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April 10, 1980
TLL 160

TMI Program Office
Attn: J. T. Collins, Deputy Manager
U. S. Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, Pa. 17057

Dear Sir:

Three Mile Island Nuclear Station, Unit II (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Submerged Demineralizer System

At the present time there exists at TMI-II water in the containment sump and reactor coolant system containing isotopic concentrations in excess of 100 μCi/ml. The decontamination of these waters is required in order that recovery operations may continue. This water exceeds the 100 μCi/ml design capability of the existing EPICOR-2 facility.

Enclosed, is our Technical Evaluation Report on the Submerged Demineralizer System (SDS), an ion exchange process that represents the method selected for decontamination of these waters. This report documents the requirement for decontamination of these waters, as well as the engineering design of the proposed system to accomplish this objective.

Presently, the calculations that provide backup support for Chapters 6 and 7 are being prepared in a form that is suitable for submittal. These calculations will be submitted by approximately April 28, 1980.

Our present plans call for the initiation of processing contaminated waters by the SDS in October, 1980. Our plan calls for processing the containment sump water first, prior to commencement of reactor coolant system water cleanup. This water processing schedule has been developed to enable reduction of radiation exposure levels by a significant amount in the containment building to minimize, as rapidly as possible, radiation exposure to personnel entering and working there. As further information is obtained, and as containment building decontamination plans are developed, it may become necessary to add water to the containment sump as a result of gross decontamination of the building surfaces. Upon completion of containment sump water processing for the reduction of the source term, cleanup of the reactor coolant system water will proceed in preparation for reactor vessel entry and fuel removal.

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Other contaminated waters may be processed by the SDS upon completion of the reactor coolant system water cleanup if it is technically feasible and prudent to do so. Details of plans for this additional water processing by the SDS will be forwarded for your information and approval as they are developed.

We believe the SDS represents an optimum system for decontamination of the containment sump water and reactor coolant system water. Your early approval for use of this system is requested.

Sincerely,

G K Hovey

G. K. Hovey
Director, TMI-II

GKH:LJL:hah

Enclosure

cc: B. Snyder

TECHNICAL EVALUATION REPORT

SUBMERGED DEMINERALIZATION SYSTEM

APRIL, 1980

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Chapter 1

Summary of Treatment Plan

1.1 Project Scope

The decontamination of TMI-II includes the processing of approximately 1,000,000 gallons of radioactively contaminated water with activities as shown in Table 1.1. Presently, this water is contained in the reactor coolant system and the containment sump.

This report describes a Submerged Demineralizer System (SDS) and the work associated with the development of the system for the expeditious clean-up and disposition of the contaminated water mentioned above. Specific design features of the system include:

1. Placement of the operating system in the spent fuel pool.
2. Radioactive gas collection and treatment.
3. Liquid leak-off collection and treatment.
4. Underwater placement of ion-exchange vessels into a shipping cask without removal from the spent fuel pool.

1.2 Identification of Radionuclides and Radioactivity Levels

Water samples were taken from the reactor coolant system and the containment sump. These samples were analyzed to identify specific radionuclides and concentrations. Typical results are listed in Table 1.1.

1.3 Alternatives Considered

During the early phases of developing a system for the control, clean-up, and disposition of the contaminated water located in the containment building of TMI-II, several methods or alternatives were evaluated. These alternatives were grouped into two categories:
(1) those with no volume reduction and
(2) those with volume reduction. Presented

below, are the alternatives considered with a discussion and conclusion about each.

Alternative I: Leave Contaminated Water in Containment Indefinitely (No Volume Reduction)

Discussion:

A. Containment Sump Water

1. The sump water contains radionuclide concentrations as depicted in Table 1.1. The radiation dose rate at the surface of the sump water measures approximately 120 R/hr. The existence of this relatively high dose rate would cause radiological exposure problems during the recovery program, i.e., increased exposure to recovery personnel, increased contamination levels, and increased possibility of airborne activity.
2. The presence of the contaminated sump water would prevent decontamination of the lower levels of the containment building.

B. Reactor Coolant System Water

The presence of the contaminated water in the reactor coolant system would inhibit disassembly of the reactor and impede defueling operations.

Conclusion: Alternative I is not deemed feasible for the following reasons:

1. The potential for increased personnel exposure exists.
2. Facility decontamination and defueling operations are seriously inhibited or perhaps prevented.
3. Continued storage of the contaminated water in the containment sump for increased periods of time increases the probability, however small, that leakage from the building may occur.

Alternative II: Transfer Water to On-site Storage Facility (No Volume Reduction)

Discussion:

1. In order to safely contain the contaminated water, the construction of an on-site liquid waste storage facility would be required.
2. Additional radiation areas on the plant site would be created if a liquid waste storage facility were built.
3. Estimates indicated the construction of a waste storage facility would exceed two years.
4. A liquid radioactive waste transfer system for the transfer of the contaminated water from the various locations to the waste storage complex would be required.
5. Handling and pumping operations may involve leakage and contamination spread.

Conclusion: Based on the above discussion, Alternative II is not a feasible method.

Alternative III: Solidification and Disposal (No Volume Reduction)

Discussion:

1. The construction of an on-site solidification facility would be required.
2. Based on 1,000,000 gallons of contaminated water to be processed, a 30-gallon availability of water volume in a 55-gallon drum, 70% availability, 24-hour/day operation, and a 45 minute cycle time, the processing time may exceed four years.
3. Based on 1,000,000 gallons of contaminated water to be processed and a 30-gallon availability of water volume in a 55-gallon drum, the number of drums of solidified waste that would be generated would exceed 33,000. Handling and transportation of this extremely large quantity of solidified waste would be prohihitive.
4. The handling evolution required to solidify the contaminated water may involve suhstantial radiation exposure to personnel.
5. The potential for leakage and contamination problems may be substantial in operating a solidification facility for processing this contaminated water in this manner.

Conclusion: Based on the above considerations, Alternative III is not considered to he feasible.

Alternative IV: Submerged Demineralizer System (SDS) in the "B" Spent Fuel Pool (Volume Reduction)

Discussion:

1. The system would be capable of reducing activity levels to levels acceptable for release to environment.
2. Processing contaminated water would result in concentrated waste requiring shielding.
3. The system incorporates remote operability features.
4. Design, construction and operation would allow for relatively short lead times; the system could be in operation by October, 1980.
5. The system would require minimal maintenance.
6. The system would be amenable to location within the Spent Fuel Pool which would utilize the natural shielding of the contained water.
7. Concentrated waste would be transported to a licensed commercial burial ground in accordance with existing regulations.

Conclusion: Based on the above considerations, Alternative IV is an acceptable method for decontamination.

Alternative V: Epicor II System (Volume Reduction)

Discussion:

1. Some contaminated waters may require dilution prior to processing in EPICOR II to decrease the activity level to less than or equal to 100 uCi/ml. Additional water volumes would be created causing a requirement for increased processed water storage volume.
2. The system is presently processing intermediate level waste waters at other locations on the plant site.
3. The curie loading levels of Epicor II vessels are limited due to shielding design considerations resulting in an increase of the following:
   a. Number of vessels and radioactive waste shipments required.

b. Processing time.

c. Additional handling requirements.

d. Personnel exposure.

4. The system requires minimal maintenance.

Conclusion: The use of EPICOR II to decontaminate the higher level waste waters is rejected for the following reasons:

1. Continued treatment of the intermediate level waste water is required.
2. Increased processed water storage volume is not practical.
3. Higher than necessary personnel exposures is not consistent with ALARA.

Alternative VI: Evaporation (Volume Reduction)

Discussion:

1. Evaporation would require the design and construction of a new facility.
2. Due to the nature of the contaminated water to be processed the design of the facility would be complex to allow for maintenance of the processing system and personnel radiological protection. The construction of the facility would require at least two years.
3. Evaporation provides the ability to process a wide range of chemical contaminants.
4. Evaporation typically provides a decontamination factor of 10^4.

Conclusion: Evaporation is an acceptable alternative for processing the contaminated waste waters. Based on the long construction time of the facility and inherent potential for higher occupational exposure due to increased maintenance requirements, this alternative is less desirable than Alternative IV, Submerged Demineralizer System (SDS).

1.4 Description of the Decontamination Process

1.4.1 General

Analysis of the alternatives previously presented has resulted in the determination that, of the two alternative categories considered, volume reduction is appropriate for the disposition of contaminated water. This conclusion was reached based on the considerations that volume reduction:

1. fixes the contaminants
2. concentrates the activity
3. minimizes storage and disposal space

Of the volume reduction category, the Submerged Demineralizer System (SDS), or Alternative IV, was chosen on the most appropriate process for the following reasons:

1. basic design simplicity
2. high performance for decontaminating liquids, i.e., decontamination factors up to 10^6
3. amenable to placement under water to take advantage of shielding properties of the water.
4. ability to implement in a timely fashion for support of the overall objective of expeditious fuel removal.

The SDS is an ion-exchange process expected to provide decontamination factors up to 10^6 for cesium and 10^4 for strontium, thus removing the majority of the activity. The remaining radionuclides, except for tritium, are also removed by the ion-exchangers with expected decontamination factors ranging from 10 to 100.

1.4.2 SDS Operating Description

Figure 1.2 is provided and is the block flow diagram that depicts the process flow in the submerged demineralizer system. Contaminated waters enter the SDS via a supply manifold that permits selection of the input water source. These waters pass through cartridge-type filters for removal of particulate matter prior to holdup in the four 15,000 gallon holdup tanks.

Contaminated waters are pumped from the holdup tanks to the processing system. A sample box is provided to enable sampling of the influent water to determine the radionuclide content.

The ion-exchange beds consist of six underwater colums (24 in. x 54 in.), each containing approximately 7 cubic feet of zeolite resin. Inlet, outlet, and vent connections are made with remotely operated couplings. The beds are arranged in two parallel trains with three colums in each train. Flow may be directed through one train of three resin beds or through both trains in parallel. Loading of the beds will be controlled by feed batch size, loading time, effluent sample analysis, and continuous monitoring.

When the desired bed loading is achieved on the first bed of the train, the feed flow to the train will be stopped, the bed will be flushed with clean water, and the first bed will be disconnected and moved to the storage rack using the pool area crane. The second and third beds will be disconnected, moved to the first and second positions, respectively. This operational concept has eliminated the potential for valving errors and also minimizes the possibility of an unexpected resin "breakthrough" (when the resin is completely loaded) which could recontaminate the water already processed.

Two additional ion-exchange columns will be located underwater and are immediately downstream of the zeolite resin beds. These exchanger beds will contain organic cation resin for

removal of residual radionuclides. Column loading will be limited to less than 75 Ci of Strontium based on primary column effluent monitoring. The columns are arranged to be operated singly or simultaneously in parallel. This loading limit is based on restricting the integrated radiation dose to the resin to less than 10⁸ RADs.

Waste water will be processed in batches, sampled and analyzed, and fed to the ion exchange system. Sampling lines are provided on the SDS feed and effluent streams from the zeolite and cation beds. Sample flows from the bed effluent may be selectively passed through beta monitors to detect breakthrough and allow comparison of the monitor readings to sample results. Monitors are also provided on the polishing unit influent lines, the leakage containment influent line and in the general area of the valve box. These monitors have an alarm system and a high radiation trip point that will automatically close a remotely operated valve on the main feed line, stopping the operation in the event of a leak or bed breakthrough.

A 195 cubic foot hed is provided downstream of the cation exchangers to remove trace fission products not removed earlier in the process. This "mixed-bed" polishing exchanger will be located above water level and loading will be determined by influent and effluent sampling.

A monitoring system consisting of holdup tanks will collect treated effluent from the polishing demineralizer. The contents of these tanks will be sampled to determine the level of residual contamination. Recycle of the treated water will be possible in the event that radioactive contaminant levels do not meet criteria for the polished effluent from the system.

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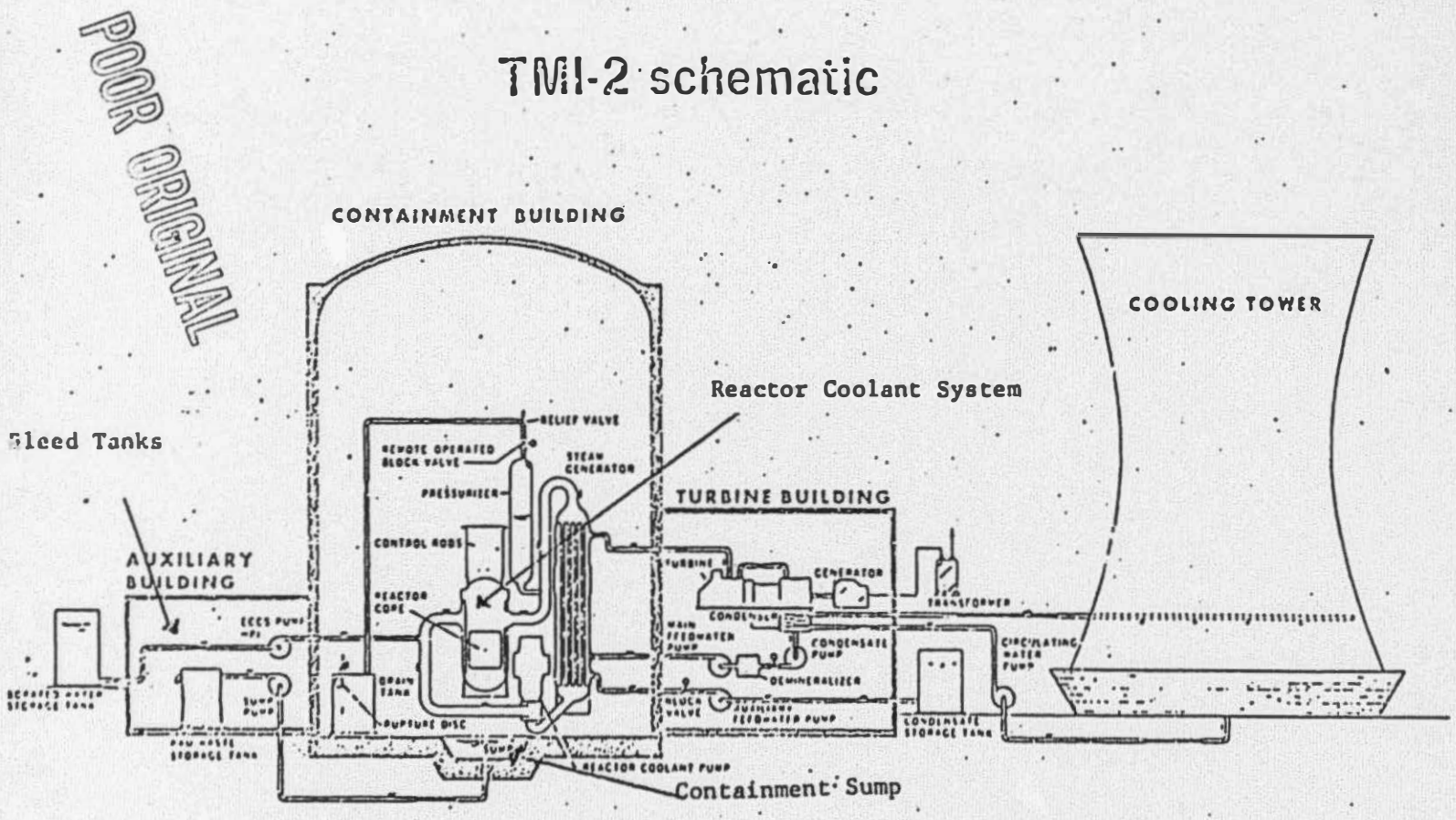

Fig. 1.1 - Illustration showing the location of contaminated water at Three Mile Island Nuclear Station Unit No. 2.

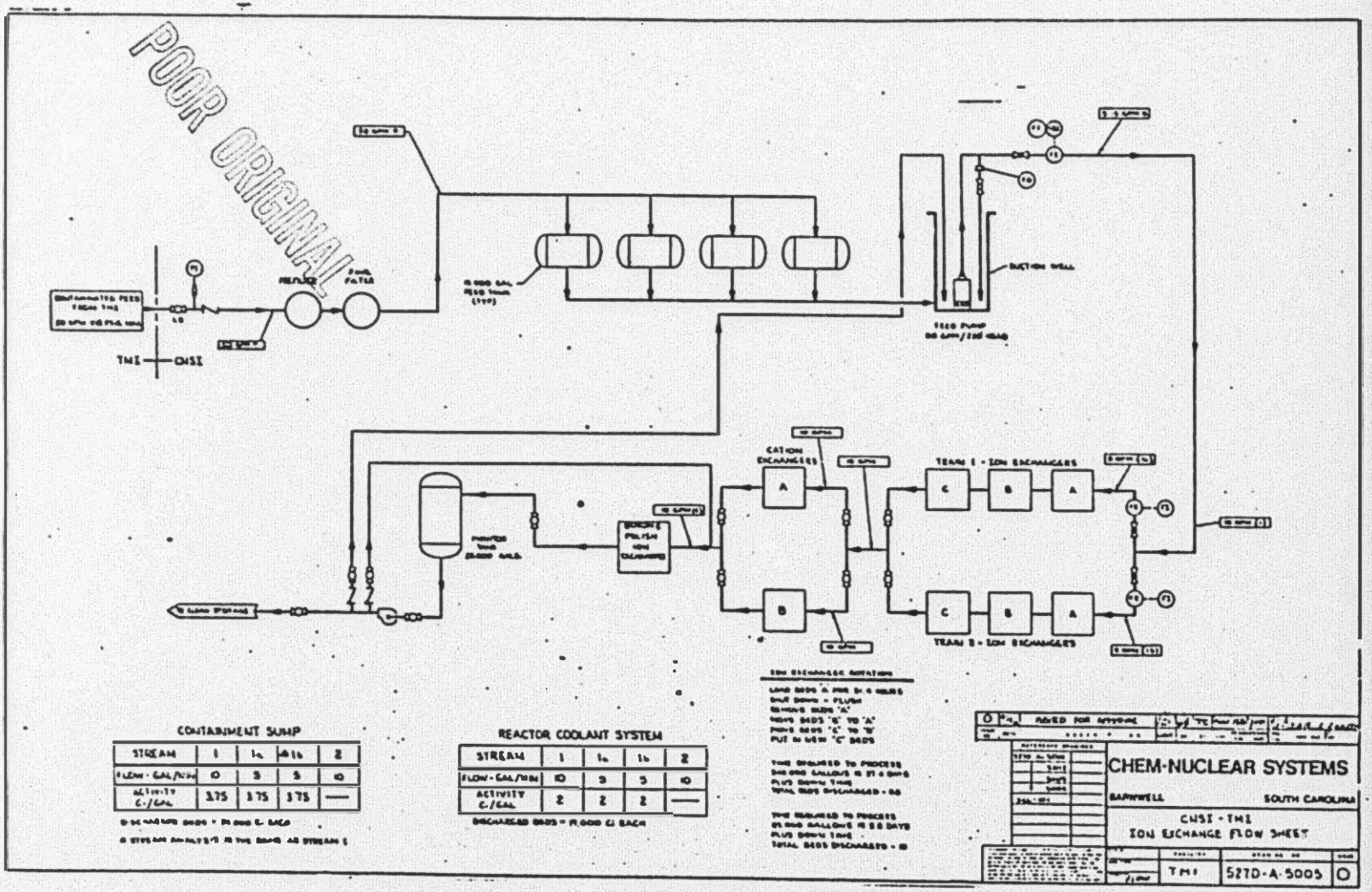

FIGURE 1.2 Ion Exchange Flow Chart

TABLE 1.1

Typical Results of Analyses from the Reactor Coolant System Water and the Containment Sump Water

Isotope	Reactor Coolant System	Containment Sump
Cs-134	8.0 µCi/ml	40.0 µCi/ml
Cs-137	42 µCi/ml	176 µCi/ml
Sr-89	33 µCi/ml	40.0 µCi/ml
Sr-90	27 µCi/ml	2.7 µCi/ml
H-3	0.15 µCi/ml	1.03 µCi/ml
pH	8.0	8.6
Boron	3900 ppm	2000 ppm
Na	1450 ppm	1100 ppm

Chapter 2
Summary of Health and Environmental Effects

2.1 Occupational Exposure During Routine Operation

The SDS has been designed to maintain exposures to operating personnel as low as reasonably achievable. To implement the ALARA concept, components carrying high level activity water, that are not contained in the fuel pool, have been provided with shielding. Shielding has been designed to limit whole body exposure rates in operating areas to less than 1 mR/hr. In addition, components carrying high level process fluids have been designed for exhaust to the SDS offgas system. This off-gas treatment will minimize the potential for airborne radioisotope releases in the work areas.

2.1.1 Exposure Planning

Several activities will be implemented prior to the SDS start up to assure occupational exposures are minimized. These include:

- Review of operating, maintenance and surveillance procedures to assure precautions are adequate.
- Review of the installed system to identify potential problems during operation and the implementation of corrective actions.
- Detailed time studies during preoperational testing and system training to update exposure estimates.

When time studies have been completed and operating and surveillance frequencies are established, total occupational exposures for various activities during SDS operation may be predicted. This exercise will permit review of those activities estimated to yield the highest man-rem expenditure. Re-examination to assure that every reasonable effort is expended to minimize personnel exposure may include the following considerations:

- Reduction of the frequency of operation
- Temporary or additional shielding
- Tool modifications

- Procedure modification
- Personnel training to reduce work time
- Component modifications

2.2 Exposures to the Public During Routine Operation of the SDS

Maximum individual dose commitments for 365 days operation of the system are 3.6×10^{-3} mrem for whole body exposures, 4.4×10^{-3} mrem for bone exposure, 3.6×10^{-3} mrem for thyroid exposure, and 3.0×10^{-3} mrem for GI tract exposure. The total dose to the entire population within 50 miles is calculated to be 0.12 man-rem.

It is important to emphasize that conservative assumptions (tending to maximize dose) have been applied throughout the calculation of maximum individual and population dose. Even with the application of conservative parameters, the population doses have been evaluated to be acceptable. A detailed summary of the method used to estimate the maximum individual dose and the population dose is included in Chapter 6.

2.3 Evaluation of Unexpected Occurrences

The radiological assessment includes the analysis of three hypothetical accidents that are assumed to occur during operation of the system. The first accident is an inadvertent pumping of containment water into the fuel storage pool until a total of 450 gallons of radioactive water is released to the pool. Significant exposures to the public do not occur since the contaminated water is contained in the spent fuel pool. The maximum exposure rate at a distance of six feet above the pool surface is estimated to be 430 mrem/hour. Since the release of water occurs underwater, no significant exposures are expected for workers. The primary impact of the accident is the creation of an additional 233,000 gallons of contaminated water.

The second hypothetical accident assumes a pipe is ruptured and contaminated water is sprayed into the building and fuel storage pool. It is possible that workers could be contaminated, however, prompt emergency decontamination procedures would prevent major radiation exposures. The maximum exposure rate three feet above an area on the floor on which the spray water resides is expected to be 11 R/hour. The radioactive materials would be contained within the building except small amounts of

radionuclides that would become airborne and subsequently be released through the monitored station discharge. This airborne radionuclide release would not result in significant exposures to the public.

The final hypothetical accident evaluated considers the inadvertent raising of a loaded prefilter above the pool surface. The exposure rate at a distance of 15 feet from the source is estimated to be 21 R/hour and could result in a dose of approximately 1.8 rem to workers who remain in the area for a five minute period.

2.4 Industrial Health and Safety

2.4.1 Public Safety

Operation of the Submerged Demineralizer System poses no risk from an industrial standpoint to the general public for the following reasons:

1. The majority of the lifting and handling activities take place within the TMI complex.
2. Hazardous chemical species, flammable or explosive substances, heavy industrial processes, and concentrated manufacturing activities are not involved in the installation or operation of the SDS.
3. The number of shipments involved will not be large enough to cause a significant difference in traffic on the highways utilized between TMI and the disposal facility.
4. "Exclusive use vehicles" ensure that the trucks will be controlled, monitored, and supervised in a safe manner and that the drivers are sufficiently trained and qualified to handle the responsibility of safe transportation of these loads.
5. No toxic substances are used in the SDS.

2.4.2 Occupational Safety

During the operation of the SDS, operating personnel will adhere to station requirements for occupational safety. All of the structural equipment and operating equipment used shall

meet Occupational Safety and Health Administration requirements as applicable. Any personnel protective equipment that would be required for the operation of the SDS will be utilized in accordance with standard station procedures.

2.5 Non-Radiological Environmental Effects

Adverse environmental effects from the construction and operation of the SDS are not anticipated. The system will be installed and operated in an existing, on-site facility and thus will not require any change in land-use. Additionally, the system is designed in such a manner as to allow zero discharge of liquid effluents to receiving waters. The final disposition of the processed water will be determined at a later date. Solid wastes (spent ion-exchangers, etc.) generated by the SDS will be secured and held until final disposal is accomplished.

2.6 Ultimate Waste Disposition

The solid waste generated by the operation of the SDS will be handled and transported by a licensed carrier in accordance with the Department of Transportation and NRC regulations to a licensed burial facility for ultimate disposition.

Chapter 3
Process Description

3.1 Introduction

A combined filtration-ion exchange process has been selected as the method for treating radioactive water contained in the reactor coolant system and containment building. The filter ion - exchange method has been used successfully (Lin, 1973 and Clark, 1978) to reduce quantities of radionuclides to levels that are in compliance with 10 CFR 20 and 10 CFR 50.

The initial processing of the waste water is filtration for the removal of solids to optimize the subsequent ion-exchange process. Filtration is necessary for the system to achieve designed decontamination factors.

After filtration, radioactive ion removal from the waste water involves the use of ion-exchange materials. The first three ion-exchange columns contain an inorganic zeolite material which effectively removes essentially all of the cesium and most of the strontium. Other trace levels of radionuclides, including ^{95}Nb, ^{103}Ru, ^{140}La, ^{131}I, are also partially removed by these zeolite exchangers. The radioactivity content in the effluent stream of each resin bed is used to determine when the bed is expended and replaced. After leaving the zeolite exchangers, the remaining strontium in the effluent stream is effectively removed by an organic cation resin contained in the next ion exchange column.

Final demineralization of the contaminated water is accomplished in a large, mixed resin bed containing both cation and anion organic materials. Essentially all remaining dissolved radionuclides are removed from the water during this process step.

3.2 Ion-Exchange Concepts

Ion-exchangers are solid inorganic and organic materials containing exchangeable cations or anions. When solutions containing ionic

species are in contact with the resin, a stoichiometrically equivalent amount of ions are exchanged. As an example, an ion-exchanger in the sodium (Na^+) form will "soften" water by an ion-exchange process. Hard water containing $CaCl_2$ is "softened" by this exchange mechanism which removes the Ca^{++} ions from solution and replaces them with Na^+ ions. In a similar manner, Sr^{++} and Cs^+ ions are exchanged with the Na^+ ions from the solid zeolite material.

Characteristic properties of ion exchangers involve micro-structural features contained in a framework held together by chemical bonds and/or lattice energy. Either a positive or negative electric surplus charge is carried within this framework which must be compensated for by ions of opposite sign. Because the exchange of ions is a diffusion process within the structural framework, it does not conform to normal chemical reaction kinetics. The preference of ion exchangers for a particular specie is due to electrostatic interactions between the charged framework and the exchanging ions which vary in size and charge number.

The decontamination factor (DF) is the ratio of the concentration in the influent stream to that in the effluent stream and is used for determining the efficiency of a purification process. The following equation is a qualitative expression for the removal of a single ionic specie from solution (Lin, 1975).

$$DF = \frac{1}{1 - \frac{KnQE_w}{C_f V}}$$

where: Q = Total exchange capacity (meq/ml wet resin)

n = Fraction of Q used

E_w = Equivalent weight of the nuclide under consideration

C_f = Nuclide concentration (weight/volume)

V = Feed throughput (number of resin bed volumes)

K = Unit conversion constant

Important variables which are considered as part of the evaluation of ion-exchangers for decontamination are resin type, selectivity and capacity, concentration of the species to be removed, total composition of the feed stream, and the presence of contaminants. Operating parameters such as resin bed size, flow rate and distribution, pH, and temperatures are specified for the resin beds in order to maximize removal of the contaminating ions.

Specifications which have been defined for this purification process include:

(1) The flow rate to provide an acceptable residence time for ion diffusion into the resin.
(2) The cross-sectional area of the resin to provide an acceptable linear velocity through the bed.
(3) The bed depth to result in an acceptable pressure drop.
(4) A uniform flow distribution and a uniform resin distribution to reduce the potential for channeling and allow the solution to pass through with sufficient time for ionic diffusion and exchange.
(5) The resin bead size to minimize attrition and large pressure drops.
(6) The curie loading to satisfy personnel exposure, radiation damage, transportation, and storage regulations.
(7) The cation form and the amount of resin impurities to maximize removal of specific nuclides.

3.3 Ion-Exchange Materials

The ion-exchangers selected for use in this processing system are an inorganic zeolite material that is commercially available and know as Ion Siv IE-95 (formerly AW-500), and organic cation and anion resins. Zeolites are aluminosilicates with framework structures enclosing large and uniform cavities. Because of their narrow, rigid, and uniform pore size, they can also act as "molecular sieves" which sorb small molecules, but which exclude molecules that are larger than the openings in the crystal framework.

Organic ion exchange resins are typically gels and are classified as crosslinked polyelectrolytes. Their framework, or matrix, consists of an irregular, macromolecular, three-dimensional network of hydrocarbon chains. In cation exchangers, the matrix carries ionic groups such as SO_3, COO, $(PO^2)_3$, and in anion exchangers groups such as NH^+_3, N^+, S^+ are carried. The framework of the organic resins, in contrast to that of the zeolites, is a flexible random network which is elastic, can be expanded, and is made insoluble by introduction of cross-links which interconnect the various hydrocarbon chains. The extent of crosslinking establishes the mesh width of the matrix and, thus, the degree of swelling and the ion mobilities within the resin. This, in turn, determines the ion exchange rates and the electric conductivity of the resin. The chemical, thermal, and radiation stability of the organic resins are limited compared with the zeolites and will be used after most of the radioactive ions have been removed from the solutions by the zeolite exchangers.

Since the mechanism of the ion exchange process involves the stoichiometric exchange of ions between the exchanger and the solution while electrical neutrality is maintained, the rate determining step is controlled by the interdiffusion of ions within the framework of the ion-exchanger. Since the rate of ion exchange is determined by diffusion processes, rate laws are derived by applying well-known diffusion equations to ion-exchange systems. However, complications arise from diffusion-induced electric forces, from selectively, specific interactions, and changes in swelling such that rate laws are applicable for only a few limited cases. Experimental efforts have been conducted at the Savannah River Laboratory to investigate the kinetics of cesium and strontium ion-exchange with the zeolite exchanger. Cesium was absorbed so rapidly that only rough estimates of the diffusion parameter could be obtained. The resulting equation, used to calculate column performance, did not involve kinetic parameters but was suitable to describe the equilibrium column behavior.

3.4 Resin Selection Criteria

Technical information obtained from previous reactor use of various ion-exchange materials and the results of recent experimental work with simulated and actual water samples from Three Mile Island were used to support the selection of specific ion exchange materials for this processing system. The performance of an ion exchange system is controlled by the physical and chemical properties of the exchange material as well as by the operating conditions specified in Section 3.2. The important criteria which were used in the ion exchanger selection process included:

(1) Exchange capacity
(2) Swelling equilibrium
(3) Degree of crosslinking
(4) Resin particle size
(5) Ionic selectivity
(6) Ion-exchange kinetics
(7) Chemical and physical stability

Both the inorganic zeolite and the organic resins satisfied the specifications necessary to treat this contaminated water and both types have been used extensively for water decontamination at nuclear facilities. Zeolite exchangers are also used in the Idaho Chemical Processing Plant fuel storage basin (Lin,1973). Most of the gross beta activity at that site consists of Cs-137 and Sr-90 which was released during the underwater cutting of spent fuel rods. Approximately 40,000 bed volumes of solution was processed before 50% breakthrough of Cs-137 occurs. In contrast, only 200 bed volumes of the Three Mile Island water are anticipated to be processed before the first zeolite bed is removed and replaced. Depending on breakthrough characteristics of the beds, a maximum of 30,000 curies can potentially be loaded on the zeolite beds. The net effect of three zeolite beds in series will result in an insignificant cesium outlet concentration (postulated DF greater than 10^4) and only a small fraction of the strontium reaching the organic resins (postulated DF greater than 100).

Experimental studies with reactor coolant water have been conducted in order to support and verify the selection of these ion-exchangers. The decontamination factors for the major contaminants were measured using a number of candidate ion exchangers including the organic resins, HCR-5 and SBR-OH, and the zeolite ION SIV IE-95. The results indicated the most favorable type of resins to be used in the cleanup process were the standard cation-anion resins in combination with the zeolite exchanger.

3.5 Predicted Performance of Ion-Exchangers

The concentrations of radionuclides in samples of water from the containment building sump have been measured. Those still detectable in September 1979 included the isotopes ^{89}Sr and ^{90}Sr, ^{134}Cs and ^{137}Cs, and ^{125}Sb and the short-lived ^{95}Nb, ^{103}Ru, ^{140}La, and ^{131}I. The expected performance of the zeolite and cation exchangers for the cesium isotopes is equivalent to a minimum DF of 3×10^4. For the strontium isotopes, the minimum expected DF is 10^4 with the cation resin reducing the last traces with a DF of 10^2. Any antimony is expected to pass through the zeolite and cation exchangers and will end up as the predominant gamma emitter in the solution entering the mixed-bed demineralizer. The concentration of ^{125}Sb in the containment building sump sample is approximately 0.012 microcuries per milliliter. Based on empirical data from field usage, a conservative O/F of approximately 100 can be obtained with the mixed bed demineralizer.

The zeolite and cation resins are not as effective for removal of the shortlived isotopes as they are for cesium and strontium. A combined DF between 10 and 100 can be expected. However, at the time of commencement of contaminated water processing, the short-lived isotopes indicated above will be decayed to insignificant levels.

3.6 Monitoring of Ion Exchangers

Methods used to monitor the effectiveness of the ion exchangers include liquid sampling and in-line radiation detectors. Liquid

samples of feed and effluent streams can also be used to establish the approximate curie loadings in the discharged beds. The detectors sampling the cation influent can provide gross activity indication to provide the necessary protection for the cation beds. This influent concentration must be limited to control the strontium loading on the organic resins where damage by radiation might occur. At a specified strontium breakthrough, the first zeolite bed of the SDS will be removed from the purification system and replaced with the second bed. Based on information obtained during the kinetic studies performed at the Savannah River Laboratory, a 1% to 2% strontium breakthrough from the third zeolite bed will be equivalent to 180 to 200 bed volumes of processed wastes.

Chapter 4
Submerged Demineralizer System Design Basis

4.1 Introduction

The Submerged Demineralizaton System (SDS) is an underwater ion-exchange system which has been specifically designed to process higher-level waste waters*, with inherent system features for reducing occupational and environmental exposures. The SDS will be submerged in the spent fuel pool (1) to provide shielding during operation, (2) to permit access to the system during demineralizer changeout, (3) to minimize the hazard from potential accidents, and (4)to utilize an existing Seismic Category I facility.

Design features include:

1. Arrangement of the zeolite beds 3 to a series train, with 2 trains in parallel to achieve desired process flow rates and decontamination factors (DF's).
2. Series operation logic that allows for sequencing the demineralization units to prevent activity breakthrough in the final zeolite bed and maximize activity loading on spent beds to accomplish the best possible volume reduction.

The design objectives are as follows:

a. A totally integrated system that is as independent as possible from existing waste systems at the Three Mile Island plant. The SDS is a temporary system.
b. A system that would reduce the fission product concentration and has optional capabilities for removing chemical contaminants in the water to a level that would meet existing regulatory requirements for release to the environment.
c. A system that could be operated with a minimum of exposure to personnel and a negligible risk to the public.
d. A system that could accomplish the objective listed above in a timely and cost effective manner.

* Higher-level waste waters are those contaminated waters having gross activity concentrations in excess of 100 uCi/ml.

4.2 Components of the SDS Waste Processing System

The SDS is comprised of the following components, all of which will be located in the Unit 2 A fuel pool, B fuel pool, or in the near vicinity of the B fuel pool. (See Figure 5.8, General Layout Plan.)

1. Feed filtering system;
2. Feed tank system - consisting of the existing tank farm, four 15,000 gallon tanks utilized as one 60,000 gallon tank;
3. Two parallel primary ion exchange trains, each comprised of three 10-cubic-foot vessels loaded with 7-cubic-feet of zeolite exchange media;
4. Two parallel ion exchange beds containing organic cation resin for residual radionuclide removal;
5. A monitoring and sampling system for control of demineralizer unit loading;
6. A secondary containment system for the filters, zeolite and cation beds and radiation shielding for piping, valves, samples, and monitors;
7. A mixed-resin ion-exchange polishing bed for removal of trace fission products that are not trapped on the primary or cation beds;
8. Two monitoring tanks for collecting and sampling the treated water prior to transfer for storage or ultimate disposition;
9. An off-gas system for treating and filtering gases and vent air from the system;
10. Associated piping, valving, and structural supports required for placement of system components;
11. Auxiliary systems including under water ion-exchange column storage, a column dewatering system, analytical equipment, and solidification capability;
12. Optional - A standard 195-cubic-foot ion exchange bed for boron removal.

4.3 Submerged Demineralizer System Design Criteria

4.3.1 Design Basis

Regulatory guidance followed during the design of the Submerged Demineralization System was extracted from the following documents:

- U. S. Nuclear Regulatory Guide 1.140 dated March, 1978
- U. S. Nuclear Regulatory Guide 1.143, dated July, 1978
- U. S. Nuclear Regulatory Guide 8.8, dated June, 1978,
- U. S. Nuclear Regulatory Guide 8.10, dated May, 1977,
- U. S. Regulatory Guide 1.21 Revision 1, June 1974

4.3.2 Process

The design shall provide for operation and maintenance in such a manner as to maintain exposures to plant personnel to levels which are "as low as is reasonably achievable", in accordance with Regulatory Guide 8.8.

This system reduces the level of radioactivity in the radioactive liquid by the process of demineralization and filtration.

4.3.3 Performance

The isotopic inventory for the water to be processed is summarized in Table 1.1. The SDS system is designed and operated such as to reduce the average isotopic specific activity of the treated waste streams to concentrations equivalent to levels normally required for discharge into the environment.

4.3.4 Capacity

Flow Rate - 5 or 10 GPM (5 GPM per train). The system will have the ability to operate continuously subject to periodic maintenance shutdown, in-process effluent sample(s) analysis and continous radiation monitoring.

4.3.5 Performance and Design Requirements

The following system requirements have been incorporated into the design of the SDS.

- Leak Tightness
- Shielding (Beta, Gamma)
- Ventilation
- Functional Design and Maintainability
- Decontamination - Decommissioning

4.3.6 Piping

1. The mechanical and structural design criteria and fabrication of piping systems and piping components are specified in ANSI B31.1, 1977 Edition with Addendum through Winter 1978.
2. Piping design shall be based on a maximum of 150 psi at 100°C.

4.3.7 Vessels, Tanks and Columns

1. The mechanical and structural design criteria and fabrication of vessels and tanks will be in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1977, Addendum through Winter 78

2. The vessels shall be of three types:
   a. Primary ion-exchangers shall contain approximately seven (7) cubic feet zeolite resin for the primary purpose of removing cesium and strontium from the waste water.
   b. Cation ion-exchangers shall contain strong acid organic cation resin to remove residual strontium.
   c. Filter units shall contain cartridge type filter assemblies or equivalent mechanisms capable of removing particles greater than approximately 10 micron.

3. All three types of ion-exchangers and filters shall be capable of functioning submerged under twenty (20) feet of water within the spent fuel pool.

4. The ion-exchangers shall be designed for 5 GPM nominal process rate, filters shall be designed for 50 GPM nominal; volume velocity through the resin bed shall be limited to prevent channeling or breakthrough.

5. Pressure loss through the ion-exchangers shall not exceed 15 psi when operating at 5 GPM with clean resins.

6. The exchangers shall be equipped with a lifting arrangement compatible with the spent fuel pool crane mechanical connectors.

7. The 10-cubic-foot vessels shall be equipped with all required nozzles, including inlet, outlet, and vents.

8. The exchanger shall be equipped with all internals required for distribution, dewatering, and venting.

9. Design Conditions

   a. The 10-cubic-foot units shall be designed to operate at a maximum of 150 psig at 100°C. The design conditions for continuous operation are 100 psi at 100°F (37.7°C).

   b. The following additional design conditions have been imposed:

. Overall Height	53½ inches
. Overall Diameter	24½ inches
. Materials	Stainless Steel
. Weight	Must have sufficient weight (negative bouyancy) to sink without first filling with water.

10. Testing

The vessels shall be hydrostatically tested at 11/2 times the design pressure for a minimum of thirty (30) minutes.

4.3.8 Shielding Design

The shielding shall be designed to reduce levels resulting from the SDS to less than 1 mR/hr, general area.

4.3.9 Leakage

To ensure that leakage from the submerged components does not introduce activity from the process streams into the pool water, these components will be contained within secondary containment enclosures from which pool water will be continuously processed through a separate ion-exchanger.

4.4 System Operational Concepts

The following is a summary operation description intended to provide a basis for detailed design.

The SDS process logic consists of the following basic steps:

1. Demineralization units will be preloaded with new resins prior to placement in the system. The primary treatment beds will utilize zeolite resin. The cation exchanger beds will use standard organic resins.
2. These preloaded demineralization units will be lowered into the Unit 2 spent fuel pool. A polishing unit containing 195 cubic feet of resin will be connected at the outlet of the primary beds.
3. Inlet/outlet/vent header connections will be made to the demineralization unit. The vent header connection will be routed back to a receiving tank.
4. Water will be introduced to fill and vent the demineralization unit.
5. The demineralization system isolation valves will be opened and treatment of the contaminated waste stream will begin at low flow rates until system integrity and acceptable outlet water quality are verified.

6. The flow rate to the demineralizer units will be increased on a gradual basis until the operational flow rate of approximately 5 gallons per minute per train is attained.
7. When the ion-exchange bed becomes depleted, the unit will be purged with processed water to ensure that radioactive waste water in the system piping is purged prior to disconnecting the quick disconnects on the demineralizer unit.
8. The demineralizer unit is dewatered prior to storage.
9. The demineralization unit will be decoupled remotely via the use of quick disconnects and will be stored in the spent fuel pool or loaded directly into a cask.

Chapter 5
System Description and Arrangement

5.1 Demineralizer System

5.1.1 Influent Water Filtration

A schematic illustration of the waste water influent system is shown in Fig. 5.1. Waste water enters the SDS through a permanently installed pump having a capacity of 50 gpm, 100 psig maximum pressure. During routine operation of the system, water passes through two filters in series before entering the four feedwater storage tanks, each having a capacity of 15,000 gallons. The purpose of the filters is to filter out solids in the untreated water before it is processed by the ion exchangers. Both filters are cartridge type and are protected by perforated metal screens. The design of the prefilter includes a 3/16 inch roughing screen and 125 micron mesh screen. The final filter is designed for particle size removal of 10 microns. Flow capacity through each filter is 50 gpm. Reverse flow is prevented by a check valve in the supply line.

Each filter is housed in a containment enclosure to enable leakage detection and containment. The filters are submerged in the spent fuel pool for shielding considerations. Contaminated water is pumped through the filters and into the feed tanks on a batch basis.

Influent waste water may be sampled from a shielded sample box located above the water level to determine the activity of contaminated water prior to and following filtration.

Inlet, outlet, and vent connections on the filters are made with quick release valve couplings which are remotely operated from the top of the pool. A gamma monitor, located adjacent to the filters, and inlet-outlet pressure gauges are provided to monitor and control solids loading. Load limits for the filters are based on filter differential pressure and/or the

surface dose limit for the filter cask. A flush line is attached to the filter inlet to provide a source of water for flushing the filters prior to removal.

5.1.2 Feed Tank System

Following filtration, waste water is pumped directly into the four 15,000 gal. storage tanks located in the tank farm (see Fig. 5.1). The tanks are equipped with a vent line connected to the off-gas treatment system. Water level in the tanks is monitored by level indicators.

A primary feed pump is submerged in a common well of the tank system. This pump discharges to the ion exchange system. Mechanical and electrical connections are designed for easy removal and rapid replacement of the pump should malfunction occur during operation. The discharge of the pump flows through piping in a shielded enclosure at a rate of 5-15 gpm and is monitored remotely by a pressure instrument and a radiation level monitor.

5.1.3 Supply Manifold

A flow diagram of the supply manifold and primary ion-exchange columns is shown in Fig. 5.2. This system consists of six underwater columns (24 in. x 54 in.), each containing seven cubic feet of Ion Siv IE-95 zeolite resin and two underwater columns containing organic cation resin. The six zeolite resin beds are divided into two trains each containing three resin beds (A, B, C) with piping and valves provided to operate either train individually or both trains in parallel.

The effluent from the zeolite beds flows through a cation exchange bed for removal of residual radionuclides. An in-line radiation monitor measures the activity level of the water exiting the cation exchangers. The valve manifold for controlling the operation of the primary ion exchange columns is located above the pool, inside a shielded enclosure that con-

tains a built-in sump to collect leakage that might occur. Any such leakage is routed back to the feed tank well. A line connects to the inlet of each primary exchanger to provide water for flushing the exchangers when they are loaded. Loading of resin columns is determined by analyzing the effluent from each exchanger through a sampling manifold, in conjunction with monitoring provided by a beta detection instrument. Waste water flow is measured by instruments placed in the line to each ion-exchange train.

5.1.4 Leakage Detection and Processing

Each submerged component is located inside a separate, special containment box that is filled with water from the pool. The box is designed with a divider inside for storage of the flexible hose connections to which the quick-disconnect couplings attach. Pool water from the containment boxes is continuously monitored to detect leakage and circulated by a pump through one of the two leakage containment ion-exchangers. Any leakage which occurs during routine connection and disconnection of the quick-disconnects will be captured by the containment boxes, diluted by pool water, and treated by ion-exchange before being returned to the pool.

5.1.5 Polishing Ion Exchangers

An effluent polishing unit (Fig. 5.3) provides final treatment of water after it passes through the cation exchanger. This unit consists of a 195 cubic foot mixed bed polishing demineralizer and has provisions for utilizing a 10 cu. ft prefilter. The purpose of the polishing unit is to remove trace fission products that may be present in the water. The polishing unit is located above the pool water level and is shielded by a cask during operation.

5.1.6 Monitoring Tank System

Effluent from the polishing exchangers flows into one of two monitoring tanks (Fig. 5.4). The purpose of the monitoring

tank system is to collect treated water and allow sampling prior to transfer. Each monitor tank is equipped with a sparger and tank level indicators that will automatically shut the inlet to the tank should a high level condition exist. Water in the monitoring tanks and can be transferred back for reprocessing or directed for hold up and final disposition.

5.1.7 Off-Gas and Liquid Separation System

An off-gas and liquid separation system collects gaseous and liquid wastes resulting from the operation of the water treatment system. The off-gas system is illustrated in Fig. 5.5. Inlet lines are connected to the feed tanks, monitoring tanks, ion exchangers, sampling manifold, and main feed supply manifold. Gaseous effluent is passed through a mist eliminator in the off-gas separator tank before being treated by an electric off-gas heater to remove residual vapors. Roughing filters, HEPA filters, and charcoal filters, are provided for further treatment. Air is moved through the system by a centrifugal blower rated at 1000 cfm. The discharge of this blower will be monitored and routed to the existing ventilation system. A pressure control regulator controls ventilation system pressure automatically. Moisture collected by the off-gas system and waste returned from the continuous radiation monitoring system is directed into a separator tank. At the top of the tank a mist eliminator separates moisture from effluent gas returned to the off-gas treatment system. The tank is located in the surge pit and is covered with a concrete shield. The level in the tank will be controlled automatically with level indicators that activate a pump to return collected water to the feed tanks.

5.2 Sampling and Radiation Monitoring System

The sampling manifold is located in a shielded enclosure to allow water samples to be taken for analysis of radionuclides and other contaminants (Fig. 5.6). Samples may he taken of the effluent from each of the zeolite resin beds and from the influent and effluent of the cation ex-

changer in service. The piping entering the manifold contains cylinders that permit draining a predetermined amount of sample into a collection bottle. Cylinders are purged by positioning valves to permit the water to flow through them and return to a waste drain header and into the off-gas separator tank. A water line connects to the inlet of the sample cylinders to allow the line to be flushed after a sample has been taken. The entire sampling manifold is located in a double partitioned glove box to minimize the possiblity of inadvertent leakage and spread of contamination during routine operation. Prior to entering the sampling manifold, a stream of water effluent from each resin bed can be selected to enable monitoring for resin breakthrough. This monitoring system provides a continuous indication of the level of contamination in water exiting each ion exchanger. The radiation monitoring instrumentation is connected to an alarm system that annunciates in case of a leak or a breakthrough in the resin beds.

5.3 Ion-Exchanger and Filter Vessel Transfer in the Fuel Storage Pool

Prior to system operation, ion exchanger and filter vessels are placed inside the containment boxes and connected with quick-disconnect couplings. When it is determined that a vessel is loaded with radioactive contaminants to predetermined limits, vessel couplings are removed with a mono-rail lifting device and a special tool for remote operation of the fittings.

Vessels are transferred using the existing fuel handling crane that is fitted with a yoke attached to a long shaft (Fig. 5.7). The purpose of this yoke-arm assembly is to prevent inadvertent lifting of a resin bed to a height greater than eight feet below the surface of the water in the pool. This device is a safety tool that will mechanically prevent the possibility of accidental exposure of operating personnel to a loaded exchanger or filter vessel.

The ion-exchange vessels are arranged to provide series processing through each of the beds; the influent waste water is treated by the bed in position "A", then by the bed in position "B", and finally by the bed in position "C".

The first vessel in each train (position A) will load with radioactive contaminants first, then the next two vessels will be moved up to the "A" and "B" positions respectively and an unused vessel put in the "C" position. The loaded vessel will then be stored until transfer to the cask. At no time during the operation of the system will a loaded vessel be taken out of the pool before it has been placed in a shipping cask. The shipping cask will be transferred from the pool with the overhead crane.

5.4 Arrangement of the Water Treatment System in the Fuel Storage Pool

Figure 5.8 illustrates the arrangment of the SDS in the fuel storage pool (viewed from above). The feed tanks and feed pump are located at the south end of the pool and are covered with concrete slabs. The filters, primary resin beds, and cation beds are located underwater in containment enclosures. These enclosures and the exchangers are supported along one side of the pool on a structural steel rack that is attached to the edge. The rack acts as a support for the system and also provides an operating platform from which the remote connections can be made. The off-gas system is mounted on the wall near the cask pool and surge tank area.

A dewatering station is located below the water level in the pool and is used for displacing the water from expended columns and filters and drying them prior to storage. An underwater storage rack, designed to handle 40 expended vessels is located on the west side of the pool. This capability allows processing to continue without interruption due to handling operations.

5.5 Solidification Capability

The capability to solidify the spent resin generated by the operation of the SDS has been developed in the event that expended resin solidification is required. This process involves insitu solidification of the 10 cubic feet vessels utilizing masonry cement as the solidification agent to produce a solid homogeneous freestanding monolith. The solidification evolutions would be performed inside a special shielded cubicle equipped with remote handling capabilities. To reduce personnel exposures the 10 cubic feet vessels would remain in the shielded cask throughout the solidification and handling process.

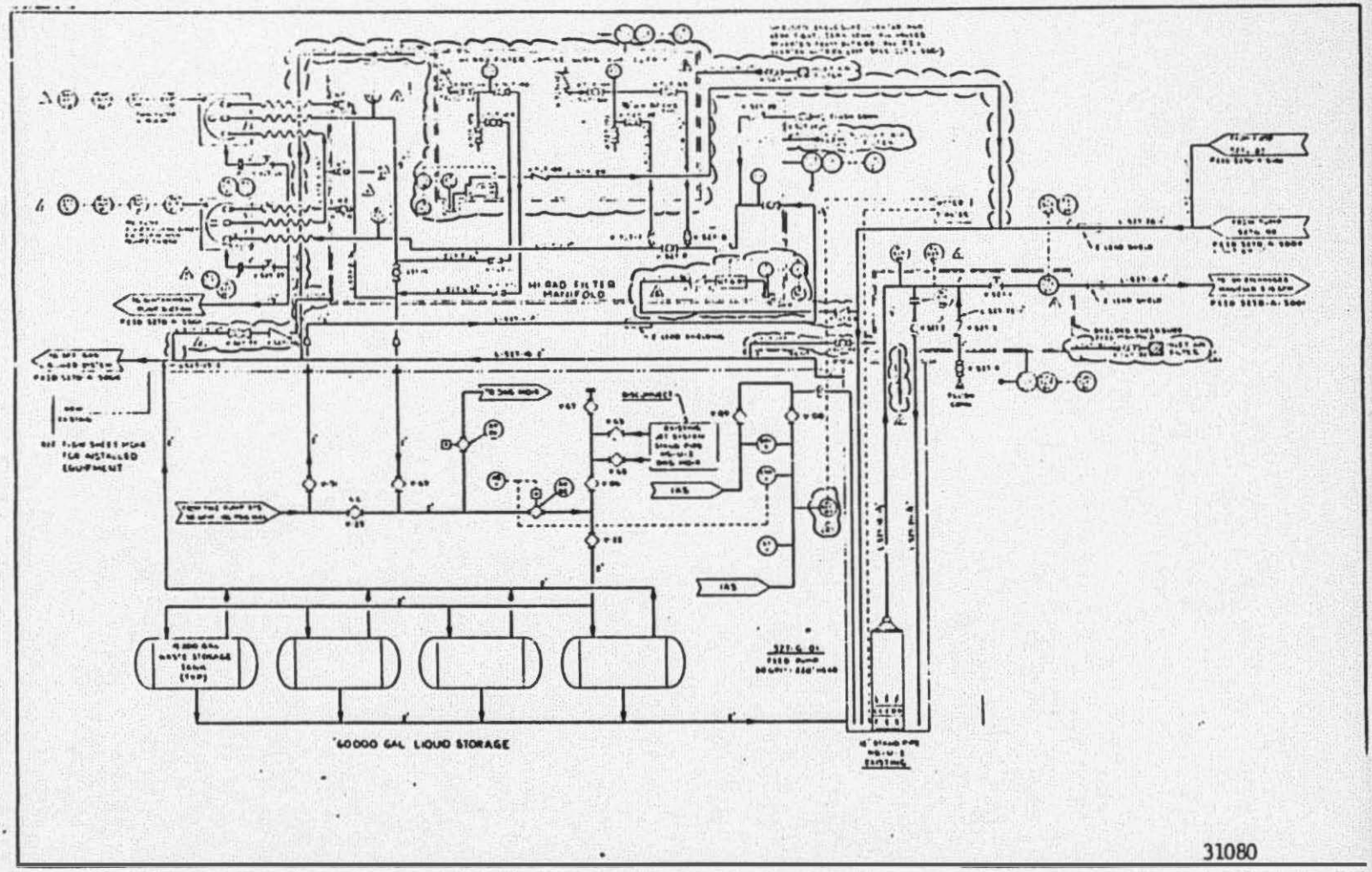

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FIGURE 5.1
WASTE WATER INFLUENT SYSTEM

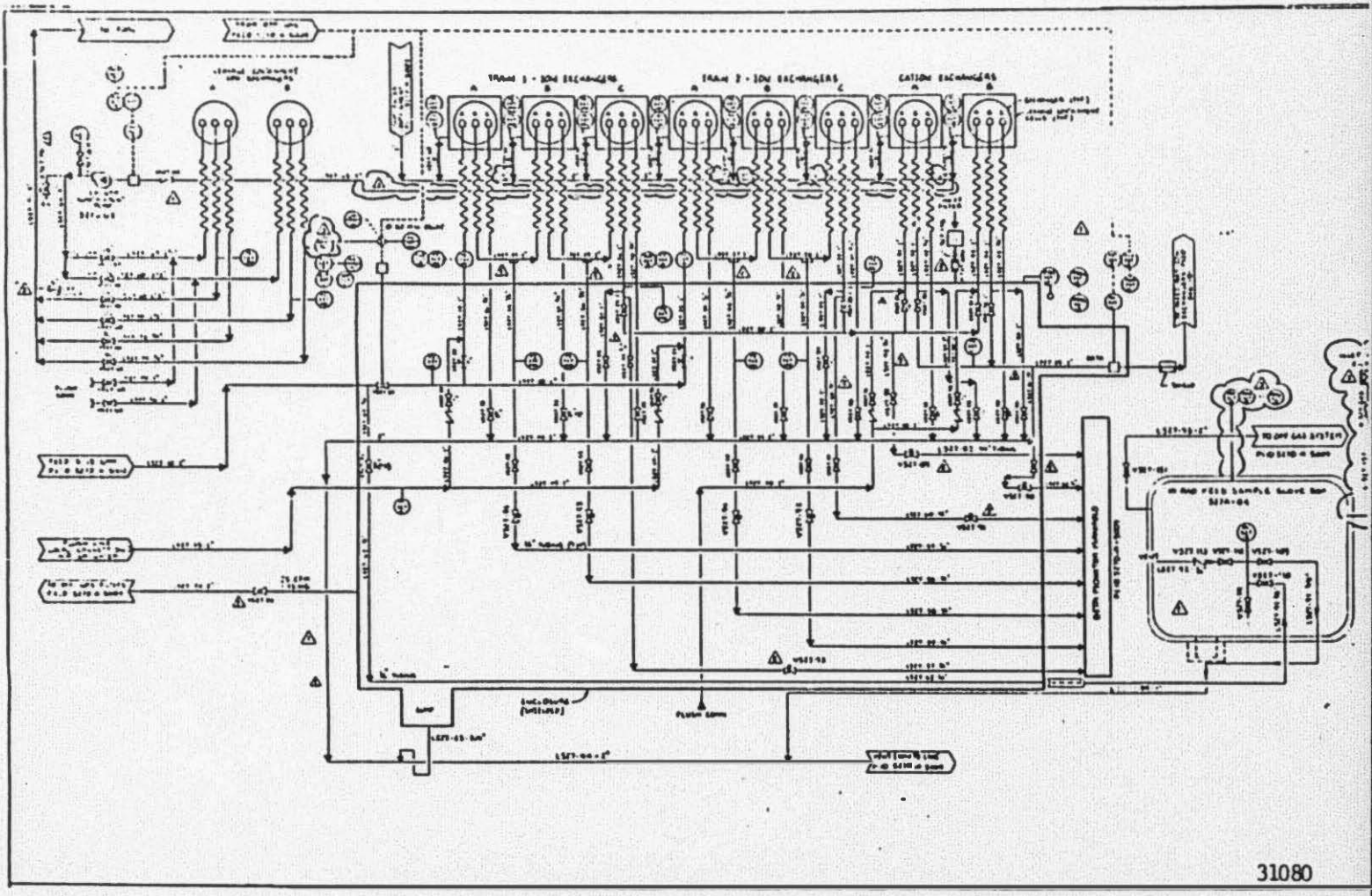

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FIGURE 5.2

ION EXCHANGE TRAINS

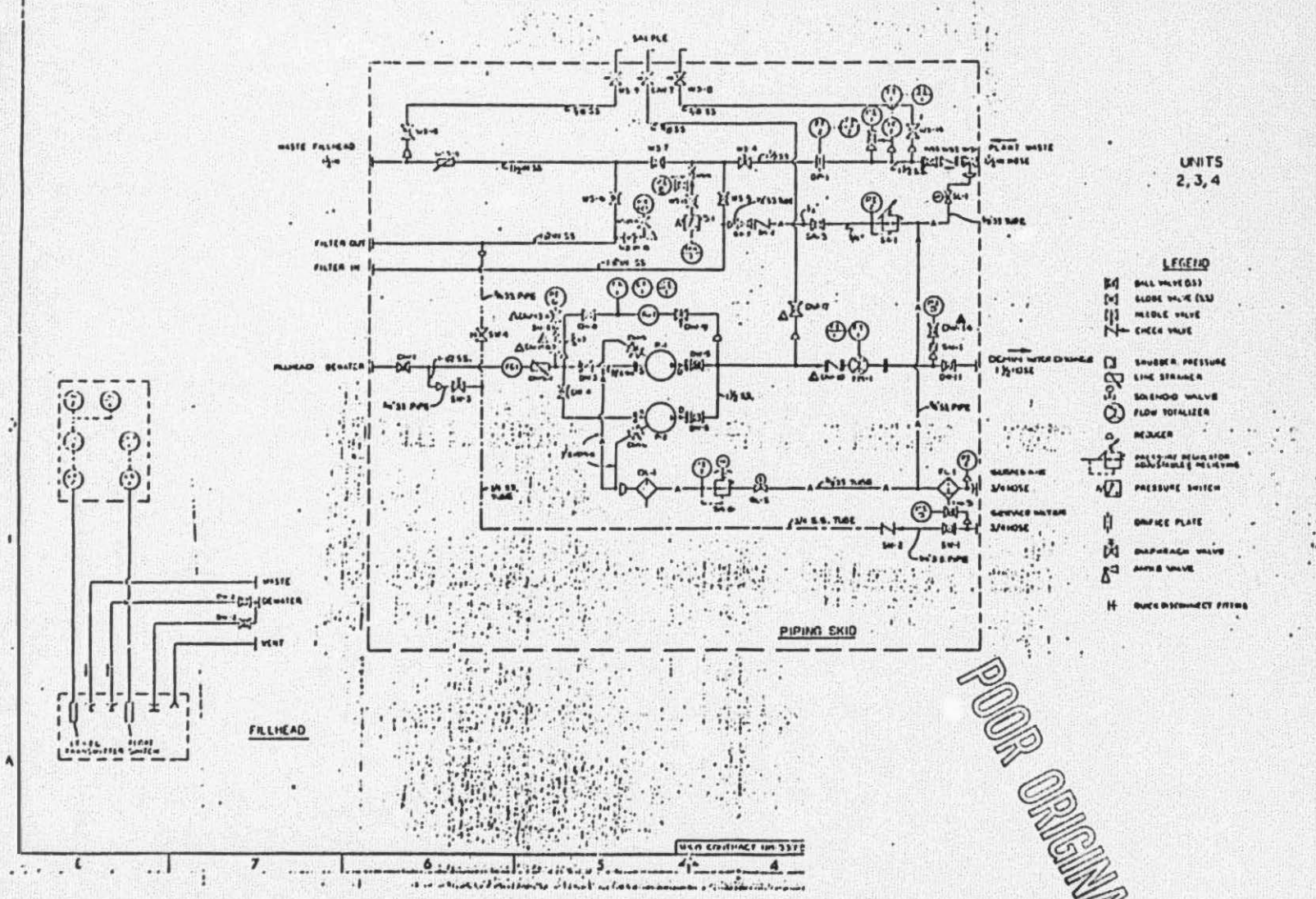

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Figure 5.3 EFFLUENT POLISHING UNIT

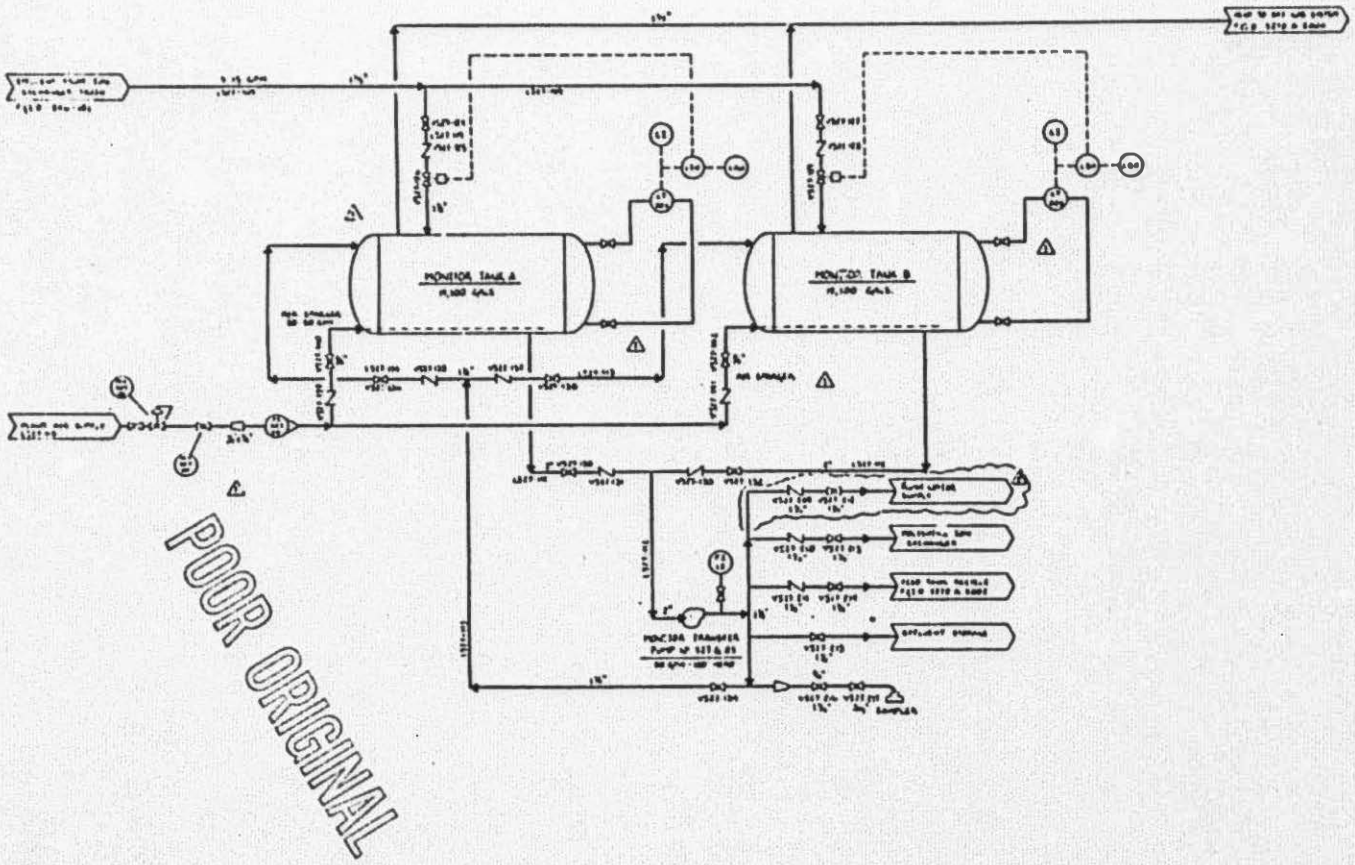

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FIGURE 5.4
MONITORING TANK SYSTEM

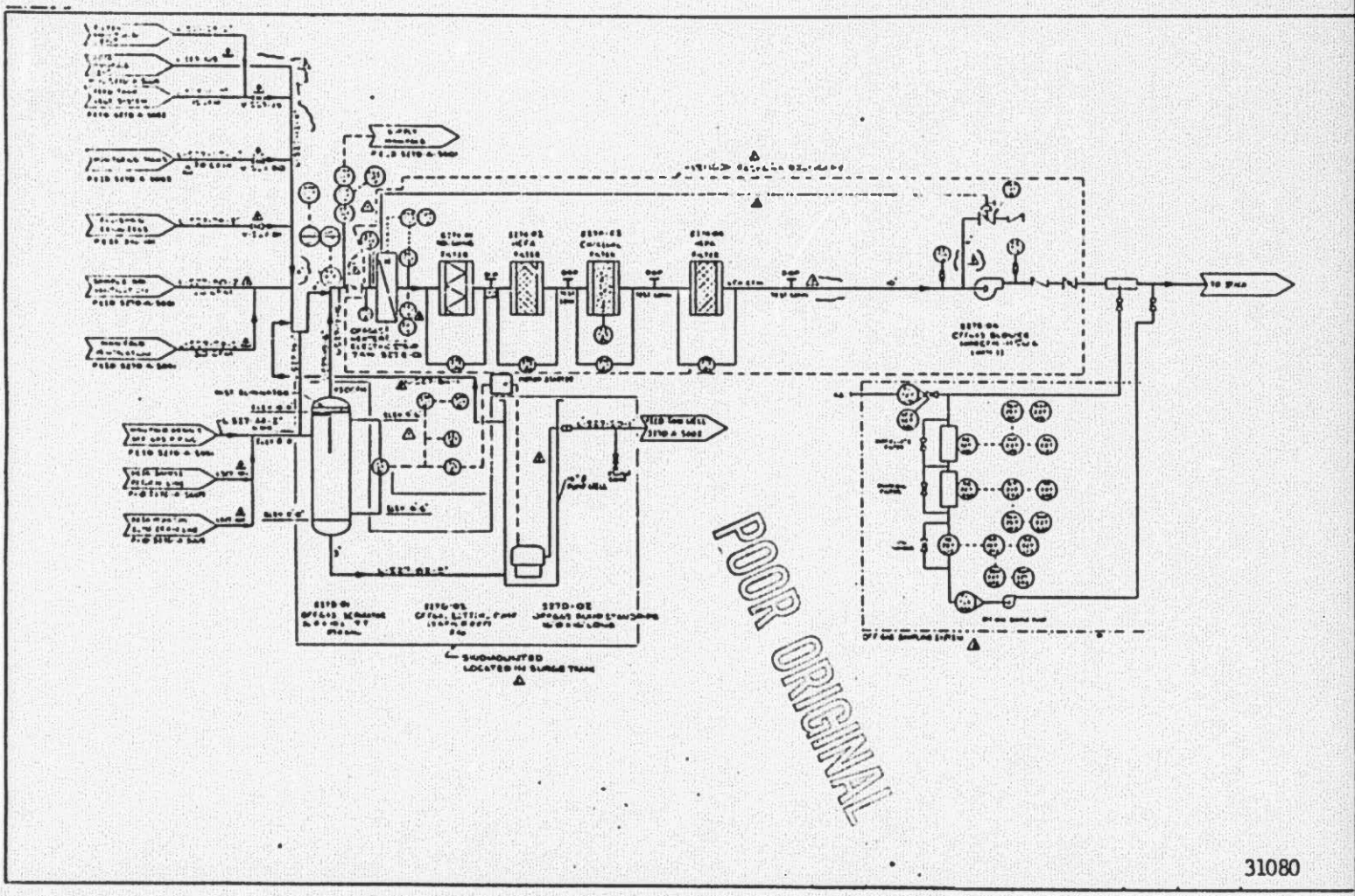


FIGURE 5.5

OFF-GAS TREATMENT SYSTEM

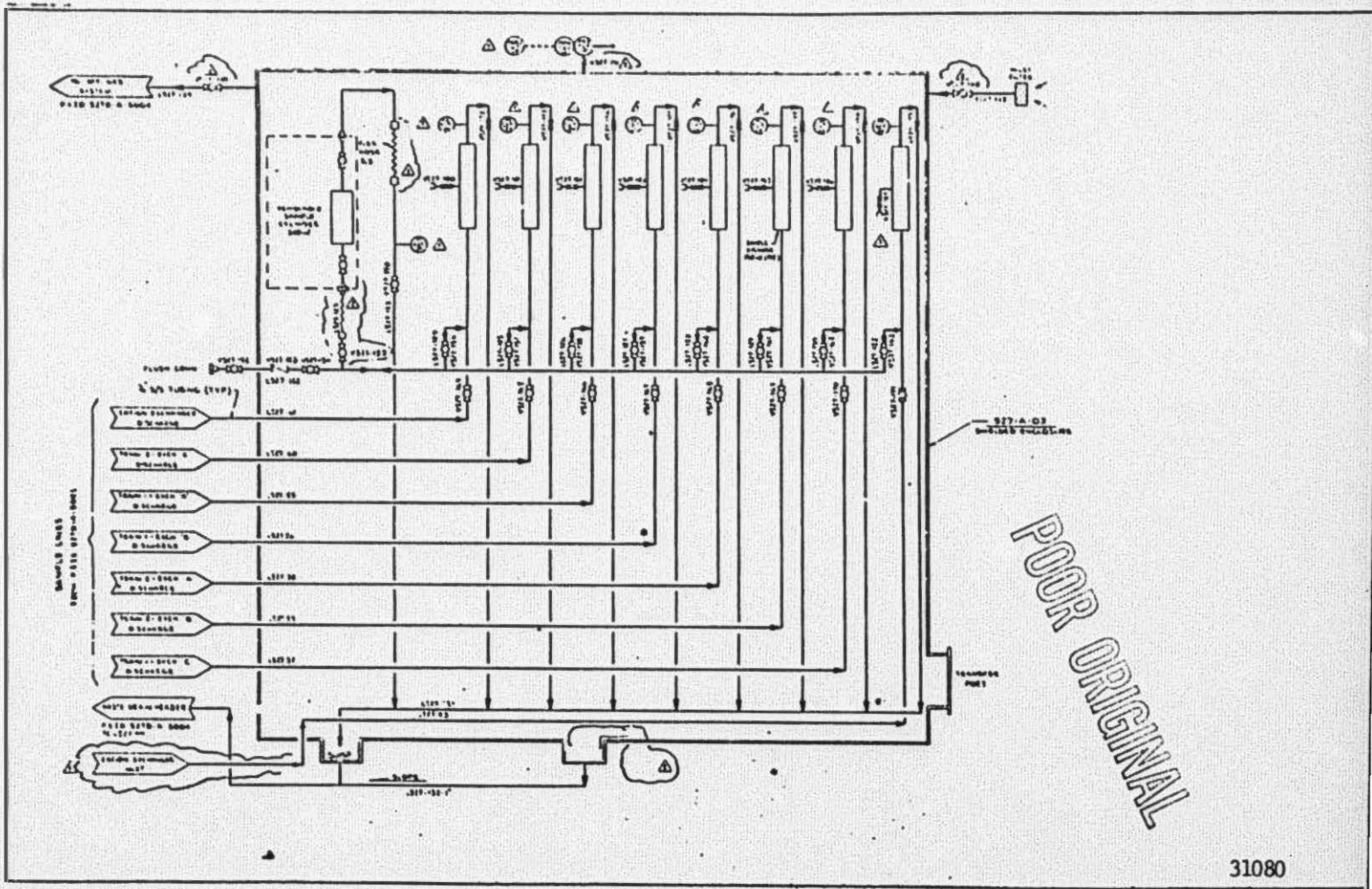


FIGURE 5.6
SAMPLING SYSTEM

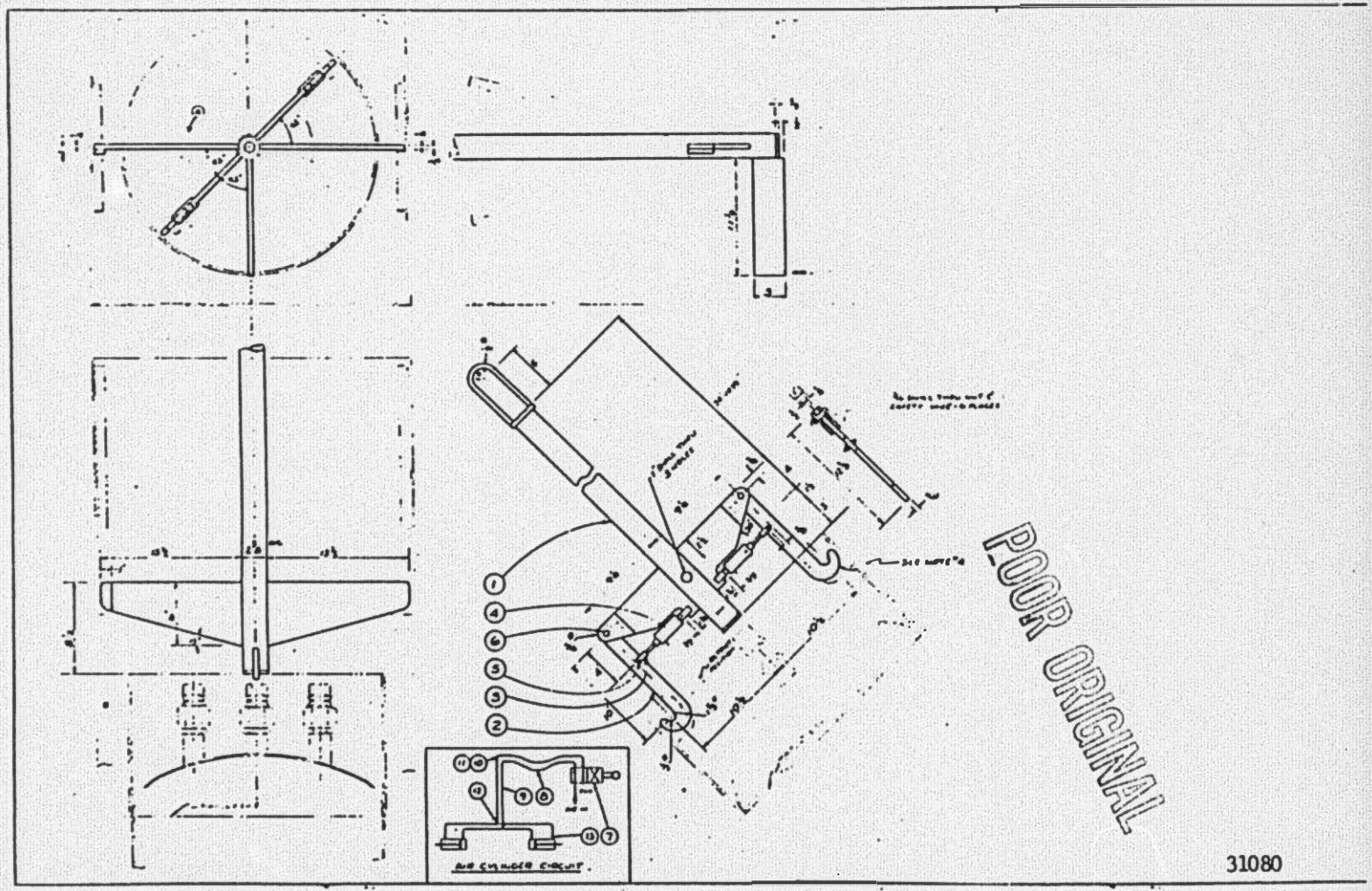
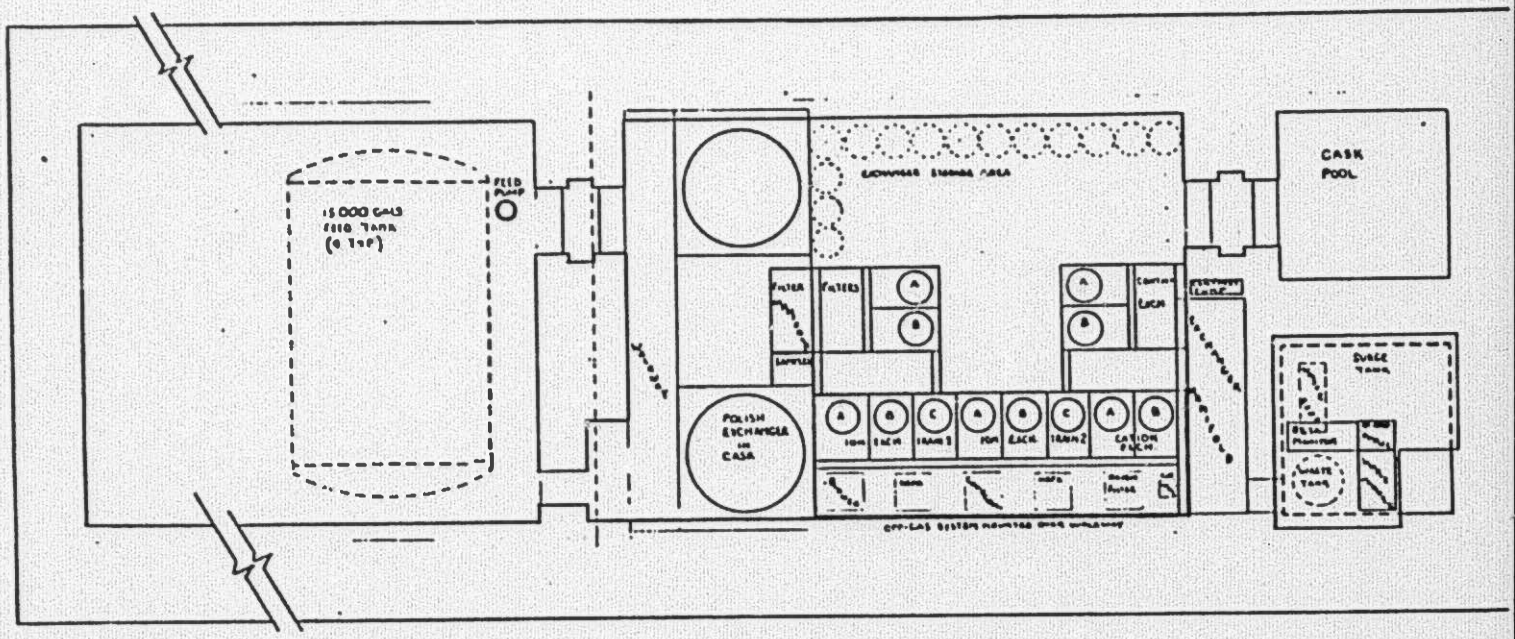


FIGURE 5.7 VESSEL HANDLING TOOL



GENERAL LAYOUT PLAN

Figure 5.8 Submerged Demineralization System

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Chapter 6
Radiation Protection

6.1 Ensuring Occupational Radiation Exposures are ALARA

6.1.1 Policy Considerations

The objectives of the Radiological Controls Department are to insure that operations conducted in support of the on-going demineralization program are conducted in a radiologically safe manner, and further, that operations associated with radiation exposure will be approached from the standpoint of maintaining radiation exposure to levels that are as low as reasonably achievable.

During the operational period of the system, the effective control of radiation exposure will be based on the following considerations:

1. Sound engineering design of the facilities and equipment.
2. The use of present radiation protection practices, including work task planning for the proper use of the appropriate equipment by qualified personnel.
3. The assignment of radiological control supervisor, solely to this operational evolution.
4. Strict adherence to the radiological controls procedures as developed for TMI-2.

6.1.2 Design Considerations

The SDS was specifically designed to maintain exposure to operating personnel to as low as reasonably achievable. To implement this concept the components carrying high level activity water will he provided with additional shielding or are submerged in the fuel spent pool. Shielding has been designed to limit whole body exposure rates in operating areas to approximately 1 mR/hr. In addition, components carrying high level process fluids

have been designed for exhaust to the SDS off-gas system. This method of off-gas treatment will minimize the potential for airborne releases in the work areas.

The specific design features utilized in meeting this requirement are discussed in detail in Section 6.2.1.

6.1.3 Operational Considerations

The system design reflects the following operational ALARA considerations:

1. Exposure of personnel servicing a specific component on the SDS will be reduced by providing shielding between the individual components that constitute substantial radiation sources to the receptor.
2. The exposure of personnel who operate valves on the SDS will be reduced through the use of reach rods.
3. Controls for the SDS will be located in low radiation zones.
4. Airborne radioactive material concentrations will be minimized by routing the off-gas effluent from the SDS to the TMI ventilation system for further treatment.
5. The sampling stations for the feedstream and filters that contain high levels of radioactive materials will be exhausted through the SDS ventilation system.
6. The sampling manifold is located in a double partitioned glove box to minimize the possibility of inadvertent leakage and spread of contamination during routine operation.
7. The SDS is being fabricated with surfaces that are smooth, nonporous, and free of cracks, crevices, and sharp corners to the level that is practically achievable. This type of finish will minimize personnel exposures incurred in decontamination of the system.

6.2 Radiation Protection Design Features

6.2.1 Facility Design Features

The system is designed to take maximum advantage of station features already in place and operational in terms of protection of the public. In addition, design features provided by the system offers are intended for the reduction of releases of radioactive material to the environment. The following features provide for protection of individuals from radiological hazards during normal operations from external exposure and unanticipated operational occurances, such as spills.

1. The SDS primary demineralization units are housed under approximately 20 feet of shielding water in the TMI-2 spent fuel pool.
2. The entire process and all equipment is housed in the Auxiliary and Fuel Handling Building which is a Seismic Category I structure with air handling and ventilation systems designed to mitigate the consequences of radiological accidents.
3. The system is designed in such a manner as to allow zero discharge of liquid effluents and operated such as to reduce the average isotopic specific activity of the treated waste streams to concentrations equivalent to levels normally required for discharge into the environment.
4. The off-gas system will be treated, filtered and monitored before input to existing ventilation exhaust systems.
5. Filters, primary ion-exchange beds, cation beds, and their associated couplings are contained in containment devices. Each containment device is connected to a pump manifold and a continuous flow of approximately 10 GPM is maintained through each containment. The combined flow from the ten (10) containments (100 GPM total) is then processed through a mixed bed resin column and then discharged back to the spent fuel pool.

6. Loaded demineralizer units will be placed in shielded casks underwater to preclude exposure of the concentrated waste product to operating personnel.
7. To the extent possible all-welded stainless steel construction is specified to minimize the potential for leakage.
8. Lead or equivalent shielding is provided for pipes, valves, and vessels (except those located under water) where necessary for personnel protection.
9. Design of a sequenced multi-bed process - three (3) beds in series to preclude breakthrough and contamination of the outlet stream.
10. Feed filter system is designed with appropriate pressure indicators, and inlet, outlet and vent connection are made with remote operated-valve quick release couplings.

6.2.2 Shielding

The minimum shielding thickness required for radiological protection has been designed to reduce levels in occupied areas to less than 1 mR/hr. Operating panels and instrumentation racks are located away from potential sources of radiation.

All movements of the demineralizer out of the fuel pool will be performed utilizing a shielded transfer cask.

6.2.3 Ventilation

The ventilation and off-gas system provided to service the SDS is designed to minimize gaseous releases. Among these design features are:

1. Automatic level controlled off-gas separator tank with mist eliminator to receive vent connections from the feed tank system, monitoring tanks, polishing demineralizers, sample boxes, and piping manifold.
2. Electric off-gas heater for maximum charcoal bed efficiency.

3. Roughing filter with differential pressure indication.
4. A HEPA filter prior to the charcoal bed with differential pressure indication.
5. A charcoal adsorber bed with temperature and differential pressure indication.
6. A HEPA filter after the charcoal bed with differential pressure indication.
7. A centrifugal off-gas blower with flow indication.
8. Sample ports for monitoring the system and DOP test ports for HEPA testing.
9. The effluent of the SDS off-gas system will be routed to the existing TMI-2 ventilation system exhaust, which is filtered again through HEPA and charcoal filters prior to discharge from the plant.

6.2.4 Area Radiation Monitoring Instrumentation

General area radiation monitors have been provided for, which can be utilized to alert personnel of increasing radiation levels during normal operations or maintenance activities.

6.3 Dose Assessment

6.3.1 On-site Occupational Exposures

Normal Operation

During the operation of the Submerged Demineralization System, there are operations that involve occupational exposures, but precautions have been taken in the design stage to minimize personnel exposures. Major operational activities involving such exposures are as follows:

A. Feed tank filling valve alignment
B. Feed sampling operation in high radiation filter box
C. System start-up valve alignment
D. Effluent sampling operations
E. System shut-down operation
F. Cask removal, decontamination and survey operations
G. System maintenance

Decommissioning

The SDS detailed decommissioning plan is being developed in conjunction with the operating procedures for the system. However, the modular design of the system is conducive to disassembly while minimizing exposure to personnel.

6.3.2 Off-site Radiological Exposures

Source Terms for Liquid Effluents

Liquid effluent from the system will be returned to station tankage for further disposition, therefore, no liquid source term is required for this report.

This SDS is designed and operated to reduce the average isotopic specific activity of each of the treated waste streams to concentrations equivalent to levels normally required for discharge into the environment.

Source Terms for Gaseous Effluents

The plant vent system is the only off-gas stream carrying airborne radioactive material, therefore, the only potential pathway for gaseous release. Radionuclides in the gaseous effluent arise from entrainment during transfer of contaminated water to various tanks, filters, ion-exchange units, and also from water sampling.

Gaseous effluent source terms were conservatively developed by assuming the system operated on the principle of evaporation. For this reason an entrainment factor of 10^{-6} is assumed from the liquid to the vapor (Gray et al., 1979, RHFP, 1976). In the case of evaporation by boiling, a higher rate of release of radionuclides with off-gas vapors occurs than would be expected from routine operation of pumps, valves, and water transfer. Therefore,

it is considered conservative to assume an entrainment factor of 10^{-6} for the solution-vent system during pump transfer of water.

It should be noted that there are several vent systems which comprise the final off-gas stream, some of which have a lesser potential for contamination. However, again for conservatism, it is assumed that the total 650 cfm has been in contact with water in the containment, which at the time of this evaluation, contains the highest specific activity of radionuclides.

The level of contamination of water in the containment sump is listed in Table 6.1. These data are based on measured values reported in Chapter 1 of this report. The pumping rate of water to the cleanup system is assumed to be 10 GPM ($3.785 \times 10^{+4}$ ml/min). From the assumed entrainment factor the amount of radioactivity introduced into the off-gas is (3.785×10^{-2}) (f_i)Ci/min where f_i is the activity of an isotope per ml.

For the SDS, a decontamination factor (DF) of 100 is assumed for the HEPA filters and a decontamination factor of 40 is assumed for I-129 for the charcoal filters (Finney et al. 1977). An additional decontamination factor of 100 is assumed for the existing filters at TMI resulting in a total DF of 10^4.

As an example, the calculation of the amount of CS-137 in the effluent gas from the SDS using the concentration in the liquid given in Table 6.1 is shown below.

$$\frac{10 \text{ gpm} \times 3.785 \times 10^{3} \text{ ml/gal} \times 176 \text{ uCi/ml} \times 10^{-6} \text{ (entr. fact)}}{650 \text{ cfm} \times 2.8 \times 10^{4} \text{ ml/cf} \times 100 \text{ (DF)}} =$$

$$3.66 \times 10^{-9} \text{ uCi/cc}$$

Each source term is corrected for decay to a projected start-up date of October 1, 1980. The release of tritium is calculated by assuming the air discharged from the vent was saturated with water vapor at 80°F. At this temperature 650 ft^3/min of air would carry 500 gm of water vapor and correlates to 2.66 x 10^{-5} uCi/cc of the tritium isotope.

No calculations were made for Nb-95, La-140 and I-131 as the levels of these isotopes will be inconsequential by the time SDS operations start.

Table 6.2 lists the concentration of radionuclide source terms in the off-gas following treatment by the system and the existing effluent treatment system at TMI. Release rates for the various radionuclides are also shown. As can been seen by Table 6.2, the concentrations in the plant effluent are below detectable levels for all isotopes except H-3.

Methodology

The radiological impact of the SDS is assessed by calculating radiation doses to individuals and populations living in the vicinity of the Three Mile Island Nuclear Generating Station. Potential pathways for internal and external exposure to man from radionuclides released to the atmosphere include inhalation, ingestion of contaminated foods, ingestion of contaminated water, exposure from contaminated surfaces, and exposure from immersion in the plume.

Radiological impact is estimated using the methodology proposed in Regulatory Guide 1.109 (USNRC, 1977). The dose from a specified intake of a radionuclide to a reference organ is calculated over the remaining lifetime of the individual. The exposed person is assumed to be

an adult (20 years of age) at the time of intake who will live to an age of 70 years. Thus, the accumulated dose is calculated by integrating the dose rate over a 50-year period, and the result is called the 50-year dose commitment.

For the purpose of calculating dose to the maximally exposed individual and to the population from operation of the SDS, X/Q (sec/m^3) values were taken from previously published data and updated to 1980. The data are calculated for a semi-elevated point of release including building wake effects. The values for X/Q for each of the sixteen sectors of the compass and downwind distance from the point of release are listed in Table 6.3.

Radioactive particulates are removed from the atmosphere and deposited on the ground through mechanisms of dry deposition and scavenging. Dry deposition represents an integrated deposition of radioactive materials by processes of gravitational settling adsorption, particle interception diffusion, and chemical-electrostatic effects and is calculated from the deposition velocity, V_d, for a one-year time interval. Deposition velocity values for particles and reactive gases commonly range from 0.1 to 6.0 cm/second (Moore et al., 1979). In this assessment a value of 1.0 cm/second has been selected for calculation of ground concentrations of radioactive particulates ^{89}Sr, ^{90}Sr, ^{125}Sb, ^{134}Cs, and ^{137}Cs. It is further assumed that radioiodine is released in molecular form and that a deposition velocity of 1.0 cm/sec is applicable to ^{129}I (Moore et al, 1979).

Scavenging of radionuclides in the plume is the process through which rain or snow washes out particles or dissolves gases and deposits them on the ground or water surfaces. In this assessment, however, the effects of scavenging have not been included based upon the methodology proposed in Regulatory Guide 1.111 (USNRC, 1976).

Organ doses may vary considerably for internal exposure from ingested or inhaled materials because some radionuclides concentrate in certain organs of the body. This assessment calculates the dose to four organs: total body, bone, thyroid, and G.I. tract.

Radiation doses to the internal organs of children in the population vary from those received by an average adult because of differences in metabolism, organ size, and diet. Differences between the organ doses of a child and those of an average adult by more than a factor of three would be unusual for all pathways except the atmosphere-pasture-cow-milk pathway (^{129}I) (Soldat, 1965). For this pathway the estimated dose to the thyroid of a one-year-old child from radioiodine in milk is approximately five times the average adult. Since the contribution to total dose from ^{129}I is small in this study, age-dependency has not been incorporated into the calculation of dose (Schkien, 1970).

Total dose commitments are calculated for the specified amount of each isotope released during a one-year period of continuous release. Several conservative assumptions are made which tend to make dose commitments higher than what would actually occur. For example, usage factors for the maximally exposed individual are taken from Regulatory Guide 1.109, Table E-5. It is also assumed that all vegetables, both leafy and non-leafy, are grown at the point where dose is calculated and that an individual lives outdoors at the reference location 100% of the time. Since there are no releases via liquid effluent it is assumed that the dose from ingestion of contaminated water is negligible. Additional details regarding assumptions made and the methodology used can be found in Regulatory Guide 1.109.

Analysis of Maximum Individual Dose

The maximum dose to a hypothetical individual is calculated for the four organs and assumes 365 days of system operation. These estimated dose exposure levels are presented below.

Bone	4.4×10^{-3} mrem
Total Body	3.6×10^{-3} mrem
Thyroid	3.6×10^{-3} mrem
GI Tract	3.0×10^{-3} mrem

This dose exposure to the total body represents only 0.072% of the allowable dose exposure recommended in 10 CFR 50, Appendix I, of 5 mrem.

Table 6.4 lists the contribution of the various exposure pathways to the dose of each organ considered. Ingestion of contaminated foods is the primary mode of exposure, contributing 80% of the dose to total body, 83% to bone, 79% to thyroid, and 76% to GI tract. Inhalation is the second most important pathway while external exposure contributes less than 1% to each organ.

The contribution from each radionuclide to total dose is shown in Table 6.5. Tritium is the primary contributor to each organ, giving aprroximately 83% of the dose to total body and thyroid and 98% of the dose to GI tract. Other contributing radionuclides are ^{90}Sr, ^{134}Cs, and ^{137}Cs. Strontium-89, antimony-125, and iodine-129 do not contribute significantly to the dose to any organ.

Even with the conservative assumptions incorporated into this assessment it is evident that the estimated dose to the maximally exposed individual is acceptable and meets recommended criteria for exposure to the public.

Analysis of Population Dose

The estimated radiological exposure to the population from continuous operation of the SDS (365 days/year) is calculated using the methodology outlined in this report section (6.3) as specified in Regulatory Guide 1.109. The population distribution is based on recent demographic data (1980) to a radious of 50 miles from the TMI site.

Calculations that have been performed are based on continuous operation (365 days/year) of the SDS. Even though this conservative assumption was used, the population dose is calculated to be 0.12 man-rem.

Table 6.1

Contamination Level of Water in Containment

(February, 1980)

Isotope	uCi/ml
Sr-89	40.
Sr-90	2.7
Cs-134	40.
Cs-137	176.
Nb-95	.0021
La-140	.036
I-131	.012
I-129	.000013
Sb-125	.012
H-3	1.03

Table 6.2

Source Terms for Gaseous Effluents

(As of October 1, 1980)

Radionuclide	Concentration * In SDS Effluent uCi/cc	Concentration ** In Plant Effluent uCi/cc	Release Rate uCi/sec
H-3	2.62×10^{-4}	1.68×10^{-8}	7.98×10^{-1}
Sr-89	3.70×10^{-12}	2.38×10^{-15}	1.13×10^{-8}
Sr-90	5.55×10^{-11}	3.58×10^{-15}	1.70×10^{-7}
Cs-134	5.62×10^{-10}	3.62×10^{-14}	1.72×10^{-6}
Cs-137	3.66×10^{-9}	2.34×10^{-13}	1.11×10^{-5}
Sb-125	2.00×10^{-13}	1.29×10^{-17}	6.14×10^{-10}
I-129	2.80×10^{-16}	1.81×10^{-20}	8.60×10^{-13}

* Based on 650 CFM; 100 DF for SDS Filters

** Based on Total Exhaust from the Plant of 100,650 CFM; DF of 100 from Existing Filters

TABLE 6.3
ATMOSPHERIC DISPERSION FACTORS FOR TMI

	DISTANCE (Meters)									
SECTOR	610	2413	4022	5631	7240	12067	24135	40225	56315	74205
N	4.23E-05	5.65E-06	2.72E-06	1.69E-06	1.19E-06	5.71E-07	1.31E-07	2.20E-08	1.44E-08	1.04E-08
NNE	4.46E-05	5.91E-06	2.85E-06	1.78E-06	1.25E-06	6.06E-07	1.40E-07	2.35E-08	1.54E-08	1.12E-08
NE	3.75E-05	3.74E-06	1.05E-06	5.48E-07	3.75E-07	1.66E-07	5.33E-08	2.37E-08	1.55E-08	1.12E-08
ENE	3.60E-05	3.63E-06	1.02E-06	5.33E-07	3.65E-07	1.62E-07	5.16E-08	2.30E-08	1.50E-08	1.09E-08
E	4.08E-05	4.12E-06	1.14E-06	5.96E-07	4.06E-07	1.78E-07	5.64E-08	2.50E-08	1.62E-08	1.17E-08
ESE	3.96E-05	3.95E-06	1.09E-06	5.70E-07	3.89E-07	1.71E-07	5.41E-08	2.40E-08	1.56E-08	1.12E-08
SE	4.12E-05	4.12E-06	1.13E-06	5.88E-07	4.00E-07	1.74E-07	5.48E-08	2.42E-08	1.56E-08	1.12E-08
SSE	5.57E-05	7.49E-06	3.56E-06	2.19E-06	1.53E-06	7.27E-07	1.64E-07	2.73E-08	1.77E-08	1.28E-08
S	3.89E-05	5.21E-06	2.49E-06	1.54E-06	1.08E-06	5.12E-07	1.16E-07	1.93E-08	1.26E-08	9.10E-09
SSW	2.50E-05	3.41E-06	1.62E-06	9.93E-07	6.93E-07	3.28E-07	7.40E-08	1.22E-08	7.93E-09	5.71E-09
SW	2.60E-05	2.63E-06	7.32E-07	3.92E-07	2.61E-07	1.15E-07	3.64E-08	1.62E-08	1.05E-08	7.59E-09
WSW	3.36E-05	3.35E-06	9.34E-07	4.89E-07	3.34E-07	1.48E-07	4.74E-08	2.11E-08	1.38E-08	9.96E-09
W	4.39E-05	4.43E-06	1.23E-06	6.41E-07	4.37E-07	1.92E-07	6.06E-08	2.68E-08	1.74E-08	1.26E-08
WNW	4.37E-05	4.35E-06	1.22E-06	6.40E-07	4.39E-07	1.95E-07	6.28E-08	2.80E-08	1.83E-08	1.33E-08
NW	4.16E-05	4.16E-06	1.17E-06	6.18E-07	4.24E-07	1.89E-07	6.10E-08	2.73E-08	1.78E-08	1.30E-08
NNW	4.04E-05	5.35E-06	2.59E-06	1.61E-06	1.14E-06	5.49E-07	1.27E-07	2.13E-08	1.39E-08	1.02E-08

Table 6.4

Contribution of exposure pathways to the dose of specific organs of the maximally exposed individual.

Pathway of Exposure	Total Body	Bone	Thyroid	GI Tract
		(% Contribution to dose)		
External Exposure[a]	<1	<1	<1	<1
Ingestion of Contaminated Food	80	83	79	76
Inhalation	20	17	20	24

[a] Includes exposure from contaminated ground surface and exposure from immersion in any plume.

Table 6.5

Contribution of specific radionuclides to the dose of organs of the maximally exposed individual.

Radionuclide	Total Body	Bone	Thyroid	GI Tract
	(% Contribution to dose)			
^{3}H	83	67	82	98
^{89}Sr	<1	<1	<1	<1
^{90}Sr	7	24	7	<1
^{125}Sb	<1	<1	<1	<1
^{129}I	<1	<1	<1	<1
^{134}Cs	8	3	8	<1
^{137}Cs	2	5	3	<1

Chapter 7
Accident Analysis

Because of the inherent safety features of the Submerged Demineralizer System and maximum utilization of existing site facilities, potential accidents which involve the release of radionuclides to the environment are minimized. Hypothetical accidents during system operation are proposed and evaluated in the following assessment.

7.1 Inadvertent pumping of containment water into the spent fuel pool.

Assumptions:

The effluent line from the final filter develops a leak and is not detected immediately. Contaminated water is released into the pool at a rate of 30 gpm for a period of 15 minutes, (450 gallons or 1350 curies).

It is assumed that the total activity is made up of Cesium, 250 Ci of Cs-134 and 1100 Ci of Cs-137 (based upon the measured concentrations as reported in Chapter 1). Analysis of the accident also assumes uniform mixing in 233,000 gallons of pool water and results in pool water contamination levels of 1.53 uCi/ml.

Occupational Exposure Effects:

The dose rate is calculated to an individual on the walkway at a point six feet above the surface of the water using equations for an infinite slab source (Rockwell, 1956) and published radionuclide decay data (USOHEW, 1970). The depth of water in the pool is 38 feet. The calculated maximum exposure rate at six feet above the surface is 430 mR/hr.

Off-site Effects:

Airborne contamination releases as a result of this hypothetical accident are a small fraction of the limits specified in 10 CFR 20, Appendix B.

No significant increases in the site boundary exposure level is expected as a result of this hypothetical accident due to the spent fuel pool configuration and inherent shielding properties of the pool side walls and the distance to the site boundary.

Conclusions:
This hypothetical accident is evaluated under conservative assumptions. Furthermore, this hypothetical accident has been performed under "worst case" conditions such that any other occurence that may cause leakage to the spent fuel pool has been conservatively bounded.

Although the analysis of this hypotetical accident provides results that indicate radiation field of 430 mR/hr at a level six feet above the pool surface, area radiation monitor alarms would indicate its presence. Personnel would be evacuated to ensure that occupational exposures are ALARA.

Off-site radiological consequences potentially resulting from this hypothetical accident are insignificant.

7.2 Pipe rupture on filter inlet line (above water level)

Assumotions:
A pipe rupture occurs in the inlet line to the filters above water level at the southeast corner of the pool. The leak proceeds for fifteen minutes before the pump is stopped. Contaminated water sprays from around the lead brick shielding. A total of 75 gallons of water is spread onto a surface area of 200 ft^2 and 675 gallons of contaminated water is drained into the pool. It is further assumed that the contaminated water contains three Ci/gallon of activity, as Cs-134 and Cs-137 in the same concentration ratios that were assumed for the previous hypotetical accident.

Occupational Exposure Effects:
As a result of this hypothetical accident, three significant effects are postulated:

1. The maximum gamma exposure rate at the surface of the contaminated floor area is estimated to be 10.8 Rem/hr.
2. The maximum beta exposure rate at a point three feet above the surface of the contaminated floor area is estimated to be 384 Rad/hr.
3. The exposure rate from the surface of the contaminated spent fuel pool waters, at a point six feet above the surface, would be approximately 650 mRem/hr.

Off-site Effects:

Airborne contamination releases at the site boundary as a result of this hypothetical accident are below those limits specified in 10 CFR 20, Appendix B.

The increase of exposure rate at the site boundary, as a result of this hypothetical accident, would not be significant due to the shielding characteristics of the fuel building walls and the distance to the site boundary.

Conclusions:

This hypothetical accident, and the consequences of it, pose no threat to the public health and safety or to the accumulation of occupational radiological exposure.

Even though high surface contamination levels exist at the floor area and the spent fuel pool waters are contaminated such that the total body could be exposed to relatively high radiaiton levels, area radiation monitors would alarm to indicate its presence. Personnel would be evacuated from the area to ensure that occupational exposures are limited.

7.3 Inadvertent lifting of prefilter above pool surface

Assumptions:

It is assumed that due to a failure in the crane control system, the overhead crane moves toward the loading bay after pulling one expended filter to the maximum height of eight feet below the pool

surface. As the crane moves toward the hay, the handling tool hits the end of the pool and the filter is dragged from the water exposing operating personnel.

Analysis of the accident is performed by using a point source approximation and calculating the exposure rate at a distance of 15 feet from the filter. The calculated exposure rate is 21 R/hr and is based on an estimated filter loading of 1000 curies.

Occupational Exposure Effects:
As the filter assembly mears the surface of the spent fuel pool water area, radiation monitor alarms will be sounded announcing the presence of high radiation fields. Personnel would be evacuated from the area to ensure that occupational exposures are limited.

Off-site Effects:
Airborne contamination as a result of this hypothetical accident would not occur since the particulate activity is fixed on the filter elements which are contained within the filter housing.

The increase in the radiation level at the site boundary would not be significant due to the shielding characteristics of the fuel building walls and the distance to the site boundary.

Conclusions:
The public health and safety is not compromised as a consequence of this hypothetical accident. Occupational exposure levels are ALARA.

Chapter 8
Conduct of Operations

The SDS program for operations is divided into a phase-wise approach. These phases are:

8.1 System Development

System development activites are devoted to assuring that components are developed specifically to meet the conditions imposed at TMI and perform in the intended manner.

The ion-exchange process is a well understood process. Even though ion-exchange resins have been in use for approximately 50 years or more, a development program was conducted at the Oak Ridge National Laboratory to ensure that the resins selected for use at TMI provided optimized performance characteristics. This development program was conducted to enable the evaluation of the performance characteristics of various resins using samples of the waters to be processed at TMI.

Additional development effort has been devoted to the verification that resin loading and dewatering can be accomplished in the intended manner and that the remote tools, necessary for the coupling and de-coupling of the resin beds, operates in the intended manner.

8.2 System Preoperational Testing

Prior to shipment each resin bed vessel will be hydrostatically tested in conformance with the requirements of applicable portions of the ASME Boiler and Pressure Vessel Code. Upon completion of construction the entire system will he hydrostatically tested to assure leak-free operations. The system will be tested to an internal pressure of no less than 1.5 times the design pressure. Pneumatic testing shall be conducted at an internal pressure of no less than 1.1 times the design pressure.

Individual component operability will be assured during the preoperational testing. Motor/pump rotation will be verified, control schemes will be verified, system flow paths and flow rates will be verified. The leakage collection sub-system, as well as the gas collection sub-system, will be tested to verify operability. Filters for the treatment of the collected gaseous waste will be tested prior to initial operation. System preoperational testing will be accomplished in accordance with approved procedures.

8.3 System Operations

System operations will be conducted in accordance with written and approved procedures. These procedures will be applicable to normal system operations, emergency situations, and required maintenance evolutions.

Prior to SDS operation, formal classroom instruction will be provided to systems operations personnel to ensure that adequate knowledge is gained to enable safe and efficient operation. During system operations on-going operator evaluations will be conducted to ensure continuing safe and efficient system operation.

In addition to the operating personnel, certain other individuals will be directly involved in the processing program. Additional training will be provided to other personnel as required.

8.4 System Decommissioning

The decommissioning plan for SDS is being developed. An outline of the planned aproach to decommissioning is shown below.

The basis for the decommissioning plan is that the Submerged Demineralization System is a temporary system; its installation and removal will cause no permanent plant changes.

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