79-01SP

Met-Ed / GPU

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Writer's Direct Dial Number

October 15, 1979
GQL 1210

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 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

Enclosed is the fourth followup report, the first quarterly report, on the March 28, 1979 incident at TMI-2. 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 report together with updates to that report. The next report will be due on January 15, 1980.

Sincerely,

J. G. Herbein
Vice President-Nuclear Operations

JGH:JPS:tas

Enclosure: TMI-2 Incident Report dated October 15, 1979

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

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

OCTOBER 15, 1979

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METROPOLITAN EDISON COMPANY

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CONTENTS

I. SEQUENCE OF EVENTS

II. RECOVERY ORGANIZATION

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IV. RADIOLOGICAL MONITORING

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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.

0 258

II. RECOVERY ORGANIZATION

Included in this section are organization charts representing the TMI Unit 2 Recovery Organization for the period of July 1 through September 30, 1979.

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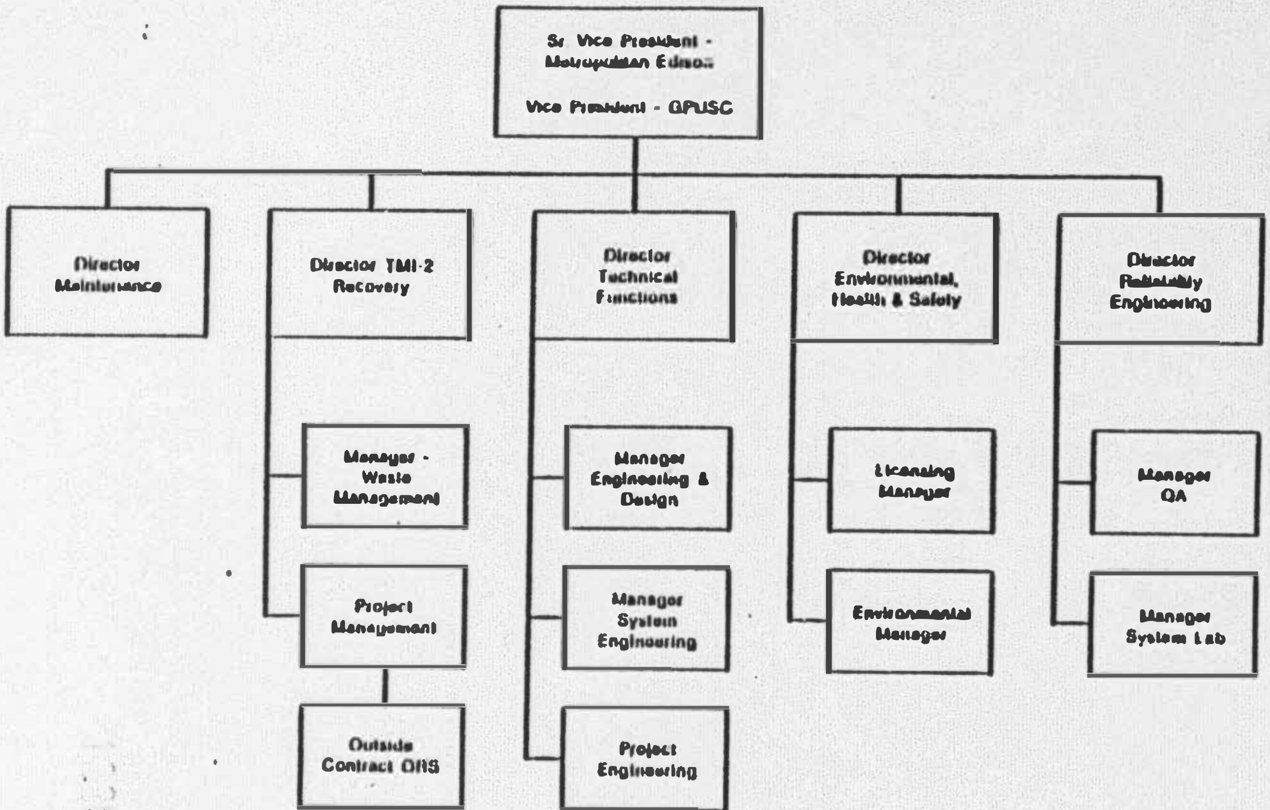

Figure B6.2-1 Corporate Management and Technical Support Staff for Radioactive Waste Management

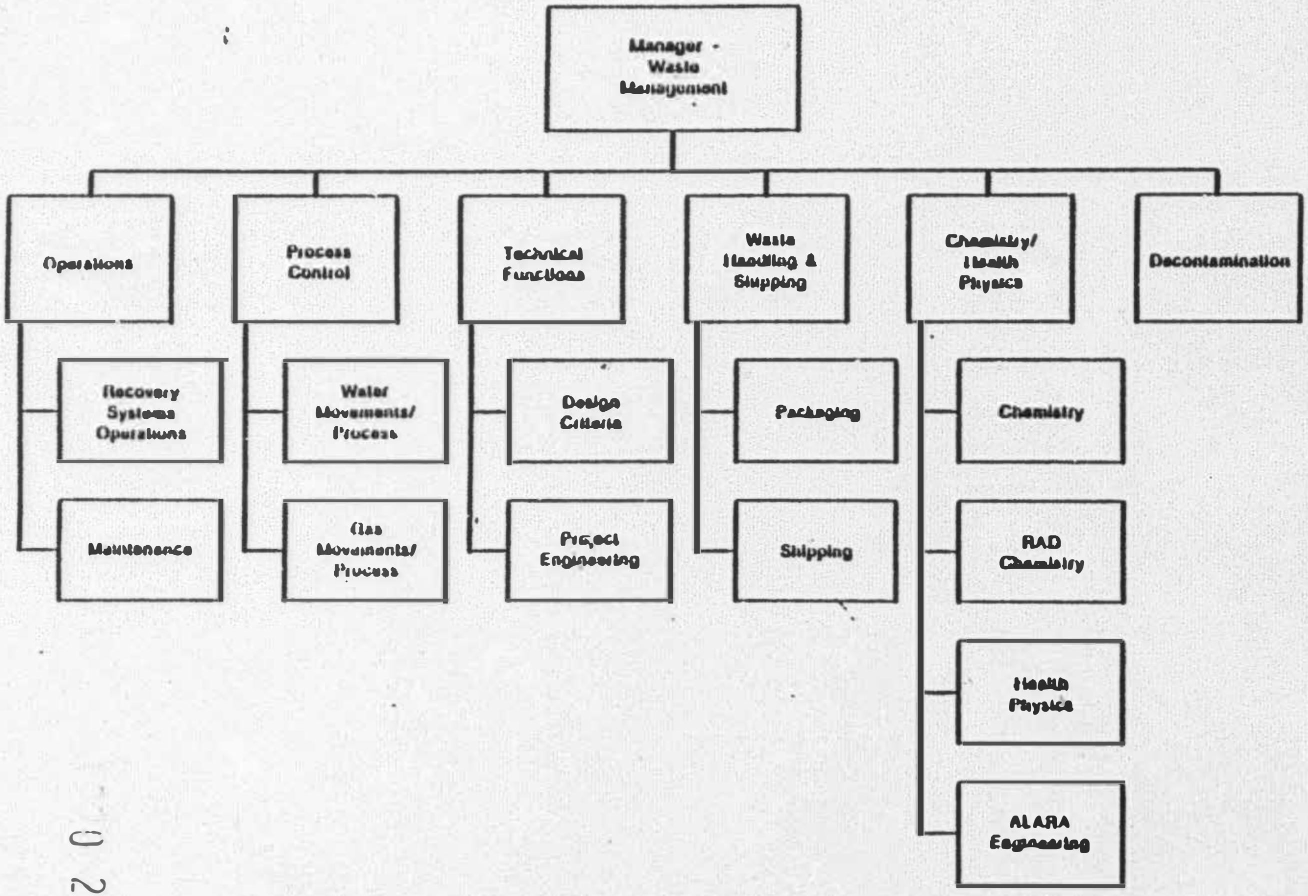

Figure B6.2-2 Radioactive Waste Management Organization

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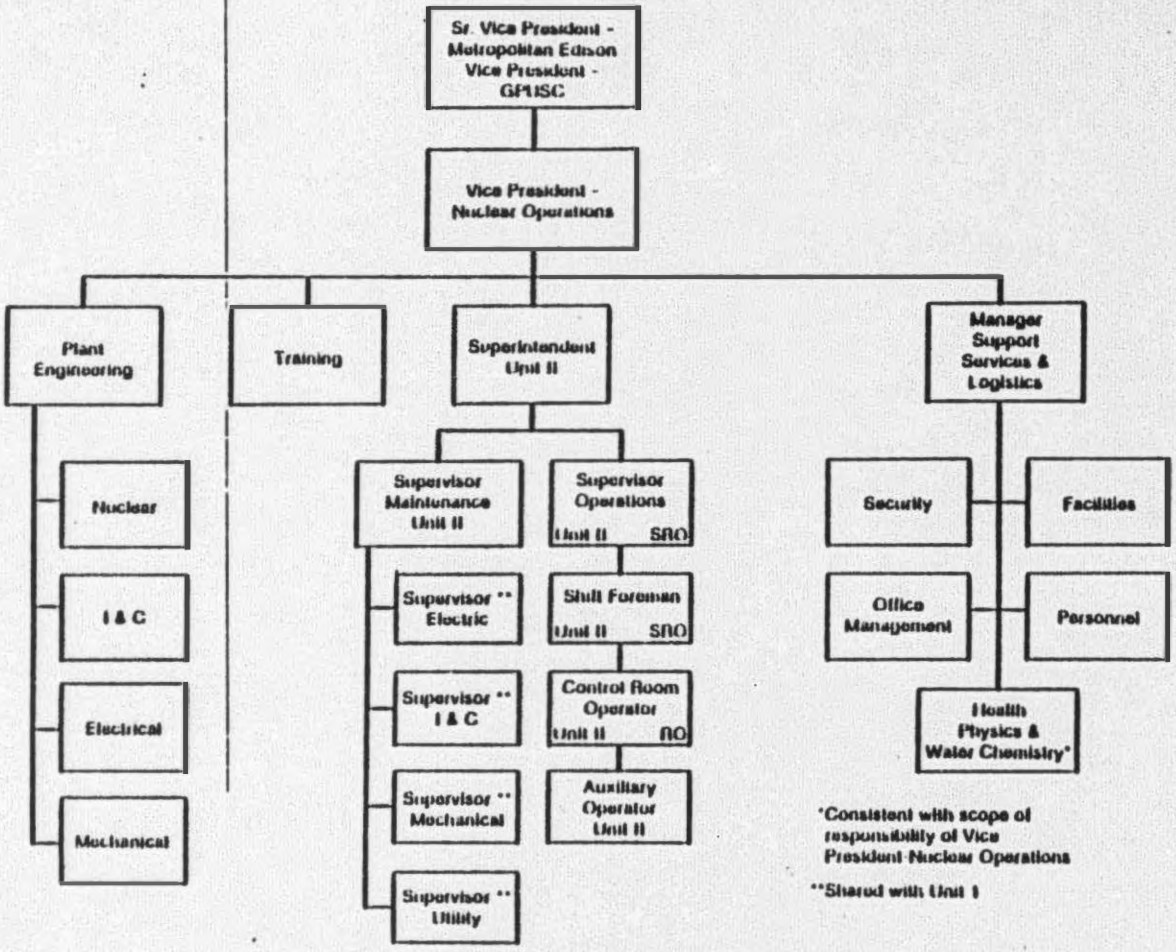

*Consistent with scope of responsibility of Vice President-Nuclear Operations

**Shared with Unit 1

Figure A6.2-2 Facility Organization - Three Mile Island Unit 2

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III. PLANT MODIFICATIONS

Included in this section are updated and amended subsections from the July 16, 1979 Third Interim Report. Changes from the previous report are denoted by change bars in the right hand margin and Rev. 3 on the bottom right hand corner of the page. Subsections from the July 16, Third Interim Report which have not had any changes are not included in this report.

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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 filter radioactive particles and absorb iodine which has passed through the normal filtration system in the building ventilation system.

2.0 System Description

The system interfaces with the Auxiliary Building HVAC System, Fuel Handling Building HVAC System, and the Service Building HVAC System.

Discharge monitoring for the supplementary system is provided at each discharge point.

3.0 System Operation

A description of the system's operation is completed. Existing plant system's component functionality is being assessed. The impact of this program on the system's operation will be addressed and any changes in the system's operation will be included in a subsequent report.

4.0 System Status

Engineering Complete

Construction Complete

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

All four (4) trains are operable. The stack is capped. Present operation is with four (4) trains.

The operating procedure, which reflects the system operation description, has been issued for use.

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POOR ORIGINAL

D. Fuel Pool Waste Storage System

1.0 System Function and Design Objectives

This Fuel Pool Waste Storage System is to be used for temporary storage of liquid waste. These tanks will add approximately 110,000 gallons to the present storage capacity of the plant, and are located within the "A" spent fuel pool. These tanks will be filled with liquid waste from both 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.

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.

POOR ORIGINAL

4.0 System Status

The system is complete and has successfully been tested using the installed steam eductors. All tanks are currently filled with contaminated water.

0 266

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 Operation

A detailed description of the system's operation is given in the operating procedure for Long Term OTSG "B" Cooling System.

A procedure has been completed to fill the "B" Steam Generator using the condensate pumps. An additional procedure to flush and vent the emergency water line has been completed as part of the fill procedure for the OTSG.

4.0 System Status

The system is installed and the preservice testing is completed.

The flushing and venting of the feedwater line has been completed.

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POOR ORIGINAL

H. Nuclear River Water System

1.0 System Function and Design Objectives

The river is the ultimate heat sink for the alternate decay heat removal (ADHR) system and the steam generator "B" closed loop cooling system.

To ensure system reliability, the nuclear services river water system was selected to supply the water.

The ADHR system requires approximately 3500 gpm, and the secondary services closed cooling water system that services the new steam generator "B" closed loop heat exchanger will require approximately 7000 gpm. These flow requirements will not be simultaneous.

2.0 System Description

Connections from the existing nuclear services river water supply and discharge headers are to be made. These connections will be made in the river water pump house and in the nuclear services river water piping between the river water pump house and fuel handling building. The former connection is for supply of river water to the "B" generator closed loop cooling scheme and the latter is to supply river water to the alternate decay heat removal system (ADHRS).

A jumper connection to supply nuclear services river water to the secondary services river water system was made in the river water pump house. The connection was made between valves NR-V3 and NR-V197 on the river water header and was fabricated in accordance with ASME Section III requirements up to and including the second isolation valve (two isolation valves are provided to segregate the safety class nuclear services river water system and the secondary services river water system). The jumper connection was made to the secondary services river water pump header downstream of valves SR-V1A, B, and C (see FSAR Figures 9.2-1 and 10.1-3).

3.0 System Operation

A detailed description of the system's operation is given in operating procedure 2104-3.1 and in the operating procedure for the Long Term OSTG "B" Cooling System for use with the Steam Generator "B" Cooling System. The description for the use of the system with the ADHR system will be included in a subsequent report.

4.0 Status

The connections for the alternate decay heat removal system to the nuclear services river water system have not been made and are not anticipated.

The connection for the nuclear services river water system to the secondary services river water system have been installed and hydro tested.

POOR ORIGINAL

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. Three pipe runs will be installed from the existing DHR system piping within 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.

POOR ORIGINAL

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3.0 System Operation

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 will not be made until the system is put into service. Tie-in of the ADHR system to the existing plant DHR system has been completed. The valve pit redesign and modification is currently under evaluation.

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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 accommodating 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.

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The SRCPC makeup system will prevent gross depressurization on the RC system when operating in a water-solid mode. Overpressurization 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 System Operation

A preliminary description of this systems operation is now available.

TITLE: Preliminary System Description Task TS-6B Standby Reactor Coolant Pressure Control System, Revision 1, dated May 23, 1979.

4.0 Status

Phase I of the SRCPC makeup system is completed and has been hydrostatically tested and operated in recirculation mode and in the make/mode to the RCS. The Phase I will allow local manual operation of the system. The design work to ultimately convert the system to control room operation (Phase II) is being implemented and is expected to be completed by October 12, 1979.

0 272

N. BOP Electrical Power System

1.0 System Function and Design Objectives

In the event of failure of normal off-site power sources to the BOP busses, the BOP Electrical Power (BOPEP) system provides an alternate source of power to serve existing components, which previously did not require loss-of-offsite power backup protection and new components that are planned to be used or may be used for decay heat removal from the primary system.

The BOPEP system is completely independent of the existing Class 1E busses.

The BOPEP busses are loaded on a 'manual only' basis in accordance with emergency operating procedures.

Modifications of power supplies associated with Steam Generator "A" cooling systems have been given priority of installation with respect to those for the Steam Generator "B" cooling systems.

The testing requirements for the BOPEP systems are to be similar to those of the Class 1E systems.

The BOPEP system shall supply power to the following components and associated auxiliaries at one time or another depending upon the specific situations:

a. Supplementary Air Compresser

b. Circulating Water Pumps

c. Condensate Pumps
   Steam Generator "A" Long Term Cooling Pumps*
   Steam Generator "B" Long Term Cooling Pump

d. New Decay Heat Removal Pump

e. Secondary System Closed Cooling Water Pumps

f. Alternate DHR System Pumps*
   Secondary Services River Water Strainer

g. Pressure and Volume Control System Charging Pumps

h. Chemical Cleaning Building Ventilation Equipment

i. Pressurizer Heaters

*Indicates components not currently planned to be put in service.

j. Alternate DHR System Closed Cooling Water Pump*
   Temporary Auxiliary and Fuel Handling Building HVAC

k. Fuel Handling Building HVAC Fans, Filters and Heaters

l. Auxiliary Building HVAC Fans, Filters and Heaters

m. Condenser Vacuum Pumps

n. Instrument and control power for above systems.

2.0 System Description

The BOPEP system includes two independent power block busses (2-3 and 2-4), each fed by a 2500 kw rated diesel generator, and two circulating water pump busses (2-5 and 2-6) fed by one 13.2 kv line. The loads associated with cooling steam generator "A" are connected to odd numbered busses. Correspondingly, loads associated with cooling steam generator "B" are connected to even numbered busses. The odd and even busses are powered by the gray and white diesel generators respectively and are, therefore, designated as the "gray" and "white" busses.

The diesel generators and associated auxiliary systems are located outdoors just south of the turbine building. Each diesel is a skid-mounted package complete with starting system, fuel injection equipment, and associated instrumentation and controls. The permanently installed fuel oil storage and supply system provides sufficient reserve for one day of rated load operation. In addition, there will be sufficient on-site fuel oil reserve to operate both diesel generators at rated load for the normal time required to obtain fuel resupply plus a four-day margin.

Suitable fire protection will be provided for the diesel generators and auxiliary systems. This may include a fire wall separating the two fuel oil tanks and diesels or a fire suppression system.

Existing circuit breakers, previously used for condensate booster pumps 2A and 2B, have been modified to connect the 2-3 (gray) and 2-4 (white) busses to their respective switchgear. Relays are provided at the busses to shed all loads on loss-of-offsite power. The existing bus transfer schemes that provide continuity of power supply by fast-transfer to the other transformer, have been left intact. To accommodate this, the new undervoltage detection schemes include a 10 second delay.

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* Indicates components not currently planned to be put in service.

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The 13.2 kv line supplies power to the circulating water pumps and their associated auxiliaries. This line is powered by a 115 kv network which is backed by combustion turbines capable of being energized independently of the 230 kv network. The 13.2 kv line has sufficient capacity to start a second circulating water pump while one pump is still operating. However, only one pump is normally required.

Circulating water pump IE is disconnected so their breakers can be used to connect the new power supply to busses 2-5 and 2-6 respectively. Buss 2-5 serves pumps 1A and 1C which are associated with steam generator "A" cooling systems. Correspondingly, buss 2-6 serves pumps 1B and 1D which are associated with steam generator "B" cooling systems.

3.0 System Operation

The BOPEP system normally provides standby power capabilities and is not operating. On loss-of-offsite power, the offsite power supply breakers will open and the diesel generators will be started and connected to their respective busses automatically.

Loading on the diesel generators, connection of the 13.2 kv line, and startup of the circulating water pumps will be performed manually from the control room in accordance with established procedures for the various potential plant conditions. For the "gray" and "white" busses, return to normal power is accomplished manually by first opening the diesel breaker and then closing the offsite supply breaker. For the 13.2 kv line, a return to normal power will be controlled manually by closing the normal supply breaker before opening the new supply breaker (not transfer).

The primary control center for the BOPEP system is the control room control and monitoring capability exists locally for the diesel generators.

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Initial startup testing will verify proper system and component operability, the adequacy of operating procedures, and ensure adequate performance capabilities of the BOPEP System. Periodic testing will be performed in accordance with procedural requirements and any additional testing and maintenance requirements by the component manufacturers. Periodic testing will verify proper breaker actuation, diesel starting and synchronizing, fuel oil quality, and breaker positions.

4.0 System Status

The work for the upgraded BOP electrical power system is approximately 95% complete.

The following work has not yet been completed:

The fire protection engineering and construction are approximately 50% complete.

Automatic lube oil system for "white" diesel is currently being designed.

POOR ORIGINAL

0 276

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

1.0 <u>System Function and Design Criteria</u>

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 being optimized with respect to ALARA considerations.

Instrumentation and controls will be provided for monitoring of system performance. Water flows will be monitored where the values are critical to the process and or system safety. Inline monitoring and a comprehensive sampling system will be provided for thorough analyses of system water cleanup performance. Radiation and airborne monitoring equipment will be 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 projected to ensure building inleakage will be established. The system is being optimized with respect to ALARA considerations.

2.0 <u>System Description</u>

Liquid Processing

The TMI StationChemical Cleaning Building is being 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 prefilter, two demineralizers and an after filter. The prefilter and demineralizers will be 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.

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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 will be 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 will be enclosed within a guard pipe, the end of which terminates inside the Chemical Cleaning Building.

Decontaminated water will be 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 1 or Unit II for discharge 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.

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 will be 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 complete and is undergoing a final operability program. Operator training and qualification is completed to the extent that sufficient operators have been qualified to operate the system.

POOR ORIGINAL

0 278

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 generated by EPICOR I and EPICOR II until they can be shipped to a burial site. WG-21 will provide space for interim staging until WG-22, Solid Waste Staging Facility is complete. Contact readings on the surface of the facility will be less than 5 mr/hr.

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 to be installed in the Unit 2 cooling tower desilting basin, backfilled for shielding and capped with 3' thick concrete plugs.

3.0 System Operation

Five (5) EPICOR I Resin Liners are staged in the facility awaiting shipment.

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 four of the five cells 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 then 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.

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POOR ORIGINAL

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

The facility has not been constructed as of this report.

4.0 System Status

The shield cask transport and transfer cask have been received on site. They are being utilized by Met-Ed and EPICOR II personnel for training purposes.

Design is complete and ECM's have been issued for construction of the staging facility except for final grading.

The mudmat, base mat, and first two lifts of concrete for the facility have been completed. All cells are in place and braced. Some delay in construction occurred due to heavy rains.

Purchase orders have been placed for all major components; fabrication of the sump liner appears to be the controlling item for facility completion.

0 280

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 shall also be 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 borocometer 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 and material procurement is in progress.

POOR ORIGINAL

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 and operating procedure will be available by October 15, 1979.

4.0 System Status

The engineering is approximately 60% complete. Piping fabrication and installation is about 10% complete. The electrical and HVAC installation are less than 5% complete. The current schedule for completion is December 24, 1979.

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V. 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 System Operation

A system description and detailed operating procedure will be available on October 15, 1979.

4.0 Status

The system is installed and currently going through startup and testing.

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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 2 turbine gland seals so that the existing auxiliary boilers (Unit 1) 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 2 auxiliary steam header.

3.0 System Operation

A detailed operating procedure will be available on October 15, 1979.

4.0 Status

The boiler skid is in place and the installation of the fuel, steam and feedwater lines is approximately 60% complete.

IV. 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 throughout August, 1979. Tables 1-9 provide additional data for liquid discharges to the Susquehanna River.

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EXECUTIVE SUMMARY

THREE MILE ISLAND UNITS I 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) Concentration (μCi/cc)	(a) 1.29E-8	(a) 7.44E-8	(a) 1.81E-7	(a) 2.59E-8	(a) 2.68E-8	(a) 8.74E-8
b) Discharges less Tritium (Ci)	(b) 0.277	(b) 0.10	(b) 1.67	(b) 1.98E-1	(b) 1.77E-1	(b) 2.05
c) Iodine-131 Released:	(a)	(a)	(a)	(a)	(a)	(a)
1) Concentration (μCi/cc)	4.97E-9	7.14E-8	2.27E-8	7.22E-10	3.68E-11	9.16E-9
2) Total Activity (ci)	0.107	0.096	2.09E-1	5.51E-3	2.43E-4	0.215
d) Tritium Released:	(a)	(a)	(a)	(a)	(a)	(a)
1) Concentration (μCi/cc)	4.83E-6	5.13E-7	7.95E-7	7.05E-7	4.60E-7	6.72E-7
2) Total Activity (Ci)	104.1	0.69	7.34	5.38	3.04	15.76
II. Airborne Iodine Released:						
a) Quarterly Release Rate (μCi/sec)	5.8E-1	5.8E-1	1.20	9.89E-3	2.12E-5	1.22
b) Total I-131 released (Ci)	4.57	4.57	9.48	7.8E-2	1.67E-4	9.6
III. Noble Gases Released:						
a) Quarterly Release Rate (Ci/sec)	1.12	1.12	1.41E-1	1.74E-4	3.00E-5	1.41E-1
b) Total noble gases released (Ci)	8.83E+6	8.83E+6	1.11E+6	1.37E+3	2.36E+2	1.11E+6

FOOT NOTES:

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

b) This data includes I-131 released to the Susquehanna River as a result of the TMI Unit II accident on 3/28/79.

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THREE MILE ISLAND UNITS I AND II LIQUID AND GASEOUS RELEASES

DISCHARGE PATHWAYS	3rd Quarter 7/1/79 to 7/31/79	8/1/79 to 8/31/79
I. Liquid Released:		
a) Concentration (µCi/cc)	(a) 3.63E-9	(a) 2.24E-9
b) Discharges less Tritium (ci)	(a) 2.54E-2	(b) 1.59E-2
c) Iodine-131 Released:	(a)	(a)
1) Concentration (µCi/cc)	4.79E-11	5.34E-11
2) Total Activity (Ci)	3.35E-4	3.79E-4
d) Tritium Released:	(a)	(a)
1) Concentration (µCi/cc)	7.20E-7	3.20E-7
2) Total Activity (Ci)	5.04	2.27
II. Airborne Iodine Released:		
a) Quarterly Release Rate (µCi/sec)	1.58E-6	1.05E-6
b) Total I-131 released (Ci)	1.24E-5	8.28E-6
III. Noble Gases Released:		
a) Quarterly Release Rate (Ci/sec)	4.12E-6	2.49E-6
b) Total noble gases released (Ci)	3.25E+1	19.62

FOOT NOTES:

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

b) This data includes I-131 released to the Susquehanna River as a result of the TMI Unit II accident on 3/28/79.

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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	8.03E -0	5.38
^{32}P	1.10E -3	4.13E -3
^{51}Cr	3.56E -4	9.43E -4
^{54}Mn	3.75E -4	1.24E -4
^{58}Co	2.08E -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
^{95}Nb	1.79E -4	4.49E -4
^{95}Zr	4.92E -5	5.58E -5
^{110m}Ag	1.14E -3	7.63E -4
^{131}I*	2.53E -1	5.51E -3
^{131m}Xe	—	6.11E -4
^{132}I	2.98E -3	—
^{133}I	1.23E -4	1.26E -5
^{133}Xe	1.12E -2	6.27E -5
^{134}Cs	1.28E -3	1.98E -3
^{136}Cs	-	1.43E -3
^{137}Cs	5.48E -3	4.17E -3
^{140}Ba	4.23E -4	3.97E -3
^{140}La	1.09E -3	4.24E -3

* ^{131}I is the only radionuclide of significance released to the river from Unit 2 accident of 3/28/79. Other radionuclides came primarily from Unit 1.

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	3.04E -0	5.04E-0
^{32}P	7.5 E -4	—
^{54}Mn	7.69E-5	3.09E-4
^{58}Co	1.63E-3	1.84E-3
^{60}Co	6.87E-4	1.28E-3
^{89}Sr	1.45E-1	1.05E-2
^{90}Sr	9.25E-3	5.71E-4
^{95}Nb	3.97E-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*	2.43E-4	3.35E-4
^{131m}Xe	9.6E-4	—
^{134}Cs	1.73E-3	5.29E-3
^{137}Cs	5.91E-3	1.29E-3
^{140}Ba	2.74E-3	8.11E-4
^{140}La	6.83E-3	1.93E-3
^{141}Ce	2.76E-5	1.63E-5
^{144}Ce	—	1.62E-4

*^{131}I is the only radionuclide of significance released to the river from Unit 2 accident of 3/28/79. Other radionuclides came primarily from Unit I.

TABLE 4

LIQUID RADIONUCLIDE DISCHARGES

BY ISOTOPE

Radionuclide	8/1/79 - 8/31/79 Activity (Ci)
^{3}H	2.27E -0
^{32}P	* -
^{54}Mn	6.40E -5
^{58}Co	1.24E -3
^{60}Co	5.88E -4
^{89}Sr	1.72E -3
^{90}Sr	1.29E -4
^{95}Nb	5.87E -4
^{95}Zr	-
^{97}Zr	-
^{103}Ru	-
^{110}Ag	-
^{110m}Ag	-
^{122}Sb	1.83E -4
^{124}Sb	7.05E -5
^{125}Sb	5.30E -4
^{126}Sb	-
^{131}I*	3.79E -4
^{131m}Xe	-
^{134}Cs	2.46E -3
^{137}Cs	7.08E -3
^{140}Ba	2.22E -4
^{140}La	5.36E -4
^{141}Ce	3.55E -6
^{144}Ce	8.17E -5

*^{131}I is the only radionuclide of significance released to the river from Unit 2 accident of 3/78/79. Other radionuclides came primarily from Unit 1.

* - Complete data not available as of 9/29/79.

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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
IWTS	2,776,600	2,348,910	1,776,070	1,821,030	1,801,030
IWFS	616,110	505,820	682,320	733,150	625,140
WECST (A&B)	93,903	112,229	41,888	125,827	56,800
UNIT I Sec. Neut.	860,037	904,694	802,475	881,262	829,303
HDCT = 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

TOTALS FOR ACCIDENT TO 8/31/79

IWTS = 10,523,640 gallons

IWFS = 3,162,540 gallons

WECST = 430,647

Sec. Neut. = 4,277,771

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CORRECTED COPY

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	2E+4 cfs	or	1E+5 cfm

*Estimate by Middle Atlantic River Forecast Center

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TABLE 8

TMI LIQUID ^3H DISCHARGE FOR 1979

	UNIT I					UNIT II	UNIT I & II				
	WECST - TANK 11A & 11B				IWFS IWTS & SEC NEUT.	WETT-TANK 9A&9B NEUT. TEST TANK 8A&8B	1051-(RML-7) 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	139 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	7.48
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	4.21E-3	0.67	0.69			6.61	4.6E-7	3.04	6/1-6/30	3.04
JULY	4.76	7.72E-3	3.67	4.15			7.00	7.2E-7	5.04	6/30-8/1	5.04
AUG.	2.15	6.15E-3	1.32	1.52	<MDA		7.10	3.2E-7	2.27	8/1-8/31	2.27
SEPT.											
OCT.											
NOV.											
DEC.											

* Sample lost due to accident on March 28, 1979

** These releases were monitored via the RML-7 composite.

TABLE 9

THI 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 10^6	^{89}Sr		^{90}Sr		TANK VOLUME DISCHARGED cc x 10^6	^{89}Sr		^{90}Sr		TANK VOLUME DISCHARGED cc x 10^6	^{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	1.94E-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	4.76	2.2E-5	1.05E-2	1.2E-6	5.71E-4	None	—	—	—	—	None	—	—	—	—
AUG.	2.15	8.0E-6	1.72E-3	6.0E-7	1.29E-4	None	—	—	—	—	None	—	—	—	—
SEPT.															
OCT.															
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

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