATES ATOMIC ENERGY COMMISSION
Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For licenses for a Spent Fuel Processing Plant
Under Sections 53, 63, 81, 104 (b), and 185 of the

Atomic Energy Act

AEC Docket No. 50-201

Submission No. 2 Final Safety Analysis Report

Report on Status of Completion of Construction and Research and Development Aspects

for the period ending

September 30, 1963

October 10, 1963

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demonstrated for the ram. This feature has been included in the design of the NFS shear.

Plant Models

There are being constructed a series of models which, when completed, will detail the entire plant. The models will be used in the field for installation of piping and equipment. They will also be used for training of operating personnel.

The models for most of the plant areas have been started. The per cent completion for those under construction is listed below:

Plant Area	Per Cent Completion
Liquid Waste Cell	70
Extraction Cell No. 1	30
Extraction Cell No. 2	40
Extraction Cell No. 3	30
Product Purification Cell	30
Uranium Product Cell	50
Chemical Process Cell	70
General Purpose Cell	70
Lower Warm Equipment Area	35
Upper Warm Equipment Area	25
Process Mechanical Cell	5
Chemical Operating Aisle	15
Lower Extraction Operating Aisle	10
Waste Gas Treatment	25

Respectfully submitted,

NUCLEAR FUEL SERVICES, INC.

By /s/ Walton A. Rodger
Walton A. Rodger
Vice President,
Research & Development

Subscribed and sworn to before me this 8th day of October, 1963.

/s/ Patricia Murchake
Notary Public

My Commission expires October 14, 1967.

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Hence for $\eta = 2.07$
$G_o = 11$ gm/liter:

$$@\ G = 20 \text{ gm/liter } k_{\infty,c} = 2.07 \frac{20}{1.07 \times 11 + 20} = 1.303^*$$

$$@\ G = 5 \text{ gm/liter } k_{\infty,r} = 2.07 \frac{5}{16.77} = 0.617^{**}$$

$$M^2 = \tau + L^2$$

M^2 = migration area
τ_2 = Fermi Age
L^2 = Square of the Thermal Diffusion length

$\tau_2 = 32\ cm^2$ for water and dilute solutions
$L^2 = 8.2\ (1 - f) = 8.2\ (1 - k_\infty/\eta)$

Hence for the inner region:

$$M_c^2 = 32 + 3 = 35\ cm^2$$

And for the outer region:

$$M_r^2 = 32 + 5 = 37\ cm^2$$

II. Obtain the Criticality Condition

Use a 1-group, 2-region model and diffusion theory:

Central Region: $$\nabla^2 \Phi_c + \frac{(k_\infty - 1)}{M^2} \Phi_c = 0 \qquad (1)$$

Outer Region: $$\nabla^2 \Phi_r - \frac{(1 - k_\infty)}{M^2} \Phi_r = 0 \qquad (2)$$

* compares to K (nitrate solution) = 1.2306 Table IV.16 DP-532
** compares to K (no nitrates) = 0.68 p. 20 K-1317

The solutions of the above equations in cylindrical geometry are:

$$\Phi_c = A J_o (\mu r), \; r \leq R \tag{3}$$

$$\Phi_r = B K_o (\nu r), \; r \geq R \tag{4}$$

where J_o is the cylindrical cosine function
K_o is the cylindrical negative exponential function

$$\mu = \frac{\sqrt{k_\infty - 1}}{M}$$

$$\nu = \frac{\sqrt{1 - k_\infty}}{M}$$

The two boundary conditions are : continuity of flux and continuity of current at the interface (r = R)

$$\Phi_c (\mu R) = \Phi_r (\nu R) \tag{5}$$

$$D_c \frac{d\Phi_c}{dr} (\mu R) = D_r \frac{d\Phi_r}{dr} (\nu R) \tag{6}$$

where D_c and D_r are diffusion coefficients and are essentially equal

Hency $A J_o (\mu R) = B K_o (\nu R)$ (7)

$$A \mu J_1 (\mu R) = B \nu K_1 (\nu R) \tag{8}$$

or $$\left[\frac{B}{A}\right]_9 = \frac{J_o (\mu R)}{K_o (\nu R)} \tag{9}$$

and $$\left[\frac{B}{A}\right]_{10} = \frac{\mu J_1 (\mu R)}{\nu K_1 (\nu R)} \tag{10}$$

In order to find the k_{eff} of the system, find the k_{eff} at which the two solutions (Equation 9 and 10) for $\frac{B}{A}$ are equivalent, that is

$$\left[\frac{B}{A}\right]_9 = \left[\frac{B}{A}\right]_{10}$$

Redefine μ, ν, according to the definition of k_{eff}:

$$\mu = \frac{\sqrt{\frac{k_{\infty,c}}{k_{eff}} - 1}}{M_c} \quad ; \quad \nu = \frac{\sqrt{1 - \frac{k_{\infty,r}}{k_{eff}}}}{M_r}$$

III. Numerical Solution

A trial-and-error method is used by assuming three appropriate values of k_{eff} and obtaining the behavior of $\left[\frac{B}{A}\right]_9$ and $\left[\frac{B}{A}\right]_{10}$ as a function of k_{eff}. The k_{eff} at which the two are equal is the k_{eff} of the system.

The table below summarizes the calculations:

Using $k_{\infty,c} = 1.303$; $k_{\infty,r} = 0.617$; Radius = 13.0 cm; $M_c^2 = 35\ cm^2$; $M_r^2 = 37\ cm^2$; $(R/M)_c = 2.20$; $(R/M)_r = 2.14$

TABLE:

k_{eff}	$\frac{k_{\infty,c}}{k_{eff}}$	$\sqrt{\frac{k_{\infty,c}}{k_{eff}} - 1}$	μR	$J_o(\mu R)$	$J_1(\mu R)$
0.95	1.372	0.610	1.342	.598	.531
0.90	1.448	0.670	1.474	.526	.554
0.85	1.533	0.731	1.608	.451	.571

k_{eff}	$\frac{k_{\infty,r}}{k_{eff}}$	$\sqrt{1 - \frac{k_{\infty,r}}{k_{eff}}}$	νR	$K_o(\nu R)$	$K_1(\nu R)$	μ/ν
0.95	0.649	0.593	1.269	.290	.390	1.059
0.90	0.686	0.561	1.200	.319	.435	1.229
0.85	0.726	0.524	1.121	.355	.493	1.435

Summary:

k_{eff}	$(B/A)_9$	$(B/A)_{10}$
0.95	2.06	1.44
0.90	1.65	1.57
0.85	1.27	1.66

The two B/A's are equal at about $k_{eff} = 0.89$, and this is therefore the k_{eff} of the system in the example given here.

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Before the

UNITED STATES ATOMIC ENERGY COMMISSION

Washington, D. C.

In the Matter of the Application of

NUCLEAR FUEL SERVICES, INC.

For Licenses for a Spent Fuel Processing Plant

Under Sections 53, 63, 81, 104 (b), and 185 of the

Atomic Energy Act

AEC Docket No. 50-201

Submission No. 8 - Final Safety Analysis Report

Replies to Questions on Criticality Related to the Solvent Wash Columns, Pu Ion Exchangers and Off-Gas Scrubbers

June 12, 1964

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down. Therefore, the uranium entrained in any vapor to the off-gas scrubber will be negligible. The scrubbing solution in the off-gas scrubber is withdrawn to the low level evaporator feed tank periodically carrying with it the trace quantities of uranium if any in the system, thus the system is maintained safe.

_____
W. A. Rodger,
Vice President

Subscribed and sworn to this _____ day of June, 1964.

_____
Notary Public

My Commission expires: