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Met-Ed / GPU

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January 15, 1980
TLL 021

Mr. B. H. Grier, Director
Office of Inspection and Enforcement
Region 1
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Sir:

Three Mile Island Nuclear Station, Unit II (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
TMI-II Incident Quarterly Report

Enclosed is the fifth followup report, the second quarterly report, on the March 28, 1979 incident at TMI-II. This submittal is being made in accordance with Met-Ed's commitment in the letter dated April 11, 1979 (GQL 0490). It provides information compiled subsequent to that contained in the July 16, 1979 and October 15, 1979 reports, together with updates to those reports. The next report will be due on April 15, 1980.

Sincerely,

R. F. Wilson
Director - TMI-II

RFW:CFM:hah

Enclosure: TMI-II Incident Report
dated January 15, 1980

cc: Director of Nuclear Reactor Regulations
Attn: S. A. Varga
Light Water Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

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FIFTH INTERIM REPORT ON THE
THREE MILE ISLAND NUCLEAR STATION
UNIT II (TMI-2) ACCIDENT

JANUARY 15, 1980

METROPOLITAN EDISON COMPANY

8001230154

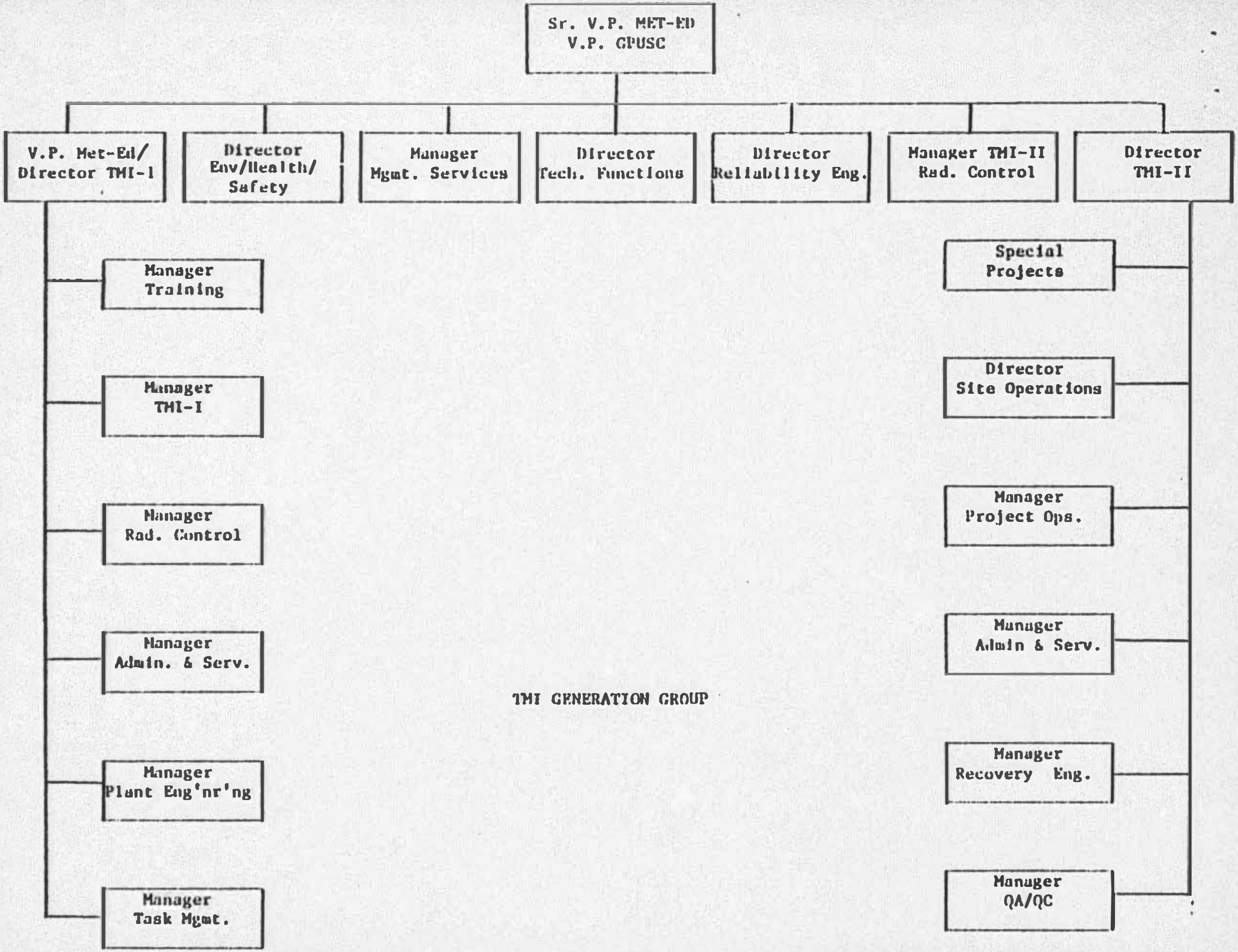
CONTENTS

I. SEQUENCE OF EVENTS

Since this section is presently undergoing updating it is not available at this time. An updated copy will be forwarded in a future submittal.

II. RECOVERY ORGANIZATION

Included in this section are organization charts representing the TMI Unit II Recovery Organization for the period of October 1 through December 31, 1979.



TMI GENERATION GROUP

III. PLANT MODIFICATIONS

Included in this section are updated and amended subsections from previous Interim Reports. Changes from the previous reports are denoted by change bars in the right hand margin and Rev. 4 on the bottom right hand corner of the page. Subsections from the previous Interim Reports which have not had any changes are not included in this report.

A. Hydrogen Recombiners

1.0 System Function and Design Objectives

In anticipation of having to process substantial amounts of hydrogen to prevent a hydrogen explosion in the reactor building, and because of the uncertainty of the quantity of hydrogen being generated, the available hydrogen recombiner capacity was increased.

Operations shall not permit an uncontrolled release of reactor building atmosphere to the environment.

2.0 System Description

A thermal-type hydrogen recombiner is installed in the fuel handling building at the spent fuel pool operating floor and is connected to the reactor building ventilation and purge system as originally intended (see FSAR Figure 6.2-30). In addition, a skid-mounted, thermal-type hydrogen recombiner has been installed next to, and has been connected in paralled with, the first hydrogen recombiner. The integrity of the system is as originally installed and will ensure that there is no uncontrolled radioactive release to the environment.

3.0 System Operation

The hydrogen recombiners are only operated as required to control reactor building hydrogen concentrations. Periodic samples are taken from the containment atmosphere to monitor hydrogen concentration levels.

Recombiner operation is monitored and controlled manually from a local panel. A recombiner "trouble" alarm is annunciated in the control room.

4.0 Status

The Hydrogen Recombiners are installed and functional. A program is in process to decontaminate one of the units and remove for eventual installation in Unit I.

B. Auxiliary and Fuel Handling Building Supplementary Air Filtration Systems

1.0 System Function and Design Objectives

Radioactive iodine, released from the Reactor Coolant System during the TMI Unit 2 accident, was transferred into the Unit 2 Auxiliary and Fuel Handling Buildings. Immediate change out of the Auxiliary and Fuel Handling Building charcoal filter trains was not feasible because of the high radiation and contamination levels in the filter areas. As a consequence of the I-131 release rate, it was decided to construct a supplementary air filtration system to reduce off-site releases. The function of the system is to remove radioactive particulates and radioiodine in the exhaust air from the Auxiliary and Fuel Handling Buildings.

2.0 System Description

The system interfaces with the Auxiliary, Fuel Handling and Service Building HVAC Systems, and is installed on the roof of the Auxiliary Building.

The system consists of four parallel filter trains, each comprised of an exhaust fan and a filter unit.

3.0 System Operation

Air is drawn from the duct connecting the penthouse plenium to the station vent through a common inlet duct which connects the four parallel filter trains.

The inlet duct is provided with an in-line radiation monitor. Air then flows through the operating filter trains, through a radiation monitor in each outlet duct and is exhausted to the atmosphere through the fans.

4.0 System Status

Engineering is Complete.

Construction is Complete.

System description, flow diagrams, and operating procedures, are complete. An operating and failure modes analysis has been prepared.

All four (4) trains are operable. The station vent is capped. Present operation is with three (3) trains.

D. Fuel Pool Waste Storage System

1.0 System Function and Design Objectives

This Fuel Pool Waste Storage System is used for temporary storage of liquid waste. These tanks add approximately 110,000 gallons to the present storage capacity of the plant, and are located within the "A" spent fuel pool. These tanks can be filled with liquid waste from the Reactor Building Sump and the Miscellaneous Waste Hold-Up Tank. This system enhances the capability of the plant to move and process radioactive waste.

2.0 System Description

The system consists basically of upper (4 at 15,000 gallons each) and lower (2 at 25,000 gallons each) tanks, forming two separate storage areas. Either storage area is capable of being filled from either the Reactor Building Sump or the Miscellaneous Waste Hold-Up Tank, and each has level indication. The tanks are protected from over-filling by automatically closing the feed valve when the storage area is nearly full. Provisions have been made to both flush the piping system after completion of the pumping operation, and to drain the piping system as required.

The vents from the tanks and the stand pipes are directed through a dryer and a charcoal filter to remove moisture and iodine before proceeding to the fuel pool ventilation system. The tanks and vent system is protected by a relief valve which vents through a parallel set of dryers and charcoal filters.

The tanks will be emptied as necessary by steam eductors. Two eductors are permanently installed in each stand pipe.

3.0 System Operation

Water is transferred from the Reactor Building Sump or the Miscellaneous Waste Storage Tank to the tank farm.. After either the lower set of tanks or upper set of tanks is full the level controllers automatically close the air operated inlet valves.

Air forced from the tanks during the filling process is vented to a charcoal filter & dryer to remove moisture and iodine. This air is then piped to the Fuel Pool Ventilation System.

The steam eductors give the capability to transfer waste water from the tank farm to the Miscellaneous Waste Storage Tank or Epicor II Rad Waste System, from the upper tanks to the lower tanks in the tank farm (or vice versa) or to recirculate the water in the tanks.

D. Fuel Pool Waste Storage System (continued)

A high temperature alarm and temperature switch to close the steam control valve, is installed in the tank vent line to prevent damage to the filter/dryer skids during use of the eductors.

4.0 System Status

The system is being used to store 93,000 gallons from the Unit 2 Miscellaneous Waste System. The steam eductors have not been used since no water has been pumped out of the tanks to date.

E. Upgraded Decay Heat Removal System

1.0 System Function and Design Objectives

Future operation of the existing decay heat removal (DHR) system may result in radiation levels possibly ranging up to 500 Rads per hour in the vicinity of the system fluid components. This condition would severely limit personnel access for routine surveillance, operation, and maintenance. The upgraded DHR system consists of a program intended to identify, evaluate, and implement modifications necessary to ensure the integrity and reliability of the system in a radiation environment, substantially exceeding the original design basis, for up to one year of operation.

2.0 System Description

Proposed DHR system modifications include additional decay heat vault shielding, a remote TV monitoring system, modified DHR pump and motor bearing oilers, a vibration monitoring system, and associated operating and testing procedures.

Vault shielding will be provided by lead bricks assembled in a steel support frame. This will reduce the ambient personnel radiation exposure levels to "as low as reasonably achievable" (ALARA) in the accessible area above the vault. Radiation surveys will be made during initial DHR system operation and periodically thereafter to determine shield effectiveness.

The TV monitoring system will provide remote surveillance capability for DHR system operation and maintenance. Two independent systems are provided, one for each vault. Each system includes a radiation-tolerant, closed-circuit television with remote controls. Specific operations to be monitored include pump and motor bearing oil level, pump packing leak-off, remote oil fill, and pump venting.

DHR pump and motor bearing oiler modifications will provide for increased oil storage capacity, a means for remotely reading oil levels, and to permit feeding of oil to the bearings.

Provision for remote venting of the pumps is also provided.

Provisions will be made for monitoring pump vibration and loose parts in the system. This is intended to provide early indication of pump and motor degradation, loose parts in the system (particularly at the heat exchanger tube inlet), and changes in flow patterns due to partial line blockages.

Monitoring and control for these modifications will be provided from the fan room at elevation 322 in the service building.

3.0 System Operation

These modifications to the DHR will not appreciably alter system operation.

4.0 Status

The TV monitoring system, the bearing oil tanks and piping, and pump venting arrangement are installed and operational.

The installation of vault hatch shielding plugging is in process in support of the Mini Decay Heat System.

F. Steam Generator "B" Closed Loop Cooling System

1.0 System Function and Design Objectives

In order to provide a high pressure, closed cooling loop for water-solid steam generator "B", a system utilizing new equipment must be installed. The closed loop must remove the decay heat from the core plus the added heat load from one reactor coolant pump. To minimize the possibility for contamination of the closed loop, the system must be operated at a higher pressure than the reactor coolant system. The heat transferred to the closed loop will ultimately be rejected to the river. The system is intended to provide backup decay heat removal capability should the present steaming from steam generator "A" be discontinued.

2.0 System Description

The system consists of a new heat exchanger, pump, surge tank, piping and valves. The hot water leaving the steam generator will pass through the tube side of the new heat exchanger and return to the steam generator via the new pump. A pressurizer surge tank will maintain the steam generator secondary side pressure above the primary coolant system pressure.

The shell side of the heat exchanger is supplied with cooling water from the secondary services closed cooling water system which, in turn, will be cooled by water from the nuclear services river water pumps piped to the turbine building via the secondary services river water piping.

The new pump discharge piping is connected to the existing feedwater piping downstream of the main feedwater pumps, and the heat exchanger inlet piping is connected to the drain pot on the main steam line between the main steam isolation valve and main turbine stop valves.

3.0 System Status

System is operational and in standby.

J. Steam Generator "A" Closed Loop Cooling System

1.0 System Function and Design Objectives

In order to provide a high pressure, closed cooling loop for water-solid steam generator "A", a cooling system utilizing new equipment has been proposed. The closed loop would remove the decay heat from the core plus the added heat load from one reactor coolant pump. To minimize the possibility for contamination of the closed loop, the system would be operated at a higher pressure than the reactor coolant system. The heat transferred to the closed loop would be rejected to the river. The system would be intended to provide primary decay heat removal capability redundant to the steam generator "B" closed loop cooling system.

2.0 Description

The system will consist of a new heat exchanger, pump, surge tank, and piping and valves. The hot water leaving the steam generator would be cooled in the shell side of the heat exchanger and returned to the steam generator by a new pump. A pressurized surge tank would maintain the steam generator secondary side at a minimum pressure greater than the primary coolant system pressure.

The tube side of the heat exchanger would be supplied with cooling water from the nuclear services river water pumps piped to the turbine building via installed secondary services river water piping.

The new pump discharge piping would be connected to the existing feedwater piping downstream of the main feedwater pumps. The heat exchanger inlet process piping would be connected to the main steam turbine bypass line between the isolation valve and the control valve at the condenser.

3.0 System Status

This system will not be installed.

L. Alternate Decay Heat Removal System

1.0 System Function and Design Objectives

The proposed Alternate Decay Heat Removal (ADHR) system augments the two existing DHR systems and the proposed water solid secondary/natural circulation system as backup to steam generator "A" steaming. An integral Decay Heat Closed Cooling Water (DHCCW) system is included to transport heat from the ADHR cooler and the ADHR pump seal coolers to the nuclear services river water system. Connection points are also provided outside the fuel handling building to connect other dedicated liquid waste processing systems.

The specific function of the ADHR system is to remove decay heat such that the reactor coolant system can be brought to and maintained at a cold shutdown condition. With the exception of gross core flow restrictions, this system is intended to provide sufficient core flow to maintain reactor coolant subcooled.

2.0 System Description

The two ADHR pumps and a new heat exchanger will be mounted on a skid located outside the west wall of the fuel handling building and penetrate the fuel handling building west wall of a valve vault. The pipe runs will terminate in the valve vault by capping each line. Hook-up to the ADHR skid will be made later if needed. In addition, three capped taps will be provided on the ADHR piping installed outside the fuel handling building. These taps may be used later to connect other dedicated liquid waste processing systems.

Motor control centers and I&C panels for operation of all ADHR system pumps and motor operated valves will be mounted in a control trailer located near the ADHR skid.

The DHCCW system provides cooling water to the ADHR system heat exchanger and pump seal coolers. It utilizes a closed loop system to provide a double barrier between the ADHR system and the river water to prevent the direct release of radioactivity to the environment. A radiation detector is provided to monitor the level of radioactivity in the DHCCW system at the outlet of the DHR cooler. A radiation level indicator with high radiation level alarm is located in the ADHR system remote control room. If radioactivity is detected, operation of the decay heat removal loop and its associated DHCCW loop can be halted and the affected decay heat removal cooler isolated. The DHCCW system is mounted on a second skid and consists of the DHCCW pump, heat exchanger, and surge tank. Both skids will be located outdoors at grade level near the west wall of the fuel handling building and adjacent to each other.

3.0 System Cperation

A detailed description of this system is in the Westinghouse turnover document.

4.0 System Status

The piping for the ADHR system has been designed, fabricated, and received on site. The skid for the ADHR system with its components, two pumps, heat exchanger, valves and piping is completed. Motor control centers are on site. The valve vault excavation is completed and piping installation up to the second isolation valves is completed. The electrical trailer is completed. Electrical power and service water connections would not be made unless the system were put into service. Tie-in of the ADHR system to the existing plant DHR system has been completed. Disposition of the valve pit is under evaluation.

M. Standby Reactor Coolant Pressure Control System

1.0 Systems Function and Design Objectives

High radiation levels and flooding in the reactor building have or could potentially render much of the reactor coolant (RC) system electrical equipment and instrumentation inoperable. With much of the instrumentation inoperable, the RCS should be maintained water "solid". An alternate system of pressure control is required to ensure safe and reliable cooling of the reactor core, should control of the existing system become unmanageable. The standby reactor coolant pressure control (SRCPC) system will ensure reliable core cooling by performing the following function:

a. Maintain the RC system in a water-solid condition for natural circulation core cooling.

b. Maintain sufficient available NPSH should RC pump operation be required.

c. Control the quality of the makeup fluid.

d. Maintain pressure within control limits while accomodating thermal and volumetric contractions in RC system inventory.

2.0 System Description

The SRCPC system ties into the existing High Pressure Injection lines (see FSAR Figure 9.3-6). RC system pressure is maintained by three surge tanks arranged in series with a pressurized nitrogen blanket over the last tank. A fluid inventory of approximately two thirds of the total tank capacity is sufficient to maintain RC system pressure during sudden RC system inventory reduction transients. A level control valve at the tanks' discharge will prevent nitrogen from entering the RC system.

Long term makeup will be provided by the charging pump taking suction from an atmospheric storage tank. Makeup fluid conditions are adjusted by chemical addition and heating to meet RC system water quality requirements.

The RC system pressure will normally be maintained between 50 and 600 psig during the intended cooldown process.

The SRCPC makeup system will be operated manually from a local panel during initial operation and from the control room after system automation is complete. Makeup is provided in response to decreasing pressure in the RC system. An alarm will annunciate at the control station when the pressure differential between the RC and SRCPC makeup system reaches or exceeds 50 psi.

The SRCPC makeup system will prevent gross depressurization on the RC system when operating in a water-solid mode. Over-pressurization protection can be provided by increased letdown resulting directly from RC system pressure increase, letdown with concurrent termination of RC pump seal injection or makeup, opening the pressurizer vent valve, opening the pressurizer electromatic safety relief block valves, or lifting the pressurizer safety relief valves (the latter two methods are undesirable and will only be considered as a last resort).

3.0 Status

Phase II is operational.

O. Liquid Radioactive Waste Processing System Title "EPICOR II"

1.0 System Function and Design Criteria

The system is designed to cleanup radioactive liquids so as to produce water capable of being released from Three Mile Island. Cleanup includes removal of radioisotopes and chemical constitutents to comply with Plant Technical Specifications for Water Releases to the Susquehanna River. The design is optimized with respect to ALARA considerations.

Instrumentation and controls are provided for monitoring of system performance. Water flows are monitored where the values are critical to the process and or system safety. Inline monitoring and a comprehensive sampling system are provided for thorough analyses of system water cleanup performance. Radiation and airborne monitoring equipment is provided for analysis of activity levels.

Shielding is being provided to minimize exposure related to the operation of this system.

An HVAC subsystem is utilized to cleanup and monitor any gases that might be released from the liquid processing system. It is the goal to minimize gas releases from the system, however, should they occur, they will be cleaned to reduce any releases to the environment. Monitoring of the air exhaust will continue to detect any potential radioactive gas. A slight negative pressure is maintained to ensure building inleakage is maintained. The system is being optimized with respect to ALARA considerations.

2.0 System Description

Liquid Processing

The TMI Station Chemical Cleaning Building is used to house the system along with the existing tankage and sump existing in that building. Piping and pumps are provided for water movement through cleanup vessels. The system is composed of a pre-filter, two demineralizers and an after filter. The pre-filter and demineralizers are designed for ease of hookup and disconnect to allow for quick installation and remote, reliable removal.

Gas Processing

The primary components are a fan, an air cleanup filter train, and necessary ducting. The main HVAC components located external to the Station Chemical Cleaning Building, but are enclosed in their own shelter.

3.0 System Operation

The Auxiliary Building Emergency Liquid Cleanup System consists of a vendor supplied liquid radwaste process system which is located in the Chemical Cleaning Building. The system is designed to decontaminate by filtration and ion exchange approximately 400,000 gallons of radioactive waste water contained in the Auxiliary Building of TMI Unit 2. Contaminated water is being pumped from a connection located on the Miscellaneous Waste Holdup Tank (WDL-T-2) by a pump located in the Chemical Cleaning Building through the yard and into the process system. Yard piping is enclosed within a guard pipe, the end of which terminates inside the Chemical Cleaning Building.

Decontaminated water is delivered to the Clean Water Receiving Tank (CC-T-2) for sampling and analysis and pumped to the Liquid Waste Disposal System of TMI Unit II for storage if within specs, or transferred to the Off Spec Water Receiving Batch Tank (CC-T-1) for recycling through the process system. Capability also exists to discharge to a tank truck. CC-T-1 may also be used for storage.

The Chemical Cleaning Building (CCB) has been made into a low leakage confinement building and provided with an exhaust ventilation system to maintain the building at a negative pressure. HEPA and charcoal filtering is provided on the ventilation system which discharges to a local stack at the roof line of the CCB where all effluent air is monitored for radioactivity.

Normal operation of the processing system is by remote means except for infrequent operations, such as sampling and chemical addition. All remote system operations are controlled from the TV Monitor Control Building located outside the northwest corner of the Chemical Cleaning Building.

Remote handling of spent resin containers from their position inside the Chemical Cleaning Building to the transport cask and truck are provided.

The system interface with the TMI Unit 2 Radwaste Disposal Miscellaneous Liquids System, the TMI Unit 1 Liquid Waste Disposal System, Demineralized Water System and the Service Air System.

4.0 Status

The system is operating successfully, and has processed approximately 90,000 gallons. A total of sixteen (16) spent resin liners have been used and are in the wasta staging area. All processed water is stored in EPICOR II storage tanks.

P. Trash Compactor

1.0 System Function and Design Criteria

Additional compaction facilities were required due to the amount of compactible waste being generated. Therefore, a system was needed to compact low level solid waste into 55 gallon drums for storage or shipment.

The system shall compact waste into 55 gallon Department of Transportation (DOT) drums meeting requirements for shipping LSA material.

2.0 System Description

A Stock Equipment Company Model 2407 compactor was installed in the Unit 2 Model Room. A drawing of the unit is attached. The unit includes roughing and HEPA filters. The discharge of the unit is vented to the Aux. Building ventilation system which contains charcoal filters.

3.0 System Operation

Trash is compacted into 55 gallon DOT approved drums. Standard plant operating procedures have been revised to include the use of the new compactor. Use is limited to compactible dry waste only. No wood, metal or liquids are permitted. All bags of trash are surveyed prior to compaction; bags in excess of 300 mr. are not compacted.

4.0 Status

Compaction of trash for LSA shipments continues. Drums, that have been compacted, are stored in the compacted waste staging facility until ready for shipment to a radwaste burial ground in Hanford, Washington. 307 drums were shipped from Oct. through December 1979 with 107 drums in storage at the end of this period.

Q. Staging Facilities for Dewatered Resins and Evaporator Bottoms

A. WG 21 - Interim Solid Waste Staging Facility

1.0 System Function and Design Criteria

Facilities are needed to stage dewatered radioactive resin and filters generated by EPICOR I and EPICOR II until they can be shipped to a burial site. WG-21 provides space for this staging.

2.0 System Description

The facility consists of 16-54" diameter cells and 12-84" diameter cells to receive 4' x 6' and 6' x 6' resin liners. The cells are installed in the Unit 2 cooling tower desilting basin, backfilled for shielding and capped with 3' thick concrete plugs.

3.0 System Operation

Eight (8) EPICOR I Resin Liners, one (1) EPICOR I Prefilter, and one (1) smaller resin liner (used to remove trace activity and fluorescein dye) are staged in the facility. Sixteen (16) EPICOR II resin liners and one (1) Unit 1 used precoat liner are also staged in the facility.

4.0 System Status

Construction of the interim solid waste staging facility has been completed and is operational. Additional shielding (lead bricks) were installed along the interface between the cell cover and facility top to provide shielding due to streaming on some of the cells that are loaded. Readings are below the 5 mr/hr design criteria.

B. WG-22 Solid Waste Staging Facility

1.0 System Function and Design Criteria

Facilities are required to stage the following radioactive wastes until they can be shipped to a burial site:

1.1 Dewatered radioactive resins from EPICOR I.
1.2 Dewatered radioactive resins from EPICOR II.
1.3 Dewatered radioactive resins or solidified evaporator bottoms from systems used to process water more radioactive than that processed by EPICOR I or EPICOR II.

The sump meets the seismic requirements of Reg. Guide 1.143. Contact readings on the sides of the facility will be less than 0.5 mr/hr and less than 2.5 mr/hr on the top.

2.0 System Description

The facility is designed as a modular one. Each module consists of 60" - 84" diameter cells imbedded in concrete and capped with 3' thick concrete plugs. Each cell has a drain line to a sump which will serve three modules. The sump is designed to collect any leakage from liners installed in the cells and meets the seismic requirements of Reg. Guide 1.143.

3.0 System Operation

Module A is near completion and is expected to be operational in early January 1980.

4.0 System Status

Work is continuing on the facility sump. Painting on some of the cells is complete. Some backfilling is complete and will be completed in early 1980.

The mudmat has been poured for Module B and construction schedules are being prepared for Modules B & C.

S. Nuclear Sampling System

1.0 System Function and Design Objectives

This nuclear sampling system is to be used as a temporary liquid waste sampling facility to allow TMI Unit 2 recovery operations to continue without interfering in the normal operations of Unit 1 when that unit is returned to service. It will provide a single controlled station whereby fluid samples may be taken from tanks other wise innaccessible for local sampling and/or from tanks that require frequent sampling for analyses of chemical and radiochemical content. Included in the sampling scope will be capability for representative samples of Unit 2 Reactor Coolant from the pressurizer steam or water space or upstream of letdown coolers, and from the Mini-Decay Heat System, samples from the three Unit 2 Reactor Coolant Bleed Tanks, Unit 2 Miscellaneous Waste Hold-up Tank and the new Fuel Pool Waste Storage System containing liquid waste from both the Unit 2 Reactor Building Sump and Miscellaneous Waste Hold-up Tank. Provisions have also been provided in the system for monitoring of boron concentration in the reactor coolant.

2.0 System Description

Unit 2 Sample Lines which presently run into Unit 1 sampling area shall be rerouted to a new sample sink to be located in the Fuel Handling Building 305' elevation of Unit 2. In an adjacent room, the so-called "model room" a boronometer shall be installed.

The system shall provide for adequate recycle, purge and return of waste liquids. Purging of radioactive piping shall be performed prior to installation of new sample lines.

Drainage from the sample sink will be routed to the Fuel Pool Waste Storage System. A shielded bottle to collect drainage will also be provided.

All piping, valves and components of the sampling system will meet the design conditions of the system with which they are associated or will meet 150 psig and 200 F. Primary coolant sampling points will have the design condition of 2500 psig and 670 F up to valve SNS-V-70.

Air exhausted from the sample hood will be filtered through charcoal and HEPA filters and discharged to the Auxiliary Building ventilation system exhaust ductwork.

3.0 System Operation

A detailed description of the systems operation is not yet available as design changes are still being made. This description shall be incorporated in a subsequent report.

4.0 System Status

The system design is essentially complete. Construction is in progress and will be completed in early 1980.

T. Mini-Decay Heat Removal System

1.0 System Function and Design Objectives

The specific function of the MDHR system is to remove decay heat such that the reactor coolant system can be brought to and maintained at a cold shutdown condition. The system is intended to provide sufficient core flow to maintain reactor coolant subcooled.

2.0 System Description

The two MDHR pumps and two heat exchangers will be mounted at the south end of the 280'-6" elevation in the fuel handling building. New pipe runs will be installed from the existing DHR system piping to the new equipment. Cooling water to the heat exchangers is provided by the existing Nuclear Services Closed Cooling System by means of new piping. One pump and one heat exchanger can accommodate the current decay heat load from the core.

The system will be capable of being monitored and controlled from a new control panel in the control room or a local control panel.

The system piping and components are small to minimize the volume of reactor coolant outside of the reactor building.

3.0 System Operation

A detail system description (Rev. 0) was issued November 16, 1979. The operating procedure has been written and is being reviewed. It will be available by January 31, 1980.

4.0 System Status

The engineering is approximately 95% complete. Piping fabrication and installation is about 95% complete. The electrical and HVAC installation is approximately 80% complete. The current schedule for completion is February 8, 1980 for the piping system and February 28, 1980 for the HVAC system.

U. Alternate Condensate Pumps Subsystem

1.0 System Function and Design Objectives

The alternate condensate pumps are intended to provide backup to the existing condensate pumps to supply feedwater to the steam generators for decay heat removal and/or provide feedwater to the temporary auxiliary boiler (see separate section for temporary auxiliary boiler).

2.0 Description

The two 50 gpm alternate condensate pumps are piped to take suction from the condenser hot well and discharge to the steam generator through either of two new condensate demineralizers.

3.0 Status

The system is installed and is being used to provide feedwater to the Temporary Auxiliary Boiler.

V. Temporary Auxiliary Boiler System

1.0 System Function and Design Objectives

The temporary auxiliary boiler system is intended to furnish steam to the Unit II turbine gland seals so that the existing auxiliary boilers (Unit I) can be shutdown and serviced.

2.0 Description

The temporary (skid mounted) auxiliary boiler is designed to receive feedwater from the alternate condensate pumps and deliver 185 psig steam to the Unit II auxiliary steam header.

3.0 Status

The boiler skid is in place and operational.

W. TMI-II Low Level Waste Processing System (EPICOR I Relocation)

1.0 System Function and Design Objectives

The low level liquid waste processing system for $\leq$ 1 µc/ml wastes shall utilize the discharge from the Contaminated Drain Pumps (WDL-P15A and B, Crane Company -Deming Division-Single Stage Centrifugal Pumps, provided as serial numbers DC - 551976 and DC - 551977 under Deming Division Customer Order No. C-0066) as a feed source. Cartridge filtration equipment (WDL - F7A and B) between the pumps and the processing system is available for use in series with the processing system prefilter.

Effluents from the process system will be collected in a WASTE PROCESS MONITOR TANK for sampling and discharge to the plant Evaporator Condensate Test Tanks (WDL - T-9A and/or B) or recycle cleanup, as indicated by the sample results.

2.0 System Status

Relocation of EPICOR I.

X. EPICOR II Solidification System

1.0 System Function and Design Objectives

Pursuant to USNRC order, EPICOR II system solid wastes (i.e.-resins) must be contained in a solid, free standing monolith, for shipment and burial. The EPICOR II solidification system is being provided to accomplish solidification of these wastes.

2.0 System Description

The system remains in the conceptual design and development phase.

3.0 System Operation

Later.

4.0 System Status

Conceptual design phase.

Y. Containment Service Building

1.0 System Function and Design Objectives

a. Provides contamination and airborne particulate control envelope at the containment equipment hatch.

b. Provides for efficient personnel access to containment.

c. Allows passage of large pieces of equipment and bulk radwaste.

d. Provides a waste staging and temporary storage area.

e. Provides a decontamination area for equipment removed from containment.

f. Provides space to handle containment service systems.

g. Allows for maintaining a hot tool crib in vicinity of containment.

2.0 System Description

Building area of approximately 20,000 FT^2 height approximately 35 ft located adjacent to the equipment hatch.

3.0 System Status

Design criteria development phase.

Z. Evaporator/Solidification Facility

1.0 System Function and Design Objectives

The Evaporator/Solidification Facility shall provide for the collection, treatment, storage and disposal of liquid radioactive wastes generated during decontamination of TMI Unit II and for the collection, treatment, storage, solidification and disposal of spent resins generated within the facility.

2.0 System Description

The Evaporator/Solidification facility occupies an area of 112' x 62' and is 67' high. It is located next to the west wall of the diesel generator building. Water from the various tanks in Unit II and from the containment service building is piped to the Facility, where it is chemically treated prior to entering the HPD Evaporator. The distillate from the Evaporator is passed through a demineralizer and stored in RC Evaporator Condensate test tanks prior to discharge to the river or recycling.

Concentrates from the Evaporator are stored in a concentrated waste storage tank from where they can be slurried to the solidification subsystem, as can be the Unit II spent resins.

3.0 System Status

Engineering of the facility is approximately 60% complete. Preliminary flow diagrams and equipment specifications have been reviewed. Final documents to be issued in February.

No fabrication or purchase of equipment other than the Evaporator has been started. The Evaporator should be on site July 1980.

IV. DECONTAMINATION OF AUXILIARY AND FUEL HANDLING BUILDING. THREE MILE ISLAND, UNIT II.

A. Function and Objectives

1. The decontamination effort in the Auxiliary and Fuel Handling Building is to decontaminate all areas of these buildings to less than 1,000 DPM, and to reduce radiation levels to design levels.

B. Decontamination Activities

1. Decontamination of open areas (corridors, stairwells, etc.) is 89% complete. Cubicle decontamination is 73% complete. General radiation levels in both buildings has been reduced to less than one (1) MR/hr. except for isolated tank cubicles and valve alleys. No tank, sump or internal piping decontamination has been accomplished to date. This effort is scheduled and will commence when the details for the flushing water to be used have been finalized.

2. Decontamination efforts are restricted by the use of water in the cleanup effort. The only liquids being used for decontamination are radiac wash (decon solution) which must be contained in drums and solidified. When an acceptable source of water has been made available, the use of a hydrolaser for decontamination will be put into effect. This will aid in the decon effort.

3. Solidification of twenty-two hundred (2200) gallons of radiac wash, the only liquids generated to date during the decon effort, has been accomplished. This solidification generated seventy-three (73) drums of solidified waste. All solidified drums were less than 100 MR/hr. on contact.

C. Decontamination Status

1. General Radiation levels have been reduced from 1 R/hr. except for isolated cubicles and valve alleys.

2. Air borne activity has been reduced from 6×10^{-7} μc/cc to 3×10^{-11} μc/cc.

3. Surface contamination has been reduced from 15×10^{6} DPM to less than 1,000 DPM in all accessible areas.

V. RADIOLOGICAL MONITORING

This section contains an Executive Summary of TMI Units I & II Liquid and Gaseous Releases as a Result of the Incident of March 28, 1979, and continuing through October, 1979. Tables 1-9 provide additional data for liquid discharges to the Susquehanna River. Also attached are updated running tables that contain the results of analyses performed on water samples taken in the vicinity of the Three Mile Island Nuclear Station.

EXECUTIVE SUMMARY

THREE MILE ISLAND UNITS 1 and II LIQUID and GASEOUS RELEASES

DISCHARGE PATHWAYS	1st Quarter 1/1/79 to 3/31/79	Incident Period 3/28/79 to 3/31/79	4/1/79 to 4/30/79	5/1/79 to 5/31/79	6/1/79 to 6/30/79	2nd Quarter 4/1/79 to 6/30/79
I. Liquid Released:						
a) Discharge less Tritium:						
1) Concentration (μCi/cc)	1.29E-8 (a)	7.44E-8 (a)	1.75E-7 (a)	2.89E-8 (a)	2.84E-8 (a)	8.63E-8 (a)
2) Total Activity (Ci)	0.277E0 (b)	1.00E-1 (b)	1.62E0 (b)	2.21E-1 (b)	1.88E-1 (b)	2.03E0 (b)
b) Iodine-131 Released:						
1) Concentration (μCi/cc)	4.97E-9 (a)	7.16E-8 (a)	1.70E-8 (a)	2.25E-9 (a)	5.60E-10 (a)	7.57E-9 (a)
2) Total Activity (Ci)	0.107E0	9.62E-2	1.57E-1	1.72E-2	3.70E-3	1.78E-1
c) Tritium Released:						
1) Concentration (μCi/cc)	4.83E-6 (a)	5.13E-7 (a)	8.45E-7 (a)	7.05E-7 (a)	4.60E-7 (a)	6.77E-7 (a)
2) Total Activity (Ci)	104.1E0	0.69E0	7.80E0	5.38E0	3.04E0	1.59E1
II. Airborne Iodine Released:						
a) Quarterly Release Rate (μCi/sec)	5.8E-1	5.8E-1	1.20E0	9.89E-3	2.12E-5	1.22E0
b) Total Iodine-131 Released (Ci)	4.57E0	4.57E0	9.48E0	7.8E-2	1.67E-4	9.6E0
III. Noble Gases Released:						
a) Quarterly Release Rate (Ci/sec)	1.12E0	1.12E0	1.41E-1	1.81E-4	9.5E-5	1.41E-1
b) Total Noble Gases Released (Ci)	8.83E6	8.83E6	1.11E6	1.43E3	7.50E2	1.11E6

FOOTNOTES:

a) Concentrations are based upon actual MDCT flows. These are concentrations in the effluent averaged over the period.

b) This data includes Iodine-131 released to the Susquehanna River as a result of the TMI Unit II accident on March 28, 1979.

EXECUTIVE SUMMARY

THREE MILE ISLAND UNITS I and II LIQUID and GASEOUS RELEASES

DISCHARGE PATHWAYS	7/1/79 to 7/31/79	8/1/79 to 8/31/79	9/1/79 to 9/30/79	3rd Quarter 7/1/79 to 9/30/79
I. Liquid Released:				
a) Discharge less Tritium:				
1) Concentration (μCi/cc)	1.12E-8 (a)	2.66E-9 (a)	2.33E-9 (a)	5.18E-9 (a)
2) Total Activity (Ci)	7.85E-2 (b)	1.89E-2	1.78E-2	1.13E-1
b) Iodine-131 Released:				
1) Concentration (μCi/cc)	4.57E-10 (a)	9.10E-11 (a)	8.62E-11 (a)	2.07E-10 (a)
2) Total Activity (Ci)	3.20E-3	6.46E-4	6.59E-4	4.51E-3
c) Tritium Released:				
1) Concentration (μCi/cc)	7.20E-7 (a)	3.20E-7 (a)	3.34E-7 (a)	4.53E-7 (a)
2) Total Activity (Ci)	5.04E0	2.27E0	2.55E0	9.86E0
II. Airborne Iodine Released:				
a) Quarterly Release Rate (μCi/sec)	1.58E-6	<MDA	<MDA	1.58E-6
b) Total Iodine-131 Realease (Ci)	1.24E-5	<MDA	<MDA	1.24E-5
III. Noble Gases Released:				
a) Quarterly Release Rate (Ci/sec)	1.27E-5	1.14E-5	8.88E-6	3.30E-5
b) Total Noble Gases Released (Ci)	100	90	70	260

FOOTNOTES:

a) Concentrations are based upon actual MDCT flows. These are concentrations in the effluent averaged over the period.

b) This data includes Iodine-131 released to the Susquehanna River as a result of the TMI Unit II accident on March 28, 1979.

EXECUTIVE SUMMARY

THREE MILE ISLAND UNITS I and II LIQUID and GASEOUS RELEASES

DISCHARGE PATHWAYS	10/1/79 to 10/31/79	11/1/79 to 11/30/79	12/1/79 to 12/31/79	4th Quarter 10/1/79 to 12/31/79
I. Liquid Released:				
a) Discharge less Tritium				
1) Concentration (Ci/cc)	1.25E-9			
2) Total Activity (Ci)	9.29E-3			
b) Iodine-131 Released				
1) Concentration (Ci/cc)	4.89E-11			
2) Total Activity (Ci)	3.62E-4			
c) Tritium Released:				
1) Concentration (Ci/cc)	7.61E-7			
2) Total Activity (Ci)	5.64E0			
d) MDCT Flow For Month (cc)	7.41E-12			
II. Airborne Iodine Released:				
a) Quarterly Release Rate (Ci/sec)	<MDA			
b) Total Iodine-131 released (Ci)	<MDA			
III. Noble Gases Released:				
a) Quarterly Release Rate (Ci/sec)	9.51E-6			
b) Total Noble Gases released (Ci)	75			

FOOTNOTES:

a) Concentrations are based upon actual MDCT flows. These are concentrations in the effluent averaged over the period.

TABLE 1
LIQUID RADIONUCLIDE DISCHARGES
FROM UNIT 1 BY ISOTOPE

Radionuclide	1/1/79 - 3/27/79 Activity (Ci)
^{3}H	2.54E+1
^{51}Cr	1.65E-3
^{54}Mn	3.36E-4
^{58}Co	2.13E-2
^{59}Fe	1.33E-4
^{60}Co	1.19E-3
^{65}Zn	3.94E-5
^{95}Nb	1.43E-3
^{95}Zr	7.71E-5
^{97}Zr	8.88E-5
^{99}Mo	8.56E-6
^{103}Ru	7.37E-5
^{110}Ag	8.32E-4
^{122}Sb	5.78E-5
^{124}Sb	3.77E-5
^{131}I	2.54E-4
^{131m}Xe	2.60E-5
^{132}I	—
^{133}I	—
^{133m}Xe	2.60E-5
^{133}Xe	9.95E-3
^{134}Cs	3.21E-3
^{136}Cs	1.22E-5
^{137}Cs	4.55E-3
^{140}Ba	2.88E-5
^{140}La	3.94E-4

TABLE 2
LIQUID RADIONUCLIDE DISCHARGES FROM UNIT 2 BY ISOTOPE

Radionuclide	1/1/79 - 3/27/79 Activity (Ci)
^{3}H	7.81E+1
^{24}Na	1.82E-2
^{41}Ar	1.19E-5
^{51}Cr	2.10E-3
^{54}Mn	1.13E-2
^{58}Co	2.11E-1
^{59}Co	2.29E-4
^{59}Fe	1.39E-3
^{60}Co	3.88E-3
^{95}Nb	4.2E-4
^{95}Zr	1.59E-4
^{99}Mo	3.85E-5
^{103}Ru	2.10E-4
^{110}Ag	1.07E-3
^{110m}Ag	1.98E-4
^{122}Sb	1.01E-4
^{124}Sb	9.26E-5
^{131}I	8.82E-4
^{133}I	6.92E-5
^{133}Xe	3.13E-2
^{133m}Xe	1.34E-4
^{134}Cs	1.94E-3
^{135}Xe	3.89E-4
^{137}Cs	2.18E-3
^{140}La	6.98E-4
^{187}W	3.43E-4

TABLE 3
LIQUID RADIONUCLIDE DISCHARGES
FROM UNITS 1 AND 2 BY ISOTOPE

Radionuclide	1/1/79 - 3/27/79 Activity (Ci)
^{3}H	1.04E+2
^{24}Na	1.82E-2
^{41}Ar	1.19E-5
^{51}Cr	3.75E-3
^{54}Mn	1.16E-2
^{58}Co	2.32E-1
^{59}Co	2.29E-4
^{59}Fe	1.52E-3
^{60}Co	5.07E-3
^{65}Zn	3.94E-5
^{95}Nb	1.85E-3
^{95}Zn	2.36E-4
^{97}Zr	8.88E-5
^{99}Mo	4.71E-5
^{103}Ru	2.84E-4
^{110}Ag	1.9E-3
^{110m}Ag	1.98E-4
^{122}Sb	1.59E-4
^{124}Sb	1.3E-4
^{131}I	3.47E-4
^{131m}Xe	2.60E-5
^{133}I	6.92E-5
^{133}Xe	4.13E-2
^{133m}Xe	1.6E-4
^{134}Cs	5.15E-3
^{135}Xe	3.89E-4
^{136}Cs	1.22E-5
^{137}Cs	6.73E-3
^{140}Ba	2.88E-5
^{140}La	1.09E-3
^{187}W	3.43E-4

TABLE 4

LIQUID RADIONUCLIDE DISCHARGES
By ISOTOPE

RADIONUCLIDE	3/28/79 - 4/30/79 Activity (Ci)	5/1/79 - 5/31/79 Activity (Ci)
^{3}H	8.49E0	5.38E 0
^{32}P	1.1E-3	4.1E-3
^{51}Cr	3.56E-4	9.43E-4
^{54}Mn	3.75E-4	1.21E-4
^{58}Co	2.15E-2	6.29E-3
^{60}Co	4.60E-3	1.23E-3
^{89}Sr	1.38E 0	1.53E-1
^{90}Sr	3.32E-2	9.16E-3
^{91}Sr	-	4.51E-4
^{95}Nb	1.79E-4	4.49E-4
^{95}Zr	4.92E-5	5.58E-5
^{110m}Ag	1.14E-3	7.63E-4
^{122}Sb	4.03E-5	-
^{131}I*	2.53E-1	1.72E-2
^{132}I	2.98E-3	-
^{133}I	1.23E-4	-
^{131m}Xe	-	6.11E-4
^{133}Xe	1.12E-2	6.27E-5
^{134}Cs	1.28E-3	2.19E-3
^{136}Cs	-	1.43E-3
^{137}Cs	5.39E-3	4.00E-3
^{140}Ba	4.23E-4	3.95E-3
^{140}La	1.09E-3	4.24E-3
^{65}Zn	-	1.05E-2
Total less ^{3}H	1.72E0	2.21E-1

* ^{131}I is the only radionuclide of significance released to the river from Unit II accident of 3/28/79. Other radionuclides came primarily from Unit I.

TABLE 4

LIQUID RADIONUCLIDE DISCHARGES By ISOTOPE

RADIONUCLIDE	6/1/79 - 6/30/79 Activity (Ci)	7/1/79 - 7/31/79 Activity (Ci)
^{3}H	3.04E0	5.04E0
^{32}P	6.5E-4	-
^{54}Mn	7.69E-5	2.37E-4
^{58}Co	1.57E-3	1.87E-3
^{60}Co	6.81E-4	1.30E-3
^{89}Sr	1.45E-1	1.04E-2
^{90}Sr	9.25E-3	5.66E-4
^{95}Nb	3.83E-4	2.46E-4
^{95}Zr	-	3.05E-5
^{97}Zr	-	1.21E-5
^{103}Ru	4.41E-4	1.63E-4
^{110}Ag	7.79E-5	-
^{110m}Ag	1.22E-4	6.34E-4
^{122}Sb	-	-
^{124}Sb	-	5.37E-6
^{125}Sb	-	-
^{126}Sb	2.95E-5	-
^{131}I*	3.70E-3	3.2E-3
^{133m}Xe	9.60E-4	-
^{134}Cs	1.70E-3	5.30E-3
^{137}Cs	1.41E-2	5.16E-2
^{140}Ba	2.74E-3	8.08E-4
^{140}La	6.79E-3	1.93E-3
^{141}Ce	2.76E-5	1.63E-5
^{144}Ce	-	1.62E-4
Total less ^{3}H	1.88E-1	7.85E-2

* ^{131}I is the only radionuclide of significance released to the river from Unit II accident of 3/28/79 Other radionuclides came primarily from Unit I.

TABLE 4

LIQUID RADIONUCLIDE DISCHARGE By ISOTOPE

RADIONUCLIDE	8/1/79 - 8/31/79 Activity (Ci)	9/1/79 - 9/30/79 Activity (Ci)
^{3}H	2,27E0	2,55E0
^{32}P	2,96E-4	3.52E-4
^{54}Mn	6.40E-5	2,14E-5
^{58}Co	1.24E-3	1.81E-4
^{60}Co	5.88E-4	2,16E-4
^{89}Sr	1.70E-3	1,17E-3
^{90}Sr	1.27E-4	1,39E-4
^{95}Nb	5,87E-4	1,60E-5
^{95}Zr	-	-
^{97}Zr	-	-
^{103}Ru	-	6,28E-5
^{110}Ag	-	-
^{110m}Ag	4,95E-4	1.00E-4
^{122}Sb	1,84E-4	2.02E-4
^{124}Sb	7,05E-5	-
^{125}Sb	5.30E-4	2,10E-3
^{126}Sb	-	1,02E-5
^{131}I	6.46E-4	6,59E-4
^{131m}Xe	-	-
^{134}Cs	2,46E-3	2,13E-3
^{136}Cs	1,93E-5	-
^{137}Cs	9,06E-3	1,03E-2
^{140}Ba	2,53E-4	-
^{140}La	5,36E-4	1,23E-4
^{141}Ce	3,55E-6	-
^{144}Ce	8.71E-5	-
Total less ^{3}H	1.89E-2	1,78E-2

TABLE (4)

LIQUID RADIONUCLIDE DISCHARGE By ISOTOPE

RADIONUCLIDE	10/1/79-10/31/79 Activity (Ci)	11/1/79-11/30/79 Activity (Ci)
^{3}H	5.64E0	
^{32}P	2.77E-4	
^{54}Mn	6.96E-5	
^{57}Co	--	
^{58}Co	1.04E-3	
^{60}Co	8.37E-4	
^{89}Sr	4.56E-4	
^{90}Sr	8.21E-5	
^{95}Nb	5.57E-5	
^{110m}Ag	2.34E-4	
^{122}Sb	6.85E-5	
^{125}Sb	3.15E-5	
^{131}I	3.62E-4	
^{134}Cs	2.01E-3	
^{137}Cs	3.77E-3	
Total less ^{3}H	9.29E-3	

TABLE 5

VOLUME OF LIQUID WASTE DISCHARGE 1/1/79 to 3/27/79

UNIT I - 293,262 gallons

UNIT II - 238,308 gallons

TABLE 6

SUMMARY OF LIQUID VOLUME DISCHARGES (GALLONS)

	3/28/79-4/30/79	5/1/79-5/30/79	6/1/79-6/30/79	7/1/79-7/31/79	8/1/79-8/31/79	9/1/79-9/30/79
IWTS	2,776,600	2,348.910	1,776,070	1,821,030	1,801,030	2,429,620
IWFS	616,110	505,820	682,320	733,150	625,140	497,740
WECST (A&B)	93,903	112,229	41,888	125,827	56,800	58,048
UNIT I Sec. Neut.	860,037	904,694	802,475	881.262	829,303	730,819
MDCT = TOTAL - (IWTS + IWFS + WECST (A&B) + UNIT I Sec. Neut.)						
TOTAL	2,793,000,000	2,017.600,000	1,745.100,000	1,848,800,000	1,875,600,000	2,019,900,000

TOTALS FOR ACCIDENT TO 9/30/79

IWTS = 12,953,260 gallons

IWFS = 3,660,280 gallons

WECST = 488,695 gallons

Sec. Neut = 5,008,590 gallons

Total Effluent = 1.23E 10

TABLE 6

SUMMARY OF LIQUID VOLUME DISCHARGES
(GALLONS)

	10/1/79-10/31/79	11/1/79-11/30/79	12/1/79-12/31/79
IWTS	2,304,070		
IWFS	558,320		
WECST (A&B)	80,189		
Unit I Sec. Neut.	752,893		
MDCT = TOTAL - (IWTS + IWFS + WECST (A&B) + UNIT I Sec. Neut.)			
TOTALS	1,957,600,000		

TOTALS FOR ACCIDENT TO 10/30/79

IWTS = 1,526E 7 gallons

IWFS = 4.219E 6 gallons

WECST = 5.689E 5 gallons

Sec. Neut. = 5.761E 6 gallons

Total Effluent = 1.426E 10

TABLE 7

SUSQUEHANNA RIVER FLOW RATES

	1st Quarter		
January	8.9 E+4 cfs	or	5.34 E+6 cfm
February	3.43 E+4 cfs	or	2.06 E+6 cfm
March	1.20 E+5 cfs	or	7.2 E+6 cfm
Average	8.11 E+4 cfs		4.87 E+6 cfm
	2nd Quarter		
April	5.7 E+4 cfs	or	3.42 E+6 cfm
May	3.86 E+4 cfs	or	2.32 E+6 cfm
June	2.78 E+4 cfs	or	1.67 E+6 cfm
Average	4.1 E+4 cfs		2.47 E+6 cfm
	3rd Quarter		
July	1.05 E+4 cfs	or	6.3 E+5 cfm
August*	2.0 E+4 cfs	or	1.0 E+5 cfm
September *	2.34 E+4 cfs	or	1.41 E+6 cfm
	4th Quarter		
October *	3.90 E+4 cfs	or	2.34 E+6 cfm
November*	4.38 E+4 cfs	or	2.63 E+6 cfm

*Estimate by U. S. Geological Survey

TABLE 8

THI LIQUID ^3H DISCHARGE FOR 1979

	UNIT I					UNIT I[illegible]	UNIT I & II				
	WECST - TANK 11A & 11B				IWFS IWTS & SEC NEUT.	WETT [illegible] A&9B NEUT. 8A&8B TANK	IOSI COMPOSITE				TOTAL DISCHARGED TO RIVER FOR MONTH
	VOLUME DISCHARGED	Composite Sample		Sum of Releases (each Release Sampled)			VOLUME DISCHARGED			Sample Dates	
	cc x 10^8	μCi/cc	Ci	Ci	Ci	Ci	cc x 10^{12}	μCi/cc	Ci		Ci
JAN.	3.2	2.61E-2	8.35	8.52	These tanks were not analyzed for ^3H prior to August, 1979 **	1.36E1	6.97	3.1E-7	2.16	1/3-1/31	22.1
FEB.	3.73	1.96E-2	7.31	7.85		2.87E1	6.25	1.54E-6	9.63	2/7-2/28	36.6
MAR.	4.18	*	*	10.7		3.49E1	7.23 1.09	1.93E-5 5.0E-7	140 0.55	3/7-3/21 3/29 only	140 0.55
APR.	3.55	1.89E-2	6.71	5.47		No liquid release from Unit II since the accident on 3/28/79	9.23	8.1E-7	7.48	4/1-4/30	7.80 ***
MAY	4.25	7.02E-3	2.98	5.38			7.64	6.2E-7	4.74	5/1-6/1	5.38
JUNE	1.59	4.23E-3	0.67	0.69			6.61	4.6E-7	3.04	6/1-6/30	3.04
JULY	4.72	7.72E-3	3.64	4.23			7.00	7.2E-7	5.04	6/30-8/1	5.04
AUG.	2.11	6.15E-3	1.30	1.52	<MDA		7.10	3.2E-7	2.27	8/1-8/31	2.27
SEPT.	2.20	1.16E-2	2.55	1.75	<MDA		7.65	1.5E-7	1.15	8/31-9/30	2.55
OCT.	3.06	1.73E-2	5.26	5.64	<MDA		7.41	6.3E-7	4.67	9/30	5.64
NOV.											
DEC.											

* Sample lost due to accident on March 28, 1979

** These releases were monitored via the IOSI composite.

*** Calculated using daily concentrations obtained from IOSI (Radiological Environmental Monitoring Program - [illegible]MP) data.

TABLE 9

THE LIQUID RADIOSTRONTIUM DISCHARGE FOR 1979

	UNIT I					UNIT II									
	WECST - Composite Tank 11A & 11B					WETT - Composite Tank 9A & 9B					NEUT TEST TANK 8A & 8B				
	TANK VOLUME DISCHARGED cc x 10^8	^{89}Sr		^{90}Sr		TANK VOLUME DISCHARGED cc x 10^8	^{89}Sr		^{90}Sr		TANK VOLUME DISCHARGED cc x 10^8	^{89}Sr		^{90}Sr	
		µCi/cc	Ci	µCi/cc	Ci		µCi/cc	Ci	µCi/cc	Ci		µCi/cc	Ci	µCi/cc	Ci
JAN.	3.2	8.3E-8	2.66E-5	7.8E-9	2.50E-6	0.401	<MDA	<MDA	<MDA	<MDA	2.31	3.5E-7	8.09E-5	8.4E-8	1.96E-5
FEB.	3.73	5.1E-7	1.90E-4	4.8E-8	1.79E-5	3.49	4.3E-7	1.50E-4	6.3E-8	2.20E-5	1.68	5.6E-8	9.41E-6	<MDA	—
MAR.	4.18	*	*	*	*	3.89	*	—	*	—	2.07	*	—	*	—
APR.	3.55	3.8E-7	1.35E-4	2.2E-8	7.80E-6	None	—	—	—	—	None	—	—	—	—
MAY	4.25	9.1E-6	3.87E-3	2.2E-7	9.35E-5	None	—	—	—	—	None	—	—	—	—
JUNE	1.59	4.8E-5	7.61E-3	1.6E-6	2.54E-4	None	—	—	—	—	None	—	—	—	—
JULY	4.72	2.2E-5	1.04E-2	1.2E-6	5.66E-4	None	—	—	—	—	None	—	—	—	—
AUG.	2.11	8.0E-6	1.70E-3	6.0E-7	1.27E-4	None	—	—	—	—	None	—	—	—	—
SEPT.	2.20	5.3E-6	1.17E-3	6.3E-7	1.39E-4	None	—	—	—	—	None	—	—	—	—
OCT.	3.04	1.5E-6	4.56E-4	2.7E-7	8.21E-5	None	—	—	—	—	None	—	—	—	—
NOV.															
DEC.															

* Sample lost due to accident on March 28, 1979.

** Special sample taken for period of 3/28/79 - 3/31/79.

Maximum MDA for 1979: ^{89}Sr = 2E-8 µCi/cc; ^{90}Sr = 5E-9 µCi/cc

TABLE 9 (Continued)

THI LIQUID RADIOSTRONTIUM DISCHARGE FOR 1979

UNIT I & II

	10-S-1 = RML-7 - Composite					Total Ci for the Month		
	VOLUME DISCHARGED	^{89}Sr		^{90}Sr		^{89}Sr	^{90}Sr	$^{89}Sr + ^{90}Sr$
	cc x 10^{12}	μCi/cc	Ci	μCi/cc	Ci	Ci	Ci	Ci
JAN.	6.97	Analysis for Sr was started on March 28, 1979				1.08E-4	2.19E-5	1.29E-4
FEB.	6.25					3.49E-4	3.99E-5	3.89E-4
MAR.	8.32	<MDA **	—	<MDA **	—	* <MDA**	* <MDA**	* <MDA**
APR.	9.23	1.5E-7	1.38E 0	3.6E-9	3.32E-2	1.38E0	3.32E-2	1.41E0
MAY	7.64	2.0E-8	1.53E-1	1.2E-9	9.16E-3	1.53E-1	9.16E-3	1.62E-1
JUNE	6.61	2.2E-8	1.45E-1	1.4E-9	9.25E-3	1.45E-1	9.25E-3	1.54E-1
JULY	7.00	<MDA	—	<MDA	—	1.04E-2	5.66E-4	1.10E-2
AUG.	7.10	<MDA	—	<MDA	—	1.70E-3	1.27E-4	1.83E-3
SEPT.	7.65	<MDA	—	<MDA	—	1.17E-3	1.39E-4	1.31E-3
OCT.	7.41	<3E-9	—	<6E-10	—	4.56E-4	8.21E-5	5.38E-4
NOV.								
DEC.								

* Sample lost due to accident on March 28, 1979.

** Special sample taken for period of 3/28/79 - 3/31/79.

Maximum MDA for 1979 ^{89}Sr = 2E-8 μci/cc

^{90}Sr = 5E-9 μci/cc

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water
μCi/ℓ

TRITIUM

Map Location	Page 6 of ___	Sample	5/25	5/26	5/27	5/28	5/29	5/30	5/31	6/1	6/1-6/7	6/8-6/14	6/15-6/21	6/22-6/28
27	Swatara Creek	1C3	270	160	210	130	190	<120	<130		150	160	180	160
27	Swatara Creek	1C3Q	<183	<197	<197	< 266	196	<179	<268		<260	<187	<259	<255
30	Brunner Island	8E1	130	170	200	160	220	<100	100	130	140	-	160	110
30	Brunner Island	8E1*	100	200	200	110	230	120	<100	-	120	150	130	<100
	Discharge Pit	10S1	1510	950	240	550	1530	540	1550					
	Discharge Pit	10S1Q	1100	381	<182	472	925	142	1040					
38	Columbia Water Plant	7G1	190	170	160	210	210	<130	100	160	160	160	150	120
38	Columbia Water Plant	7G1*										180	160	130
38	Columbia Water Plant	7G1Q	<267	<197	<280	<266	<180	<179	<268		<260	<187	<259	<255
35	Steelton Water Works	15F1*	100	170	190	120	150	<130	<130		160	170	160 *	140 *
35	Steelton Water Works	15F1Q*	<267	<197	<280	<266	<180	<179	<268		<260	<187	259*	<249*
48	YHGS	8C2	190	150	180	120	150	110	<120	130	110	130	180	100
48	YHGS	8C2Q	<267	<197	<280	<266	<180	<179	<268		<260	<187	<259	<255
	Lancaster	7G3*											160	120
	York	9G2*	210	140	160	120	190	290	<100		130	110	180	180
	York	9G2Q*	<267	<197	<280	<266	<180	192	<268		<260	<276	<259	<255
	Wrightsville	7G2	140	160	160	160	210	100	170		140	100	140	150
	Wrightsville	7G2*	<100	150	150	160	·100	<120	<100		140	130	100	160

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water pCi/l TRITIUM

Map Location	Page 3 of ___	Sample	6/29-7/5	7/6-7/12	7/13-7/19	7/23-7/26	7/27-8/2	8/3-8/9	8/10-8/16	8/16-8/23	8/23-8/30	8/30-9/6	9/7-9/13	9/13-9/20
27	Swatara Creek	1C3	<120	160	200	290	230	180	250	200	230	300	410	170
27	Swatara Creek	1C3Q	<195	147	143	160	<135	157	<177	<283	<170	<273	<136	<304
30	Brunner Island	8E1	<130	180	240	180	240	180	160	210	180	190	130	180
30	Brunner Island	8E1*	120	200	210	210	230	120	300	220	250	320	110	160
38	Columbia Water Plant	7G1	110	210	270	250	240	150	220	200	240	230	140	160
38	Columbia Water Plant	7G1*	<150	270	220	190	160	210	250	210	220	190	160	190
38	Columbia Water Plant	7G1Q	<134	177	<177	129	<185	<171	<177	<283	<170	<278	<128	<304
35	Steelton Water Works	15F1*	100*	280	210	250	210	180	300	210	260	160	130	170
35	Steelton Water Works	15F1Q*	<184*	<173	<177	<177	<185	<171	<177	<283	<170	<278	<128	<209
48	YNGS	8C2	120	250	220	240	290	190	200	210	230	200	150	230
48	YNGS	8C2Q	122	155	<177	<177	<185	<171	146	<283	<170	<278	<128	93.8
	Lancaster	7G3*	230	220	230	230	240	170	140	180	170	170	170	160
	York	9G2*	<150	220	220	170	220	270	230	180	220	220	180	170
	York	9G2Q*	148	<178	117	<177	<185	<283	<177	<283	<170	<278	<128	<125
	Wrightsville	7G2	<110											
	Wrightsville	7G2*	160	150	160	170	200	210	240	210	170	200	170	210

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water
pCi/l

TRITIUM

Page 8 of ___	Sample	9/21-9/27	9/28-10/4	10/4-10/11	10/11-10/18	10/19-10/25	10/25-11/1	11/2-11/8	11/9-11/15	11/15-11/22	11/22-11/29		
Swatara Creek	1C3		120	170	180	200	200	190					
Swatara Creek	1C3Q	<125	<288	<296	<296	<205	<295	<229	494	<222	<261		
Brunner Island	8E1		140	160	170	190	<120	310					
Brunner Island	8E1*		120	160	160	190	190	150					
Columbia Water Plant	7G1		130	150	180	230	220	120					
Columbia Water Plant	7G1*		160	150	180	230	200	130					
Columbia Water Plant	7G1Q	<125	<288	<296	<296	<205	<295	<229	<291	<222	<248		
Steelton Water Works	15F1		110	210	170	170	190	100					
Steelton Water Works	15F1Q	79.4	<288	<296	<296	<205	<295	<229	<291	<222	<248		
YHGS	8C2		170	160		200	170	<110					
YHGS	8C2Q	<125	<288	<296	<296	<205	<295	<229	<291	<222	<248		
Lancaster	7G3*		180	190	210	180	240	170					
York	9G2*		140	180	140	210	200	170					
York	9G2Q*	<125	<245	<296	<296	<205	<295	<229	<291	<222	<248		
Intake	13S2						200	270					
Wrightsville	7G2*		100	180	180	170	160	130					

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water
pCi/l

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	Sample	11/9	11/10	11/11	11/12	11/13	11/14	11/15	11/16	11/17	11/18	11/19	11/20
Swatara Creek	1C3	<0.3	<0.3	<0.3	<0.3	<0.3	<0.4	<0.4	<0.3	<0.3	<0.4	<0.3	<0.4
Swatara Creek	1C3Q	<0.5	<0.3	<0.5	<0.4	<0.3	<0.4	<0.5	<0.4	<0.4	<0.4	<0.3	<0.2
Brunner Island	8E1	<0.1	<0.3	<0.4	<0.3	<0.3	<0.3	<0.3	<0.5	<0.4	<0.5	<0.4	<0.4
Brunner Island	8E1*	<0.2	<0.3	<0.2	<0.3	<0.3	<0.3	<0.2	<0.3	<0.4	<0.5	<0.3	<0.4
Columbia Water Plant	7G1	<0.2	<0.3	<0.2	<0.4	<0.3	0.5	0.9	<0.3	<0.3	<0.3	<0.5	<0.4
Columbia Water Plant	7G1*	<0.3	<0.3	<0.5	<0.2	<0.4	<0.4	<0.3	<0.4	<0.3	<0.4	<0.5	<0.5
Columbia Water Plant	7G1Q	<0.4	<0.4	<0.4	<0.4	<0.5	0.8	1.2	<0.5	<0.4	<0.4	<0.4	<0.4
Steelton Water Works	15F1	<0.4	<0.5	<0.3	<0.3	<0.2	1.0	<0.2	<0.4	<0.4	<0.5	<0.4	<0.4
Steelton Water Works	15F1Q	<0.4	<0.3	<0.4	<0.4	<0.4	<0.4	<0.3	<0.4	<0.4	<0.4	<0.4	<0.4
YHGS	8C2	<0.4	<0.3	<0.3	<0.3	<0.5	<0.3	<0.3	<0.5	<0.5	<0.6	<0.3	<0.4
YHGS	8C2Q	<0.3	<0.3	<0.4	<0.4	<0.5	<0.3	<0.3	<0.4	<0.4	<0.3	<0.5	<0.3
Lancaster	7G3*	<0.4	<0.5	<0.2	<0.4	<0.3	<0.4	1.1	<0.5	<0.3	<0.5	<0.5	<0.4
York	9G2*	<0.5	<0.3	<0.3	<0.3	<0.2	<0.4	<0.4	<0.3	<0.4	<0.5	<0.3	<0.5
York	9G2Q*	<0.4	<0.3	<0.4	<0.3	<0.3	<0.4	<0.3	<0.4	<0.3	<0.3	<0.4	<0.5
Intake	13S2	<0.3	<0.4	<0.3	<0.3	1.6	3.2	<0.3	<0.3	<0.4	<0.4	<0.6	<0.3
Wrightsville	7G2*	<0.4	<0.4	<0.5	<0.3	<0.3	<0.4	<0.5	<0.5	<0.4	<0.5	<0.5	<0.5

[illegible] 12/18

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water
pCi/l

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	Sample	11/21	11/22	11/23	11/24	11/25	11/26	11/27	11/28	11/29	11/30	12/1	12/2
Swatara Creek	1C3	<0.3	<0.6	<0.6	<0.4	0.4	<0.4	<0.3	<0.4	<0.3	<0.3	<0.3	<0.4
Swatara Creek	1C3Q	<0.5											
Brunner Island	8E1	<0.4	<0.3	<0.4	<0.4	<0.5	<0.3	<0.4	<0.4	<0.3	<0.4	<0.4	<0.3
Brunner Island	8E1*	<0.3	<0.4	<0.4	<0.3	<0.5	<0.4	<0.3	<0.2	<0.4	<0.2	<0.3	<0.4
Columbia Water Plant	7G1	<0.4	<0.4	<0.4	<0.4	<0.5	<0.4	<0.5	<0.4	<0.4	<0.4	<0.3	<0.4
Columbia Water Plant	7G1*	<0.4	<0.4	<0.3	<0.4	<0.5	<0.3	<0.3	<0.4	<0.3	<0.5	<0.5	<0.4
Columbia Water Plant	7G1Q	<0.3											
Steelton Water Works	15F1	<0.4	<0.4	<0.4	<0.5	<0.4	<0.4	<0.4	<0.3	<0.4	<0.3	<0.3	<0.4
Steelton Water Works	15F1Q	<0.3											
YNGS	8C2	<0.4	<0.3	<0.4	<0.5	<0.4	<0.3	<0.4	<0.3	<0.3	<0.3	<0.4	<0.4
YNGS	8C2Q	<0.2											
Lancaster	7G3*	<0.4	<0.4	<0.5	<0.3	<0.4	<0.3	<0.3	<0.4	<0.3	<0.4	<0.3	<0.3
York	9G2*	<0.5	<0.4	<0.3	<0.3	< 0.3	<0.4	<0.3	<0.3	<0.3	<0.4	<0.3	
York	9G2Q*	<0.3											
Intake	13S2	<0.4	<0.5	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.3	<0.2	<0.4	<0.3
Wrightsville	7G2*	<0.4	<0.4	<0.4	<0.4	<0.4	<0.5	<0.5	<0.4	<0.3	<0.3	<0.3	<0.4

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water
pCi/ℓ

IODINE - 131

Page 22 of ___	Sample	12/3	12/4	12/5	12/6	12/7
Swatara Creek	1C3	<0.3	<0.4	< 0.4	<0.3	<0.5
Swatara Creek	1C3Q					
Brunner Island	8E1	<0.4	<0.4	<0.5	<0.3	<0.3
Brunner Island	8E1*	<0.2	<0.4	<0.4	<0.4	<0.3
Columbia Water Plant	7G1	<0.3	<0.5	<0.3	<0.5	<0.4
Columbia Water Plant	7G1*	<0.4	<0.3	<0.4	<0.4	<0.4
Columbia Water Plant	7G1Q					
Steelton Water Works	15F1	<0.3	<0.3	<0.3	<0.4	<0.5
Steelton Water Works	15F1Q					
YHGS	8C2	<0.2	<0.4	<0.4	<0.4	<0.4
YHGS	8C2Q					
Lancaster	7G3*	<0.3	<0.3	<0.2	<0.4	<0.5
York	9G2*	<0.3	<0.4	<0.3	<0.5	<0.3
York	9G2Q*					
Intake	13S2	<0.3	<0.3	<0.4	<0.4	
[illegible]	7G2*	<0.3	<0.3	<0.3	<0.4	<0.3

Rev 1

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2,
*Indicates Finished (Treated) Water 7G3, &15F1.

Water
pCi/ℓ

GROSS BETA

Map Location	Page 7 of ___	Sample	7/6 - 7/12	7/13 - 7/19	7/20 - 7/26	7/27 - 8/2	8/3 - 8/9	8/10 - 8/16	8/16- 8/23	8/24- 8/30	8/30- 9/6	9/6- 9/13	9/13- 9/20	9/21- 9/27
27	Swatara Creek	1C3	4.4	6.8	4.6	4.2	3.3	4.5	3.7	5.6	14.0	5.4	4.6	3.6
27	Swatara Creek	1C3Q	< 3.2	4.2	3.3	4.2	3.0	4.8	2.7	4.4	17.1	3.4	<3.7	4.7
30	Brunner Island	8E1	4.2	8.1	8.0	4.3	6.6	2.8	3.0	4.9	6.6	9.7	4.5	5.5
30	Brunner Island	8E1*	1.7	2.0	2.0	2.4	5.9	2.6	1.3	2.6	1.9	2.8	<1.0	3.2
38	Columbia Water Plant	7G1	3.4	3.2	8.4	4.2	3.1	5.2	5.0	5.1	11.0	9.7	3.6	2.9
38	Columbia Water Plant	7G1*	2.3	2.4	2.8	3.3	2.5	3.5	2.3	3.1	2.6	3.1	3.8	3.2
38	Columbia Water Plant	7G1Q	2.6	2.3	3.4	12.8	10.4	16.3	3.4	5.0	16.3	<3.3	<3.7	4.5
35	Steelton Water Works	15F1	2.5	2.1	4.6	2.5	2.8	3.2	3.4	2.3	2.7	3.9	2.4	1.9
35	Steelton Water Works	15F1Q	< 3.2	1.8	< 2.8	2.5	7.9	2.3	<3.3	<3.2	3.0	3.2	<3.7	<3.3
48	YHGS	8C2	3.1	1.6	3.2	2.3	3.1	2.5	2.3	2.6	2.5	3.5	2.2	2.6
48	YHGS	8C2Q	< 3.2	2.0	2.1	2.3	3.2	< 3.2	2.2	3.0	2.2	<3.3	<3.6	2.8
	Lancaster	7G3*	3.5	3.1	4.5	8.5	3.8	3.1	3.5	3.7	3.4	4.5	4.1	5.0
	York	9G2*	3.6	3.1	2.0	2.3	3.9	3.3	2.6	2.7	2.5	2.9	3.7	3.9
	York	9G2Q*	< 3.2	2.1	4.5	< 3.0	3.7	< 3.2	<3.3	2.4	<3.3	3.1	<3.6	<3.3
	Wrightsville	7G2												
	Wrightsville	7G2*	2.8	2.2	2.5	2.6	3.3	3.0	2.7	3.0	3.3	4.1	3.3	4.5

Note: Composite samples are taken at 8E1, 8C2, 7G1, 7G2, 7G3, &15F1.
*Indicates Finished (Treated) Water

Water
pCi/l

GROSS BETA

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	Sample	9/28-10/4	10/4-10/11	10/11-10/18	10/18-10/25	10/25-11/1	11/1-11/8	11/8-11/15	11/15-11/22				
Swatara Creek	1C3	3.7	2.3	2.7	3.2	3.3	2.5	3.4					
Swatara Creek	1C3Q	6.5	6.2	2.8	2.4	7.8	<3.2	5.1					
Brunner Island	8E1	6.1	3.2	2.7	4.9	2.4	3.0	3.0					
Brunner Island	8E1*	2.6	1.8	<1.0	2.0	1.5	2.3	<2.0					
Columbia Water Plant	7G1	5.5	3.0	3.7	3.6	2.9	5.5	4.3					
Columbia Water Plant	7G1*	3.4	4.5	2.2	2.9	3.2	2.5	2.0					
Columbia Water Plant	7G1Q	17.2	21.6		3.0		14.2	6.9					
Steelton Water Works	15F1	2.4	3.3	2.5	4.2	1.4	5.4	2.1					
Steelton Water Works	15F1Q	5.4	13.1	3.0	<3.1	4.3	2.7	7.5					
YHGS	8C2	2.1	2.9	3.4	1.9	1.3	2.8	<2.0					
YHGS	8C2Q	2.1	7.4	2.3	3.1	5.7	2.6	3.2					
Lancaster	7G3*	2.1	3.3	2.0	4.1	3.5	3.1	2.6					
York	9G2*	3.1	4.0	4.3	2.7	2.5	2.3	3.8					
York	9G2Q*	2.6	<3.4	2.0	2.1	4.7	<3.2	7.7					
Intake	13S2					3.5	3.1	2.4					
Wrightsville	7G2*	3.2	3.6	3.5	5.1	4.6	4.1	4.1					

VI. RECOVERY PLANNING

During the past quarter the "Summary Technical Plan for TMI-II Decontamination and Defueling" was issued. More detailed and comprehensive plans are under preparation covering TMI-II decontamination and defueling. The technical plans are being developed from the many and various topical studies underway for recovery. The technical plans will contain a summary of the sequence of events considered necessary to accomplish specific recovery objectives, including the options and alternatives being evaluated.

VII. SPECIAL PROJECTS

Special Projects programs consist of three areas:

1. Containment surveillance experimental program
2. Reactor Building purge program
3. Reactor Building reentry program

Containment Building Surveillance Experimental Program: The major experiment conducted during the fourth quarter of 1979 was the cutting of the hole through penetration R-626. This hole was used to insert a camera and radiation monitors and to take air samples and swipe surveys. Additionally, a TLD probe was inserted through the hole and radiation readings on TLD's, film badges and dosimeters were obtained. The major results from the experiments through R-626 are as follows:

Visual inspection of the containment building above the operating floor through R-626 showed no significant damage to the containment building.

Condensation was noted in the building and dust or dirt was noted on the floor.

No other significant findings were obtained using the camera.

Radiation readings inside penetration R-626 have varied due to problems using standard radiation monitoring equipment in the Krypton-85 atmosphere. The initial readings have been determined to be in error (high) due to interference in the instruments by the Krypton-85 cloud. These initial readings showed constant beta readings of 390 rad/hr regardless of position inside the building and gamma readings of 2.7 to 4.6 R/hr.

Subsequent readings taken inside penetration R-626 have invalidated the initial gamma readings and Metropolitan Edison now considers the proper reading to be 250 to 400 mr/hr general area gamma radiation readings inside the reactor building.

Although beta radiation data is still inconclusive, the best estimate as determined through penetration R-626 is 300 to 400 rad/hr general area beta radiation levels.

Air sample data taken through penetration R-626 is roughly the same as sample data taken through the normal sample path, i.e. through HPR-227.

Particulate levels taken through penetration R-626 are in the range of 10^{-9} to 10^{-10} microcuries per millileter. The major particulate isotope determined was Cesium 134 and Cesium 137.

There was no detectable iodine in the samples.

Krypton-85 concentrations in air samples from penetration R-626 were lower than those taken from other sources. Readings of Krypton-85 ranged from .4 to .7 microcuries per millileter. These readings are lower than the current values being obtained through the normal sample path of .8 to 1 microcurie per millileter of Krypton-85.

Metropolitan Edison is continuing its weekly sampling program through panel HPR-227 and intends to use HPR-227 data to control the reactor building purge program.

Swipe samples taken from the wall and penetration flange of penetration R-626 showed mainly Cesium 134 and Cesium 137 as the plateout sources. Cesium 134 was found in the range of 4 to 7 x 10^{-2} microcuries per swipe. Cesium 137 was found in the range of 2 to 4 x 10^{-1} microcuries per swipe.

Cobalt 58, Cobalt 60 and Niobium 95 were detected in the range of 10^{-4} to 10^{-5} microcuries per swipe.

Data from the TLD probe is still being evaluated and is therefore not being included in this report.

Temperature and humidity inside the reactor building were 84°F and 100%, respectively.

Metropolitan Edison has sent the flange cutout from penetration R-626 and the inlet hydrogen recombiner spoolpiece to Oak Ridge for analysis. Analysis results have not been received. As part of its initial entry program, Metropolitan Edison expects to conduct an entry into #2 personnel airlock in January. This entry will provide better radiation information on the 305' elevation of containment. This entry will require opening the outer airlock door only.

Reactor Building Purge: Metropolitan Edison submitted its request to purge the reactor building to the Nuclear Regulatory Commission on November 13th. A reactor building purge program safety analysis and environmental report accompanied this purge request. The Nuclear Regulatory Commission has asked 33 questions concerning the initial submittal and Metropolitan Edison has answered those questions in writing to the Nuclear Regulatory Commission. Procedure preparation and engineering change modifications required to support the reactor building purge are proceeding and preparations to support the reactor building purge should be completed in February.

Containment Entry: Metropolitan Edison is proceeding with plans to conduct the initial entry into the reactor containment building. Entry team training, entry procedure preparation and engineering change modifications required to support the initial entry are proceeding. Metropolitan Edison is reviewing experimental data to determine if reactor building entry prior to building purge is feasible. Metropolitan Edison expects preparations to conduct the initial reactor building entry to be completed by the end of February, 1980.

VIII. REPORTABLE OCCURRENCES

This section addresses all LER's whose period of reportability falls in the time interval subsequent to the March 28, 1979 incident. The occurrences that fall in this category have been assigned LER numbers 79-12 through 79-22. These LER's have been reported to NRC in varying degrees ranging from LER completed to not reported. All LER's that have not been previously submitted are attached to this report. Table VIII.1 is a status summary of all LER's that fall in this time interval.

TABLE VIII. 1 REPORTABLE OCCURRENCE SUMMARY

LER NO.	DESCRIPTION	DATE OF OCCURRENCE	DATE REPORTED	COMMENTS
79-12	Failure of Fire Barrier Seals	3/5/79	9/11/79	LER Transmitted via GQL #1163
79-13	Station RW discharge ΔT exceeded ETS limit	3/21/79	4/6/79	Instrumentation was faulty. ΔT remained within limits. Non reportability transmitted via GQL #0474.
79-14	TMI-II Incident	3/28/79	--------	Will be submitted in the final report.
79-15	Personnel Overexposure	3/28/79	1/15/80	Previously reported via GQL #0620. LER attached.
79-16	Overheating of fire pumps FS-P1	4/28/79	1/15/80	LER attached.
79-17	Personnel Overexposure	3/29/79	1/15/80	Previously reported via GQL #1094. LER attached.
79-18	Personnel Overexposure	3/28/79	1/15/80	Previously reported via GQL #1188 and GQL #1499. LER attached.
79-19	Overheating of fire pump FS-P1			Non reportability was determined by Plant Operations Review Committee (PORC).
79-20	Fish monitoring studies terminated	11/1/79	12/21/79	LER transmitted via GQL 1561.
79-21	Emergency diesel failed to start	12/5/79	1/15/80	LER attached.
79-22	NR pump failure	12/19/79	1/15/80	LER attached.

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0 1 | P A T M I 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | | (5)

LICENSEE CODE / LICENSE NUMBER / LICENSE TYPE / CAT

CON'T 0 1 REPORT SOURCE L (6) DOCKET NUMBER 0 5 0 0 0 3 2 0 (7) EVENT DATE 0 3 2 8 7 9 (8) REPORT DATE 0 1 1 5 8 0 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 While investigating leakage of the radwaste and ma eup system station personnel

0 3 received radiation doses in excess of the allowable quarterly limit for whole

0 4 body dose. The individual exposures were less than five man-rem. This is

0 5 reportable under 10CFR 20.405b. For further information, see Met-Ed letter

0 6 dated May 1, 1979 (GQL 0620).

0 7

0 8

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	M A (11)	X (12)	Z (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 9	—	0 1 5	/	1 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	N (23)	N (24)	Z (25)	Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 Subsequent samples of the primary coolant and area surveys around the radwaste

1 1 and makeup systems were suspended until additional precautions could be

1 2 implemented to minimize future exposures.

1 3

1 4

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	X (28)	0 0 0 (29)	Emergency Shutdown	C (31)	Survey of auxiliary building

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	N (44)	NA	

NAME OF PREPARER C. F. McPhatter PHONE: (717) 948-8552

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CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0	1	P A T M I 2 (2)	0 0 - 0 0 0 0 0 - 0 0 (3)	4 1 1 1 1 (4)	(5)
7	8	9 LICENSEE CODE 14	15 LICENSE NUMBER 25	26 LICENSE TYPE 30	57 CAT 58

CON'T

0 1	REPORT SOURCE	L (6)	0 5 0 0 0 3 2 0 (7)	0 4 2 8 7 9 (8)	0 1 1 5 8 0 (9)
7 8		60	61 DOCKET NUMBER 68	69 EVENT DATE 74	75 REPORT DATE 80

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 Fire pump FS-P2 had been restored to service but successful surveillance testing
0 3 was not completed until May 5, 1979. An overheating problem which required
0 4 rebuilding of the FS-P3 diesel occurred while FS-P2 was not available for use.
0 5 During the one day period with only two fire pumps available, TMI-II fire pump
0 6 FS-P1 overheated and was off line for a three hour period. With only one fire pump
0 7 available, temporary measures were considered but not taken since FS-P1 was restored
0 8 to operable status in three hours and FS-P2 was made available (Continuing)

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	A B (11)	D (12)	X (13)	P U M P X X (14)	(15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 9	—	0 1 6	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
G (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	N (23)	N (24)	(25)	(26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 This event occurred as the result of system leakage in excess of the jockey pump's
1 1 capacity which resulted in the diesel driven fire pumps running continuously.
1 2 A cooling system reservoir was added to FS-P3 but determined not to be needed on
1 3 FS-P1 (Unit 2). OP 2104-6.1 (Unit 2) and O.P. 1104-45 (Unit 1) have been revised
1 4 to require operator attendance during diesel runs and to outline newly (Continuing)

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	0 0 0 (29)	NA	B (31)	Operator Observation

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	Y (44)	NA	

NAME OF PREPARER T. O'Connor PHONE: (717) 948-8135

January 15, 1980

LER 79-016/03L-0

Event Description and Probably Consequences: (10)

with untested capacity. This is considered reportable under T.S. 6.9.1, equipment failure as described in T.S. action statement 3.7.10.

Cause Description and Corrective Actions: (27)

developed cooldown procedures at the termination of each run. The Unit II diesel driven pump (FS-P1) overheating was the result of several intermittant runs to raise system pressure with no cooldown performed between runs. Full operational status was restored on May 5, 1979 for all four fire pumps which is within the action statement requirements of T.S. 3.7.10.

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CONTROL BLOCK: | - | | | | | | (1)

(PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

| 0 | 1 | P A T M I 2 (2) 0 0 - 0 0 0 0 0 - 0 0 (3) 4 1 1 1 1 (4) | | (5)

LICENSEE CODE — LICENSE NUMBER — LICENSE TYPE — CAT

CON'T | 0 | 1 | REPORT SOURCE L (6) 0 5 0 0 0 3 2 0 (7) 0 3 2 9 7 9 (8) | | | | | | (9)

DOCKET NUMBER — EVENT DATE — REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

| 0 | 2 | While manipulating valves controlling coolant flow and handling laboratory
| 0 | 3 | bottles and flasks containing coolant samples station personnel received
| 0 | 4 | extremity exposures in excess of allowable limits. This is reportable under
| 0 | 5 | 10CFR 20.405b. For further information, see Met Ed letter to V. Stello
| 0 | 6 | dated August 21, 1979 (GOL 1094).
| 0 | 7 |
| 0 | 8 |

| 0 | 9 | SYSTEM CODE: Z Z (11) CAUSE CODE: D (12) CAUSE SUBCODE: Z (13) COMPONENT CODE: Z Z Z Z Z Z (14) COMP. SUBCODE: Z (15) VALVE SUBCODE: Z (16)

(17) LER/RO REPORT NUMBER: EVENT YEAR 7 9 — SEQUENTIAL REPORT NO. 0 1 7 / OCCURRENCE CODE 1 2 REPORT TYPE L — REVISION NO. 0

ACTION TAKEN C (18) FUTURE ACTION Z (19) EFFECT ON PLANT Z (20) SHUTDOWN METHOD Z (21) HOURS (22) 0 0 0 0 ATTACHMENT SUBMITTED N (23) NPRD-4 FORM SUB. N (24) PRIME COMP. SUPPLIER Z (25) COMPONENT MANUFACTURER Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

| 1 | 0 | For subsequent samples of reactor coolant, detailed coolant sampling procedures
| 1 | 1 | were developed and intensive training was given to station sampling personnel,
| 1 | 2 | shielding was properly positioned and better dosimetry coverage was provided,
| 1 | 3 |
| 1 | 4 |

| 1 | 5 | FACILITY STATUS X (28) % POWER 0 0 0 (29) OTHER STATUS (30) Emergency Shutdown METHOD OF DISCOVERY C (31) DISCOVERY DESCRIPTION (32) Handling of reactor coolant samples

| 1 | 6 | ACTIVITY RELEASED Z (33) CONTENT OF RELEASE Z (34) AMOUNT OF ACTIVITY (35) NA LOCATION OF RELEASE (36) NA

PERSONNEL EXPOSURES

| 1 | 7 | NUMBER 0 0 3 (37) TYPE B (38) DESCRIPTION (39) Operations personnel - extremity exposures - 130, 38, 40R,

PERSONNEL INJURIES

| 1 | 8 | NUMBER 0 0 3 (40) DESCRIPTION (41) NA

LOSS OF OR DAMAGE TO FACILITY (43)

| 1 | 9 | TYPE Z (42) DESCRIPTION NA

PUBLICITY

| 2 | 0 | ISSUED N (44) DESCRIPTION (45) NA NRC USE ONLY | | | | | | | | | | | | |

NAME OF PREPARER C. F. McPhatter PHONE: (717) 948-8552

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0 1 | P A T M I 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | | (5)

LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T 0 1 REPORT SOURCE L (6) 0 5 0 0 0 3 2 0 (7) 0 8 2 8 7 9 (8) 0 1 1 5 8 0 (9)

DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 While attempting to isolate leakage from the makeup system in the fuel handling
0 3 building, station personnel received whole body and extremity exposures in excess
0 4 of allowable limits. This is reportable under 10CFR 20.405b. For further
0 5 information see Met Ed letters dated September 28, 1979 (GQL 1188) and
0 6 December 5, 1979 (GQL 1499).
0 7
0 8

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	Z Z (11)	D (12)	Z (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 9	—	0 1 8	/	1 2	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
G (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	N (23)	N (24)	Z (25)	Z 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 Detailed reviews by radiological engineering personnel and radiological control
1 1 supervision are conducted for any job involving access to highly contaminated
1 2 areas. These administrative controls will remain in effect until further
1 3 investigation indicates the proper precautions to be taken to minimize further
1 4 exposures. Improved beta detecting equipment will be utilized. (Continuing)

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	0 0 0 (29)	NA	A (31)	Routine Maintenance

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 6 (37)	B (38)	H. P. and auxiliary operators - extremity exposures - (Continuing)

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	N (44)	NA	

NAME OF PREPARER C. F. McPhatter PHONE: (717) 948-8552

January 15, 1980

LER 79-018/12L-0

Cause Description and Corrective Actions: (27)

Extra protective clothing and dosimetry will be used to properly protect and monitor workers beyond normal requirement.

Discovery Description: (32)

82, 39R whole body doses 166, 161, 29, 13, 26, 40R.

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01 | P A T M I 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T 01 REPORT SOURCE L (6) | 0 5 0 0 0 3 2 0 (7) | 1 2 0 5 7 9 (8) | 0 1 1 5 8 0 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 On 12/5/79 Diesel Generator (DF-X-1B) tripped 20 seconds after a manual start
03 attempt due to low lube oil pressure. The failure posed no threat to continned
04 core cooling; since the redundant diesel generator was verified operable, offsite
05 power was available, the BOP diesel generators were operable, and the 13.2KV
06 alternate feed was available. This event is reportable under T.S. 6.9.1.9.c
07 as described in T.S. 3.8.1.1.
08

09	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
	E E (11)	D (12)	Z (13)	Z Z Z Z Z Z (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
7 9	—	0 2 1	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
G (18)	N (19)	Z (20)	Z (21)	0 0 0 0	N (23)	N (24)	Z (25)	X 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 The failure appeared to be due to low lube oil level. The lube oil level was
11 restored to the full level and the diesel was satisfactorily tested. Appropriate
12 procedures are being revised to insure adequate lube oil level prior to starting
13 the diesel generators.
14

15 FACILITY STATUS G (28) | % POWER 0 0 0 (29) | OTHER STATUS (30) NA | METHOD OF DISCOVERY B (31) | DISCOVERY DESCRIPTION (32) Operator Observation

16 ACTIVITY RELEASED Z (33) | CONTENT OF RELEASE Z (34) | AMOUNT OF ACTIVITY (35) NA | LOCATION OF RELEASE (36) NA

PERSONNEL EXPOSURES
17 NUMBER 0 0 0 (37) | TYPE Z (38) | DESCRIPTION (39) NA

PERSONNEL INJURIES
18 NUMBER 0 0 0 (40) | DESCRIPTION (41) NA

LOSS OF OR DAMAGE TO FACILITY (43)
19 TYPE Z (42) | DESCRIPTION NA

PUBLICITY
20 ISSUED N (44) | DESCRIPTION (45) NA | NRC USE ONLY

NAME OF PREPARER C. F. McPhatter

PHONE: (717) 948-8552

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U.S. NUCLEAR REGULATORY COMMISSION

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CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 | P A T M I 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T 01 REPORT SOURCE L (6) | 0 5 0 0 0 3 2 0 (7) | 1 2 1 9 7 9 (8) | 0 1 1 5 8 0 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 During disassembly of NR-P-1D as a result of high pump vibration, it was
03 discovered that a top lug on a split ring coupling which joins two sections
04 of the pump shaft had sheared. Therefore, although the shaft was still intact,
05 the pump's ability to pump water over an extended period of time is unlikely.
06 This event is considered reportable under T.S. 6.9.1.9.b equipment failure.
07
08

SYSTEM CODE (11)	W E
CAUSE CODE (12)	E
CAUSE SUBCODE (13)	B
COMPONENT CODE (14)	P U M P X X
COMP. SUBCODE (15)	B
VALVE SUBCODE (16)	Z

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
7 9	—	0 2 2	/	0 3	L	—	0

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
A	X	Z	Z	0 0 0 0	N	N	A	B 5 8 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 The cause for the coupling failure is most likely the result of high vibration
11 induced by badly worn bearings. The defective component and the worn bearings
12 are being replaced. The cause for the bearing wear is under investigation.
13
14

	FACILITY STATUS (28)	% POWER (29)	OTHER STATUS (30)	METHOD OF DISCOVERY (31)	DISCOVERY DESCRIPTION (32)
15	G	0 0 0	NA	A	High pump vibration observed

	ACTIVITY RELEASED (33)	CONTENT OF RELEASE (34)	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
16	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER (37)	TYPE (38)	DESCRIPTION (39)
17	0 0 0	Z	NA

PERSONNEL INJURIES

	NUMBER (40)	DESCRIPTION (41)
18	0 0 0	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE (42)	DESCRIPTION
19	Z	NA

PUBLICITY

	ISSUED (44)	DESCRIPTION (45)	NRC USE ONLY
20	N	NA	

NAME OF PREPARER Ron Warren PHONE: (717) 948-8138