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TECHNICAL EVALUATION REPORT

SUBMERGED DEMINERALIZATION SYSTEM

MARCH 1981

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Chapter 1

Summary of Treatment Plan

1.1 Project Scope

The decontamination of TMI-2 includes the processing of approximately 710,000 gallons of radioactively contaminated water contained in the reactor coolant system and the reactor building containment sump. The activity level of this water is given in Table 1.1. In addition, 300,000 gallons of water to flush the sump is estimated to be required. This water will also require processing.

This report describes the 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 to take advantage of shielding provided by the water in the pool.
2. Radioactive gas collection and treatment prior to release.
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.
5. Use of existing EPICOR-II equipment for polishing of SDS effluent.

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. The Reactor Coolant System (RCS) specific radionuclides and concentrations are based upon actual sample data taken. The RCS activity is decreasing due to radioactive decay and leakage from the RCS. The containment sump activity is based upon samples decayed to Oc. ber 1, 1980. The activity level for the containment sump shows good agreement (within 10%) with the estimates made in ORNL/TM-7448.

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-2, 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 program personnel, increased contamination levels, and increased possibility of airborne radio 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. Therefore, compliance with the principles of ALARA is not possible.
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 that leakage from the building may occur. Leakage of contaminated water from the reactor building sump may threaten the public health and safety.
4. Continued storage of the water in the containment building for an extended period of time is undesirable. The primary isotopes of concern (Cs-137 and Sr-90) exhibit decay half-lives of approximately 30 years. Storage in the containment sump for approximately 300 years would be required for 10 half-life decay. Maintenance of containment integrity for this interval of time cannot be assured.

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

Discussion:

1. To safely contain the contaminated water, the construction of an on-site liquid radwaste storage facility would be required.
2. Additional radiation areas on the plant site would be created if a liquid radwaste storage facility -e built.

3. Estimates indicate the construction of a liquid radwaste storage facility would require two to three years, at a minimum.

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 the spread of contamination.

6. The reactor building sump water contingency plan, submitted to the NRC previously, does not represent adequate long-term storage locations for the water.

7. Disposal of the water prior to natural decay is required because of the long radioactive decay half-lives. This alternative is not representative of an acceptable long-term solution.

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, transportation and disposal of this extremely large quantity of solidified waste would be prohibitivley expensive and violate basic principles of minimizing radioactive waste volumes.

4. The handling evolution required to solidify the contaminated water may involve substantial 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 be feasible.

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

Discussion:

1. The system would be capable of concentrating fission products on a medium to effectively remove those products from the water.
2. Processing contaminated water would result in concentrated waste requiring additional shielding.
3. The system incorporates remote operability features.
4. Design, construction and operation would allow for relatively short lead times.
5. The system would require minimal maintenance.
6. The SDS is amenable to location within the Spent Fuel Pool which would utilize the shielding capability of the pool water.
7. Containers of highly loaded ion exchange media arising from operation of the SDS would not be acceptable at shallow land disposal sites.

The SDS design and selection of ion exchange media allows volumes of such highly loaded media to be minimized to permit interim storage and probable ultimate disposal in a geological repository. It is believed that the EPICOR-II liners, generated as a result of polishing the SDS effluent, will be suitable for shallow land disposal because of their low curie content.

8. The EPICOR-II system, used in conjunction with SDS, will provide the capability to remove trace quantities of radionuclides from the SDS effluent.

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

Alternative V: Epicor II System only (Volume Reduction)

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 has processed 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.

5. Based on current regulatory positions and proposed regulations, it is not clear that the more highly loaded liners from EPICOR-II, as a result of direct processing of the sump waters, would be acceptable for disposal at existing commercial shallow land disposal sites. If this is so, then reduction of volume to the maximum extent possible is prudent to permit efficient interim storage and probable ultimate disposal in a geological repository.

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

1. Increased processed water storage volume, number of vessels, radioactive waste shipments, processing time and additional handling requirements resulting from EPICOR II inlet dilution, is not desired.
2. Higher than necessary personnel exposures is not consistent with the principles of 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 may require at least four years.
3. Evaporation provides the ability to process a wide range of chemical contaminants.

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) coupled with the EPICOR II system.

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) in conjunction with EPICOR II for final polishing, or Alternative IV, was chosen as the most appropriate process for the following reasons:

1. Basic design simplicity.
2. High performance for decontaminating liquids, i.e., decontamination factors up to 10^7, or higher.
3. Amenable to placement under water to take advantage of shielding properties of the water
4. Ability to implement water processing in a timely fashion for support of the overall objective of fuel removal.

5. Ability to use existing proven plant structures, equipment and technology for containment of the processed water and final process polishing (EPICOR-II)

The SDS with EPICOR II is an ion-exchange process expected to provide decontamination factors of up to 10^7 for cesium and 10^5 for strontium (see Table 3.1), thus removing the majority of the activity from the water prior to placement in the Processed Water Storage Tanks.

1.4.2 SDS Operating Description

Figure 1.1 shows a block diagram of the process flow of the Submerged Demineralizer System (SDS) with the EPICOR II System. Radioactive water enters the SDS via the RCS manifold or the ion exchange manifold which allows processing of the Reactor Coolant System (RCS) or the containment sump water. This feature is provided so that a means of SDS processing of the RCS water is available, if required. These two sources of water can pass through two cartridge-type filters for removal of particulate matter. The water is temporarily stored in four 15,000 gallon storage tanks located in the "A" SFP, which are a source of feed to the SDS.

Containment sump water is pumped from the waste storage tanks to the ion-exchange system. RCS water, if processed by SDS, bypasses the waste storage tanks and can be pumped directly to the ion-exchange system.

Sample connections are provided on the influent and effluent of the filters, and influent to the ion-exchange system to determine radionuclide content and concentrations of the water to be processed.

The first part of the SDS ion-exchange system consists of six underwater vessels (24 1/2 in. x 54 1/2 in.), each containing approximately 8 cubic feet of zeolite ion exchange media. Zeolite media volumes may be changed to reflect different processing scenarios. Inlet, outlet, and vent connections are made with remotely operated couplings. The vessels are arranged in two parallel trains with three columns in each train. Flow may be directed through one train of three vessels or through both trains in parallel. Loading of the vessels will be controlled by feed batch size, residence time, influent and effluent sample analysis, and continuous monitoring.

Present SDS operations are envisioned to provide for radionuclide loading of the zeolite media to approximately 40,000 Ci at the time of shipping. This loading level is based on restrictions imposed based on the shielding provided by the Chem-Nuclear 1-13C shipping cask. From the point of view of waste volume generation it is desirable to load the zeolites to higher levels. Presently, General Public Utilities is directing the performance of studies to provide for higher radiological loadings of the media (60,000 Ci and 120,000 Ci) to provide a lower waste volume generation. Should it be determined to be desirable to utilize higher radiological loadings in the zeolite media, we will inform the NRC and request permission to proceed to higher radiological loadings. A different shipping cask will be identified at that time.

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 in the spent fuel pool using the pool area crane. The second and third beds will be disconnected, moved to the first and second positions, respectively. A new ion exchanger vessel is then installed in the third position. Following installation of the new ion-exchanger, the treatment of the contaminated water will recommence. This operational concept, which is the currently intended mode of operation, has eliminated the potential for valving errors and also minimizes the possibility of an unexpected radionuclide "breakthrough" which could recontaminate the water already processed. This mode of operation may change if the processing senario changes.

The second part of the SDS ion exchange system consists of two parallel ion-exchange vessels located underwater and immediately downstream of the zeolite beds. These ion exchange beds will contain cationic exchange media primarily for removal of the remaining strontium isotopes. The columns are intended to be operated singly. When the SDS is processing contaminated sump water, the effluent from the ion exchangers can be sent to EPICOR-II for polishing. When processing reactor coolant the effluent is routed to installed tankage for injection back into the Reactor Coolant System as a source of makeup.

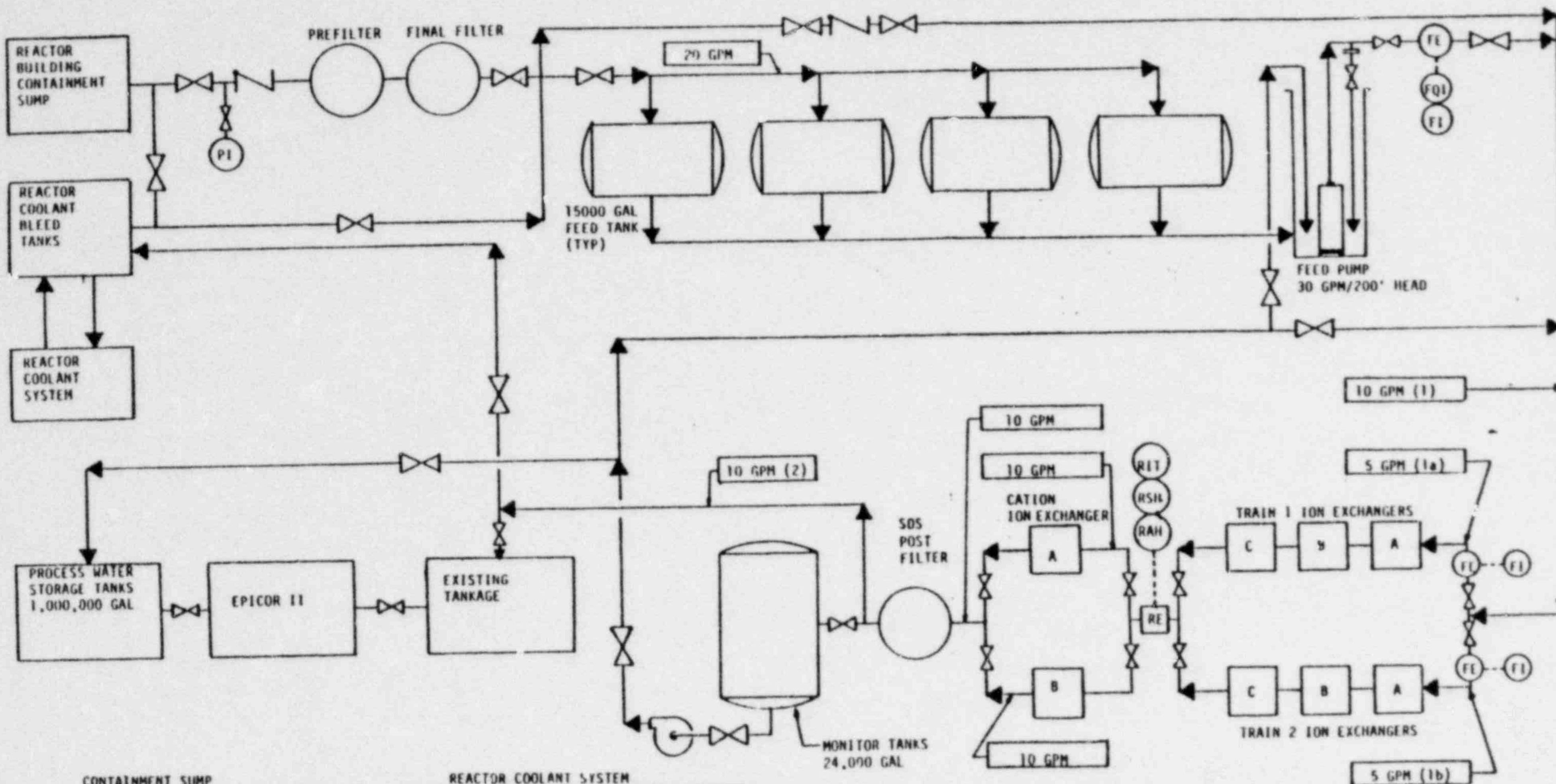
The spent ion-exchangers and filters of the SDS will be retained under water in the spent fuel pool until removed. To transport spent ion-exchangers and filters, they will be loaded into shielded casks while under water and removed from the spent fuel pool. Following decontamination of the cask surface, the cask can then be loaded onto a trailer for transportation.

TABLE 1.1

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

	Radionuclide Concentrations uCi/ml	
Isotope	Reactor Coolant System	Containment Sump
	(Sample Results February, 1981)	(Decayed to October, 1980)
H-3	0.066	0.97
Sr-89	0.25 (not analyzed 2-81)	0.18
Sr-90	23	2.64
Sb-125	1.6×10^{-3} (not analyzed 2-81)	0.0091
Cs-134	3.4	27.7
Cs-137	25	172
pH	7.6	8.6
Boron	3800 ppm	2000 ppm
Na	1240 ppm (not analyzed 2-81)	1100 ppm
Volume	88000 gallons	625000 gallons[1]

[1] The containment sump volume is increasing about 150 gallons/day due to leakage from the Reactor Coolant System.



CONTAINMENT SUMP

STREAM	1	1a	1b	2
FLOW GAL/MIN	10	5	5	10
ACTIVITY CI/GAL	0.77	0.77	0.77	4×10^{-3}

REACTOR COOLANT SYSTEM

STREAM	1	1a	1b	2
FLOW GAL/MIN	10	5	5	10
ACTIVITY CI/GAL	0.22	0.22	0.22	6×10^{-4}

FIGURE 1.1 ION EXCHANGE FLOW SHEET

Chapter 2

Summary of Health and Environmental Effects

2.1 Occupational Radiation Exposure During Routine Operation

The SDS has been designed to maintain radiation exposures to operating personnel as low as reasonably achievable. To implement the ALARA concept, the following features have been incorporated into the SDS design.

- Shielding has been designed to limit whole body dose rates in operating areas to less than 1 mRem/hr. The filters and ion-exchangers are located approximately 16 feet underwater for shielding. Components and piping carrying high activity water not contained underwater in the fuel pool have been provided with shielding.
- Controls and instrumentation are located in low radiation areas.
- Components containing high activity water have been designed for exhaust to the SDS Off Gas System. The Off-Gas System will minimize the potential for excessive airborne radioactivity releases in the work areas and to the environment.

Additional design and operational ALARA features are given in Section 6.

The occupational exposure for the EPICOR-II system was assessed in NUREG-0591. The occupational radiation exposure for the EPICOR-II system will be lower for the processing of the effluent from the SDS than previously procesed by EPICOR-II since the influent activity to the EPICOR-II from the SDS has been substantially reduced by processing the radioactive water through the SDS.

2.1.1 Exposure Planning

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

- Review of operating, maintenance and surveillance procedures to assure precautions and prerequisites are adequate.
- Review of the installed system to identify potential problems during operation and the implementation of corrective actions.
- Operational evaluations during preoperational testing and system training will be performed to update exposure estimates.
- Determination of radiation dose rates during normal operations and maintenance evolutions will be performed.

As these reviews are completed, operating and surveillance frequencies can be established; total occupational exposures can be updated for the various activities during SDS operation. This exercise will permit review of those activities estimated to yield the highest man-rem expenditure. Pre-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 and EPICOR-II

Maximum individual dose commitments based on the radionuclide concentrations presented in Table 1.1, for 50 days of operation of the the system, (the total time required to process 710,000 gallons of water at 10 gpm) are 9.47 x 10^{-3} mrem for total body exposure, 3.57 x 10^{-4} mrem for bone exposure, 4.65 x 10^{-1} mrem for thyroid exposure, and 8.82 x 10^{-3} mrem for GI tract exposure. The dose to the thyroid of a one year old child is estimated to be 3.49 x 10^{-1} mrem. The total body dose to the entire population within 50 miles is calculated to be 0.15 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 of unexpected occurrences includes the analysis of five hypothetical accidents that are postulated 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. No exposures occur to the public since the contaminated water is contained in the pool. The maximum exposure rate at a distance of six feet above the pool surface is estimated to be 116 mR/hour. Since the release of water occurs underwater, no significant exposures are expected for workers. The primary impact of the accident is the contamination of the water in the Spent Fuel Pool (233,000 gallons). (Refer to Section 7.1)

The second hypothetical accident assumes a pipe is ruptured and containment water is sprayed into the building and fuel storage pool. It is possible that workers could be contaminated, however, prompt implementation of emergency procedures would minimize radiation exposures. The maximum exposure rate three feet above an area on the floor on which the spray water resides is expected to be 8.64 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. (Refer to Section 7.2)

The third hypothetical accident evaluated considers the inadvertent raising of a loaded prefilter above the pool surface. The dose rate at a distance of 15 feet from the source is estimated to be 21 Rem/hour and could result in a dose of approximately 1.8 rem to workers who remain in the area for a five minute period. (Refer to Section 7.3)

The fourth hypothetical accident evaluated considers the inadvertant raising of a loaded zeolite ion exchanger above the pool surface. The dose rate at a distance of 20 feet from the source is estimated to be approximately 297 Rem/hr. (Refer to Section 7.4)

The final hypothetical accident considers the inadvertant drop of the SDS shipping cask containing a loaded zeolite ion exchanger. The SDS shipping cask is assumed to be dropped from the maximum height of the Fuel Handling Building crane to the EL 305′ floor. The dose rate resulting from a complete rupture of the SDS shipping cask at a distance of 20 feet is approximately 297 Rem/hr and assumes rupture of both the cask and the vessel. The small amounts of radionuclides assumed to become airborne would not result in significant exposures to the public. Also there would not be a significant effect from direct radiation exposure to the public. (Refer to Section 7.5)

The evaluation of unexpected occurrences for the EPICOR-II system was analyzed in NUREG-0591. The potential releases from processing SDS effluent water will be significantly lower because of the lower concentration of water being processed through EPICOR-II from the SDS. (See Table 3.1)

2.4 Industrial Health and Safety

2.4.1 Public Safety

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

1. Lifting and handling activities described 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. 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. Structural equipment and operating equipment used shall meet Occupational Safety and Health Administration requirements as applicable. 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 stored and held until final disposal is accomplished.

2.6 Ultimate Waste Disposition

There are several open issues surrounding ultimate disposition of radioactive wastes arising from the operation of the SDS. GPU has been informed by the NRC (Ahern to Diekamp, January 12, 1981) that: "Certain types of waste generated by TMI-2 cleanup operations will contain types and amounts of radioactivity that are significantly greater than normal reactor low level wastes. Such wastes will have to be put into an interim form which can be safely stored until subsequent steps can be developed." Discussions are underway with the Department of Energy with regard to offsite interium storage and ultimate disposition of this material. Until final decisions are made regarding off-site interim storage and ultimate disposition, storage on site can be conducted safely. It is believed that the lower loaded liners arising from the use of EPICOR II as a polishing system are acceptable for shallow land disposal.

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 to reduce quantities of radionuclides to levels that are in compliance with 10 CFR 20 and 10 CFR 50.

Furthermore, experiments conducted at ORNL, documented in ORNL report TM-7448, provide evidence that SDS processing, followed by EPICOR-II polishing, should provide an effective method for water decontamination.

The initial processing of the waste water is filtration for the removal of solids to optimize the subsequent ion-exchange process. Filtration is believed to be necessary to protect the zeolite beds from particulates in the sump water.

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 much of the strontium. Other trace levels of radionuclides are also partially removed by the zeolite media. The radioactivity content in the effluent stream of each 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 the strontium - specific cationic exchange media contained in the next ion-exchange column.

Final demineralization of the contaminated sump water is intended to be processed by EPICOR-II equipment. Essentially, all remaining dissolved radionuclides are expected to be 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 for radionuclide removal.

The following equation is a qualitative expression for the removal of a single ionic specie from solution.

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

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 ion-exchange bed volumes)

K = Unit conversion constant

Important variables which are considered as part of the evaluation of ion-exchangers for decontamination are ion exchange media 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, flow distribution, pH, and temperatures are specified for the ion-exchange 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 and exchange to occur.

(2) The cross-sectional area of the ion-exchange media 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 media distribution to reduce the potential for channeling.

(5) The ion-exchange media 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 ion-exchange media impurities to maximize removal of specific nuclides.

3.3 Ion-Exchange Materials

The ion-exchanger media selected for use in this processing system are an inorganic zeolite material that is commercially available and known as Ion Siv IE-95 (formerly AW-500), cationic ion exchange media, and cation and anion resins to be used in EPICOR II.

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" to sorb small molecules, but to exclude molecules that are larger than the openings in the crystal framework.

Other media are also being evaluated. Should our plans change with regard to ion exchange media to be employed, we will notify the NRC.

Organic ion exchange resins are typically gels and are classified as cross-linked 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^+_4, 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 cross-linking 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.

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 adsorbed 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 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, radiolytic and physical stability

(8) Previous demonstrated performance (EPICOR-II)

Experimental studies with reactor coolant water have been conducted to support and verify the selection of these ion-exchangers; refer to ORNL TM-7448. 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 ion exchange media to be used in the cleanup process were the available cation-anion resins in combination with the zeolite exchanger.

Furthermore, as a result of processing approximately 500,000 gallons of radioactively contaminated water in the Auxiliary Building, we are confident that the SDS, with EPICOR-II used as a polishing system for treatment of SDS effluent, can provide an effective means to decontaminate the highly contaminated waters. EPICOR-II resin loadings may be altered to improve polishing effectiveness, if required.

3.5 Predicted Performance of Ion-Exchangers

The concentrations of radionuclides in samples of water from the containment building sump have been measured. Those radionuclides still detectable in September, 1979 included the isotopes Sr-89 and Sr-90, Cs-134 and Cs-137, and Sb-125 and the short-lived Nb-95, Ru-103, La-140, and I-131. By October 1980, the only remaining significant isotopes are Sr-89, Sr-90, Sb-125, I-129, Cs-134 and Cs-137. The expected performance of the SDS ion-exchangers, and the EPICOR-II ion exchangers is shown in Table 3.1. The concentrations of strontium and cesium are expected to be significantly reduced by processing through the SDS and EPICOR-II system.

Antimony is expected to pass through the SDS ion exchangers and will end up as the predominant gamma emitter in the solution entering the EPICOR-II system. The concentration of Sb-125 in the containment building sump sample is approximately 0.0091 microcuries per milliliter.

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 loaded beds. The detectors sampling the cation influent can provide gross activity indication to provide the necessary protection for the cation beds.

TABLE 3.1
Expected activity concentrations[a] in SDS process streams after 200 bed volumes through each zeolite bed
(Based on continuous flow three zeolite columns)

Nuclide	Effluent concentrations,[a] uCi/ml.						Effluent
			Zeolite columns			Cation	
	Feed	Filter	First	Second	Third	column	EPICOR-II[d]
^{3}H	1.0	1.0	1.0	1.0	1.0	1.0	1.0
^{60}Co	b	b	6E-5	6E-5	6E-5	6E-5	3E-6
^{89}Sr	5.3E-1[c]	5.2E-1[c]	6.6E-3	6E-4	5.9E-4	5.2E-4	2E-6
^{90}Sr	2.29[c]	2.26[c]	3.2E-2	2.8E-3	2.7E-3	2.4E-3	9E-6
^{95}Nb	b	b	1.9E-5	1.0E-5	1.0E-5	1.0E-5	8E-8
^{103}Ru	b	b	2.9E-5	2.4E-5	2.4E-5	2.4E-5	<1E-6
^{106}Ru	b	b	2.4E-3	2.0E-3	2.0E-3	2.0E-3	<1E-6
^{125}Sb	b	b	1.9E-2	1.9E-2	1.9E-2	1.9E-2	6E-6
^{134}Cs	2.62E+1	2.62E+1	2.4E-3	5.3E-4	5.0E-4	5.0E-4	3E-6
^{137}Cs	1.56E+2	1.56E+2	1.4E-2	3.3E-3	3.1E-3	3.1E-3	3E-5
^{144}Ce	b	b	4.7E-4	4.7E-4	4.7E-4	4.7E-4	<1E-6

[a] In uCi/ml as of July 1, 1980 based on ORNL/TM - 7448

[b] Not detected.

[c] Differences in strontium concentrations between feed and filter effluent based on estimate of 500 gal of solids in 700,000 gal of water.

[d] Based on EPICOR-II System performance

Chapter 4

Submerged Demineralizer System Design Basis

4.1 Introduction

The Submerged Demineralization System (SDS) is an underwater ion-exchange system which has been specifically designed to process higher-level waste waters*, with inherent system features for reduction of 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. In conjunction with the SDS, the EPICOR-II system is used to provide final polishing of the SDS effluent water for removal of trace quantities of radionuclides.

Design features for the SDS include:

1. A prefilter and final filter in series, two parallel trains of 3 zeolite ion-exchangers in series, and two cationic ion-exchangers in parallel followed by the EPICOR-II equipment 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 for the recovery of TMI-2.

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

b. A system that has the capability to reduce the fission product concentration in the contaminated water and has optional capabilities for removing chemical contaminants to permit future disposition of the concentrated waste form.

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.

e. A system that incorporates known and demonstrated processing equipment, materials and techniques. (EPICOR-II)

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 5.6, 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 8 cubic-feet (nominal) of zeolite exchange media;
4. Two parallel ion exchange beds containing cationic ion exchange media primarily for strontium 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, sampling, and monitoring systems;
7. Two monitoring tanks for collecting treated water.

8. An off-gas system for treating and filtering gases and vent air from the system;
9. Associated piping, valving, and structural supports required for placement of system components;
10. Auxiliary systems including underwater ion-exchange column storage, a dewatering system, and analytical equipment;
11. Vent system to allow for venting of stored vessels.

The EPICOR-II system is downstream of the SDS process flow stream for removal of trace fission products that are not removed in the ion exchange media of the SDS.

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. Nuclear Regulatory Guide 1.21 Revision 1, June 1974
- Code of Federal Regulations, 10 CFR 20, Standard for Protection Against Radiation
- Code of Federal Regulations, 10 CFR 50, Licensing of Production and Utilization Facilities.

4.3.2 Process

The design shall provide for operations 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.

4.3.3 Performance

The isotopic inventory for the water to be processed is summarized in Table 1.1. The SDS followed by the EPICOR-II systems is designed and operated such as to reduce the average isotopic specific activity of the treated waste streams. The expected performance of these systems is given in Table 3.1.

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).

4.3.5 Performance and Design Requirements

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

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

4.3.6 <u>Piping System</u> (piping, valves and pumps)

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 and Table 1 of Regulatory Guide 1.143.
2. Piping system design shall be based on a maximum of 150 psi at 100°F.
3. Piping runs are designed to permit water flushing.
4. Instrument connections to piping systems are located to provide clearance for attachment, operation and maintenance.

4.3.7 <u>Vessels and Tanks</u>

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 eight (8) cubic feet of zeolite ion exchange media for the purpose of removing cesium and strontium from the waste water. Should our processing scenario be changed it may be necessary to alter the volume of the zeolite media. Should changes occur, we will inform the NRC.
   b. Cation ion-exchangers shall contain cationic ion exchange media to remove residual strontium.
   c. Influent filter units are planned to contain cartridge type filter assemblies or equivalent mechanisms capable of

removing particles greater than approximately 10 microns. SDS effluent filter capability has been provided to incorporate the capability to filter out ion-exchange media fines from the process stream should fine carryover occur.

3. The SDS ion-exchangers and filters shall be capable of functioning submerged under approximately 16 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 loaded ion-exchangers shall be limited to prevent channeling or breakthrough.

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

6. The ion-exchangers shall be equipped with a lifting arrangement compatible with the spent fuel pool crane to permit movement of the vessels in the pool.

7. The 10-cubic-foot vessels will be equipped with all required nozzles, including inlet, outlet, vent connections, and fill and sluicing connections.

8. Each ion-exchanger shall be equipped with all internals required for media distribution, dewatering, and venting.

9. <u>Design Conditions</u>

   a. The 10-cubic-foot vessels will be compatible with the piping design conditions of 150 psig at 100°F. The vessel design conditions for continuous operation will be, at least, equivalent to the piping design conditions.

b. The following additional design conditions have been imposed:

- o Overall Height — 54 1/2 inches
- o Overall Diameter — 24 1/2 inches
- o Materials — Stainless Steel
- o Weight — will have negative buoyancy (loaded with ion-exchange media)

10. Testing

The vessels shall be hydrostatically tested at 1.5 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 1mR/hr, general area. The shielding for the EPICOR-II equipment is adequate for the processing of the SDS effluent because the SDS effluent water activity will be lower than the activity level of the water for which EPICOR-II shielding was originally designed.

4.3.9 Leakage

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

4.3.10 Building and Auxiliary Service Interfaces

The SDS has been designed to meet the following building interface requirements.

1. All components of the SDS located in the Fuel Handling Building do not exceed the normal load capacities of the cranes in this area. The Fuel Handling Building auxiliary and main cranes have capacities of 110 tons and 15 tons, respectively.
2. The SDS will operate in the ambient conditions of the Fuel Handling Building as supplied by the building heating, ventilating and air conditioning system, and lighting system.
3. During installation of the system, no equipment will be permanently attached to the fuel pool liner and no penetrations will be made in the fuel pool liner.
4. Structural supports for the system will be designed to take the dynamic and static loads associated with the normal operation of the system.

4.3.11 Controls and Instrumentation

4.3.11.1 General System Description

1. The control and instrumentation systems shall be designed to control and monitor the various normal process functions throughout the system and will permit a safe, orderly shutdown of the system.
2. The controls and instrumentation systems will enable the operators to perform the designated functions efficiently and safely.

3. Where portions of the process must be operated remotely, sufficient instrumentation shall be included to assure safe operation and permit analysis of a process upset or remote detection of equipment malfunction.

4. Control and instrumentation systems shall be categorized as: (1) controls and instrumentation systems essential for the maintenance of process fluid confinement, and (2) process controls instrumentation systems essential for the determination of process operating parameters.

5. Radiation monitoring and surveillance instrumentation essential for the protection of operating personnel, the public and the environment will be provided.

4.3.11.2 Performance and Design Requirements

1. Remote controls and instrumentation shall have provisions for remote connection of electrical leads.

2. Alarms and/or indicators are provided for adequate surveillance of process operation.

3. Process-connected instrumentation shall be constructed of material compatible with that used for the construction of the process equipment.

4. Electrical wiring shall be designed in such a manner as to minimize noise and spurious signals.

5. Instrumentation identification and numbering should follow the standards and practices of the Instrument Society of America (ISA).

6. Radiation monitors shall be provided for the detection of gamma radiation. In-line radiation monitors will also monitor beta radiation.

7. Specific instruments shall be designated to function in a fail-safe mode and will alert to a failure condition.

4.4 System Operational Concepts

The following is a summary operation description. This operating sequence depicts the processing scenario as currently planned and could be changed based on operating experiences.

The SDS process logic as currently planned, is based on the following steps:

1. Ion-exchanger units will be preloaded with new ion exchange media prior to placement in the system. The primary treatment beds will utilize zeolite media. The cation exchanger units will use cationic ion exchange media.
2. Water will be introduced to fill and vent the ion-exchange unit.
3. These preloaded SDS ion-exchange units will be lowered into the Unit 2 spent fuel pool and placed in the containment enclosures.
4. Inlet and outlet header connections will be made to the ion-exchange unit.
5. The ion-exchange 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 ion-exchange 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 first ion-exchange bed becomes depleted, the unit will be flushed 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 ion-exchange unit will be decoupled remotely via the use of quick disconnects and will be stored in the spent fuel pool. However, loading directly into a cask prior to shipment is possible.
9. After the first ion-exchange unit has been removed, the second ion exchange unit will be placed into the position of the first unit and the third ion exchange unit will be moved to the second position. A new ion-exchange unit will be installed in the third position.

Chapter 5

System Description and Arrangement

5.1 Demineralizer System

5.1.1 Influent Water Filtration

A flow diagram of the waste water influent system is shown in Fig. 5.1. Contaminated water is pumped into the SDS from either the containment sump or the RCS. The containment sump will employ either a pump floating on the containment sump water surface or the use of the presently installed WG-P-1 pump.

Two filters have been installed to filter out solids in the untreated contaminated water before the water is processed by the ion-exchangers. Both filters are cartridge type whose filter elements are protected by 3/16 inch perforated metal plate as a roughing screen and 125 micron filter cartridges to remove debris and suspended solids from the contaminated water. The design of the final filter is similiar to the prefilter except that the filter cartridge is designed for removal of suspended solids of greater than 10 microns in size from the contaminated water. The flow capacity through each filter is 50 gpm. Reverse flow through filters is prevented by a check valve in the supply line to each filter.

Each filter is housed in a containment enclosure to enable leakage detection and confinement of potential leakage. The filters are submerged in the spent fuel pool for shielding considerations. Contaminated water can be 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 disconnect valved couplings which are remotely operated from the top of the pool. Inlet-outlet pressure gauges are provided to monitor and control solids loading. Load limits for the filters are based on filter differential pressure, filter influent and effluent sampling, and/or the surface dose limit for the filter vessel. 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 interconnected by piping with no valves and therefore utilized as one 60,000 gallon tank. 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 Ion Exchanger Units

A flow diagram of the ion exchange manifold and primary ion-exchange columns is shown in Fig. 5.2. This system consists of six underwater columns (24 1/2 in. x 54 1/2 in.), each containing eight cubic feet of Ion Siv IE-95 zeolite media and two underwater columns containing cationic ion exchange media. The six zeolite beds are divided into two trains each containing three 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 cationic ion exchange bed primarily for removal of strontium radionuclides. An in-line radiation monitor measures the activity level of the water exiting the cation exchanger. The valve manifold for controlling the operation of the primary ion exchange columns is located above the pool, inside a shielded enclosure that contains a built-in sump to collect leakage that might occur. Any such leakage is routed back to the feed tank standpipe. A line connects to the inlet of each primary exchanger to provide water for flushing the exchangers when they are loaded. Radionuclide loading of ion exchange vessels is determined by analyzing the influent and effluent from each exchanger. Process water flow is measured by instruments placed in the line to each ion-exchange train.

When processing containment sump water, effluent from the SDS is directed to the EPICOR-II polishing unit, if desired. When the SDS is being utilized to process reactor coolant, the effluent can be valved into the RCS clean-up manifold then back into the Reactor Coolant System via installed tankage, bypassing EPICOR-II.

5.1.4 Leakage Detection and Processing

Each submerged vessel is located inside a secondary containment box that contains spent fuel pool water. During operation the secondary containment lid is closed. This lid is slotted to permit a calculated quantity of pool water to flow past the vessels and connectors. Pool water from the containment boxes is continuously monitored to detect leakage and is circulated by a pump through one of the two leakage containment ion-exchangers (See Figure 5.2). 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 EPICOR-II

EPICOR-II (Figure 5.3) can provide final treatment of water after the water is processed through the SDS cation exchanger. When processing containment sump water, the processing plan is to polish with EPICOR-II. EPICOR-II consists of filters, ion-exchangers and receiver tanks. The purpose of EPICOR-II is to remove trace fission products that may be present in the water. The EPICOR-II safety assessment is provided in NUREG-0591.

5.1.6 Monitoring Tank System

Effluent from the cation ion-exchanger can flow into one of two monitoring tanks (Figure 5.4). The purpose of the monitoring tank system is to collect treated water. 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 can be transferred back for reprocessing by SDS or used as flush water in the SDS, or directed to existing tankage.

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 Figure 5.5. Gaseous effluent lines from the feed tanks, ion exchange vessels, sampling glove boxes and shielded valving manifolds are connected to the off-gas system. Gaseous effluent is passed through a mist eliminator in the off-gas separator tank before being treated by an electric off-gas heater to reduce the off-gas relative humidity to 70%. A roughing filter and two HEPA 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 prior to the gas entering the off-gas treatment system. The tank is located in the surge pit and is covered with a concrete and lead shield. The level in the tank will be indicated and controlled automatically with level control instrumentation that activates a pump to return collected water to the feed tank standpipe for reprocessing.

5.2 Sampling and Process Radiation Monitoring System

The sampling glove boxes are shielded enclosures which allow water samples to be taken for analysis of radionuclides and other contaminants. The piping entering the glove boxes 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.

5.2.1 Sampling System

Sampling of the SDS process to monitor performance is accomplished from three shielded sampling glove boxes. One glove box is for sampling the filtration system, the second is for sampling the feed for the first zeolite bed and the third for sampling the effluents of the zeolites and the cation bed.

The entire sampling sequence is performed in shielded glove boxes to minimize the possibility of inadvertent leakage and spread of contamination during routine operation.

5.2.2 Proc' s Radiaiton Monitoring System

The SDS is equipped with a process radiation monitoring system which provides indication of the radioactivity concentration in the process flow stream at the effluent point from each ion echanger vessel. The purpose of this monitoring system is to provide indication and alarm of radionuclide breakthrough of the ion exchange media.

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 as specified in the Process Control Program, the system will be flushed with low-activity processed water. This procedure flushes away waterborne radioactivity, thus minimizing the potential for loss of contaminants into the pool water while decoupling vessels. Vessel decoupling is accomplished remotely. Vessels are transferred using the existing fuel handling crane utilizing a yoke attached to a long shaft. The purpose of this yoke-arm assembly is to prevent inadvertent lifting of the ion exchange bed or filter vessel 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 lifting a loaded vessel out of the water shielding and preclude the possibility of accidental exposure of operating personnel.

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. The loaded vessel will then be stored until transfer to a shielded 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 shielded cask. The loaded 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 5.6 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 fuel handling building, in the "A" spent fuel pool and are covered with concrete slabs. The filters, zeolite ion exchanger vessels, and the cationic ion-exchanger vessels are located underwater in containment enclosures in the "B" spent fuel pool. These enclosures and the exchangers are supported along one side of the pool on a structural steel rack that is attached to the pool curb. The racks act 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 curb near the surge tank area.

A dewatering station is located in the "B" SFP cask pit below the water level and is used for displacing the water from expended columns and filters and dewatering them prior to placement in the cask. An underwater storage rack, designed to handle 60 expended vessels is located in the pool. This storage capacity allows processing to continue without interruption due to handling operations or vessel diposal or shipping. Stored vessels will be vented via a common header connecting to the liquid separation module to continually vent gas byproducts that may be generated in the vessels during storage.

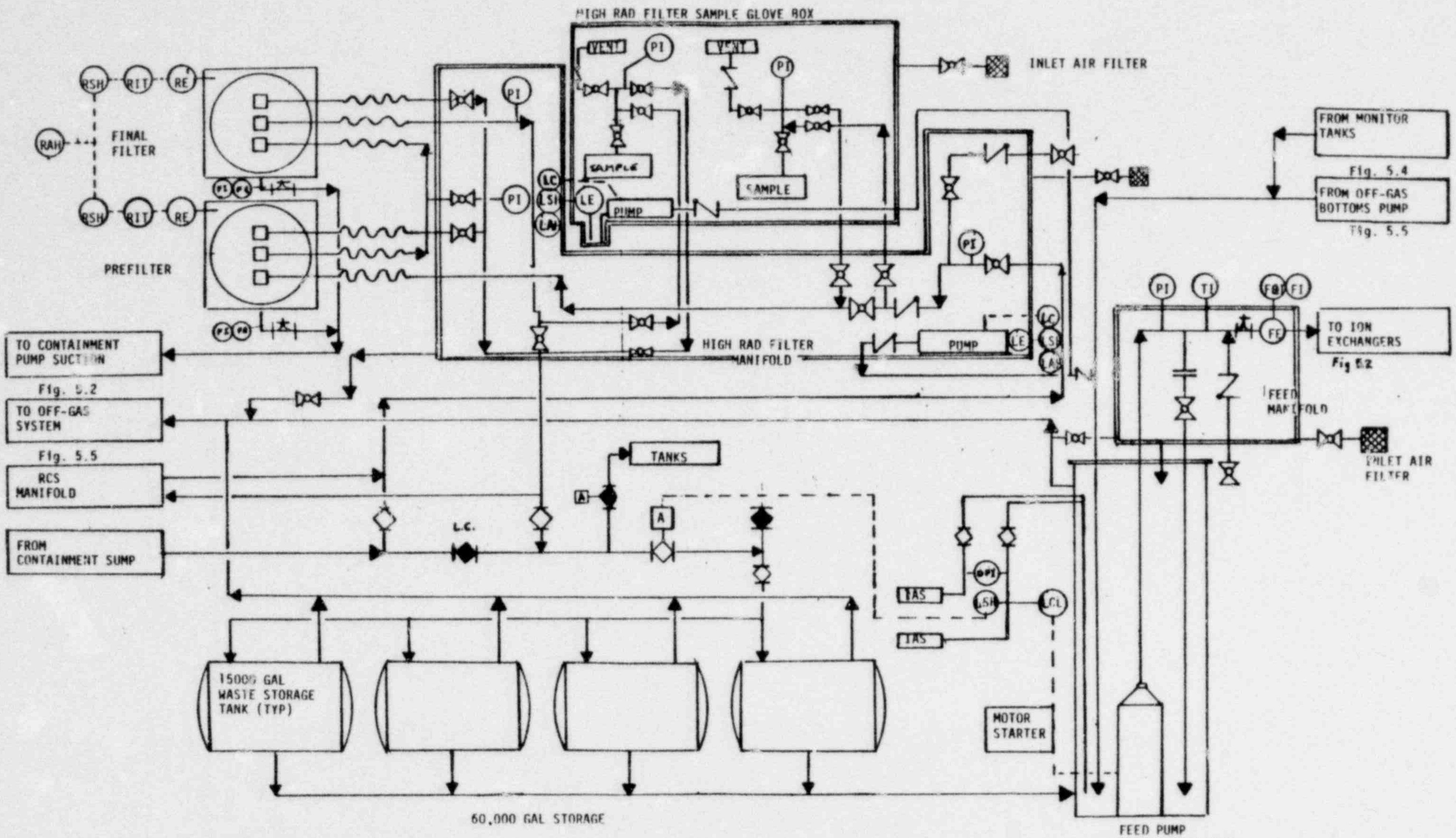


FIGURE 5.1
WASTE WATER INFLUENT SYSTEM

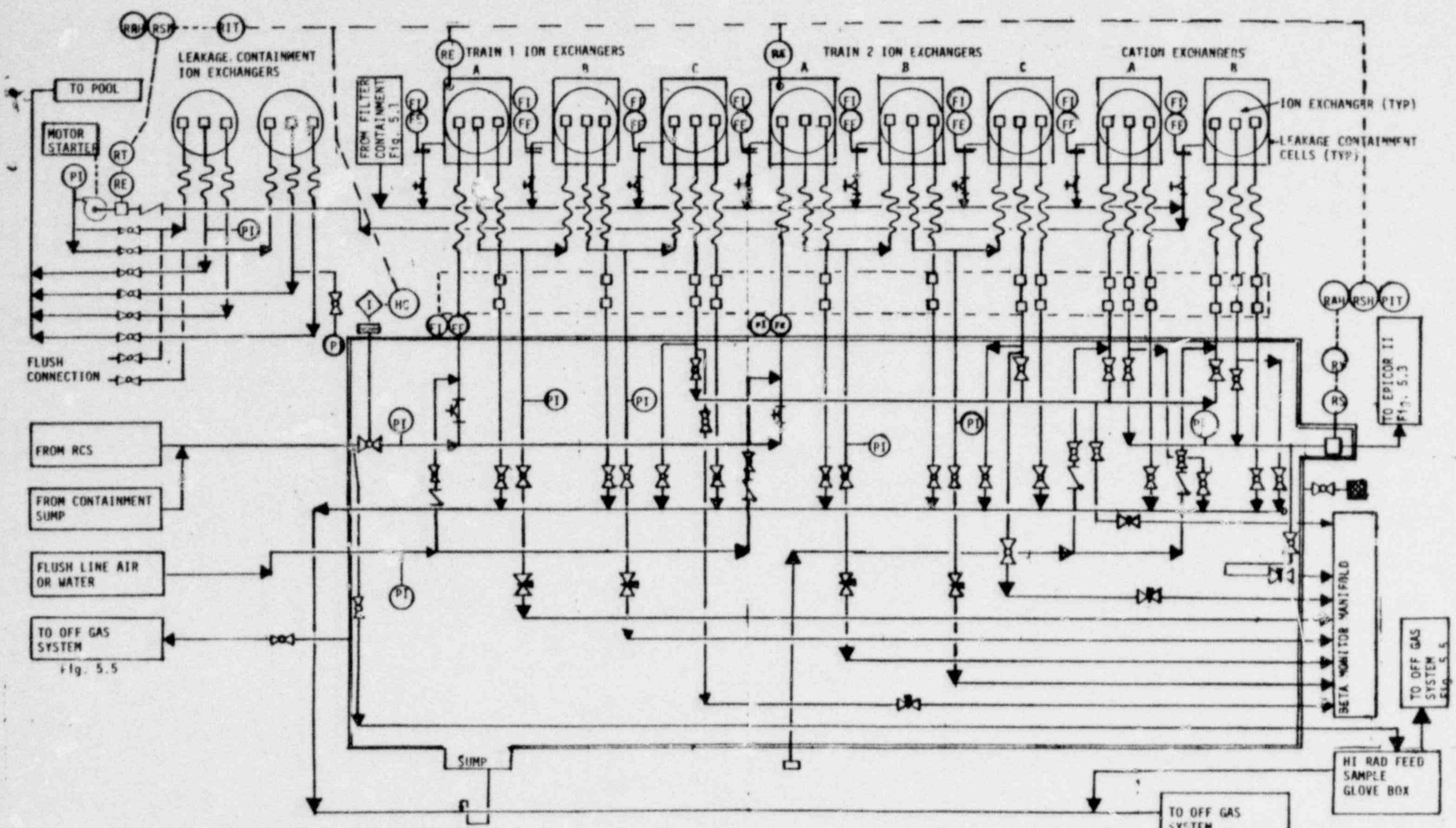


FIGURE 5.2
ION EXCHANGER TRAINS.

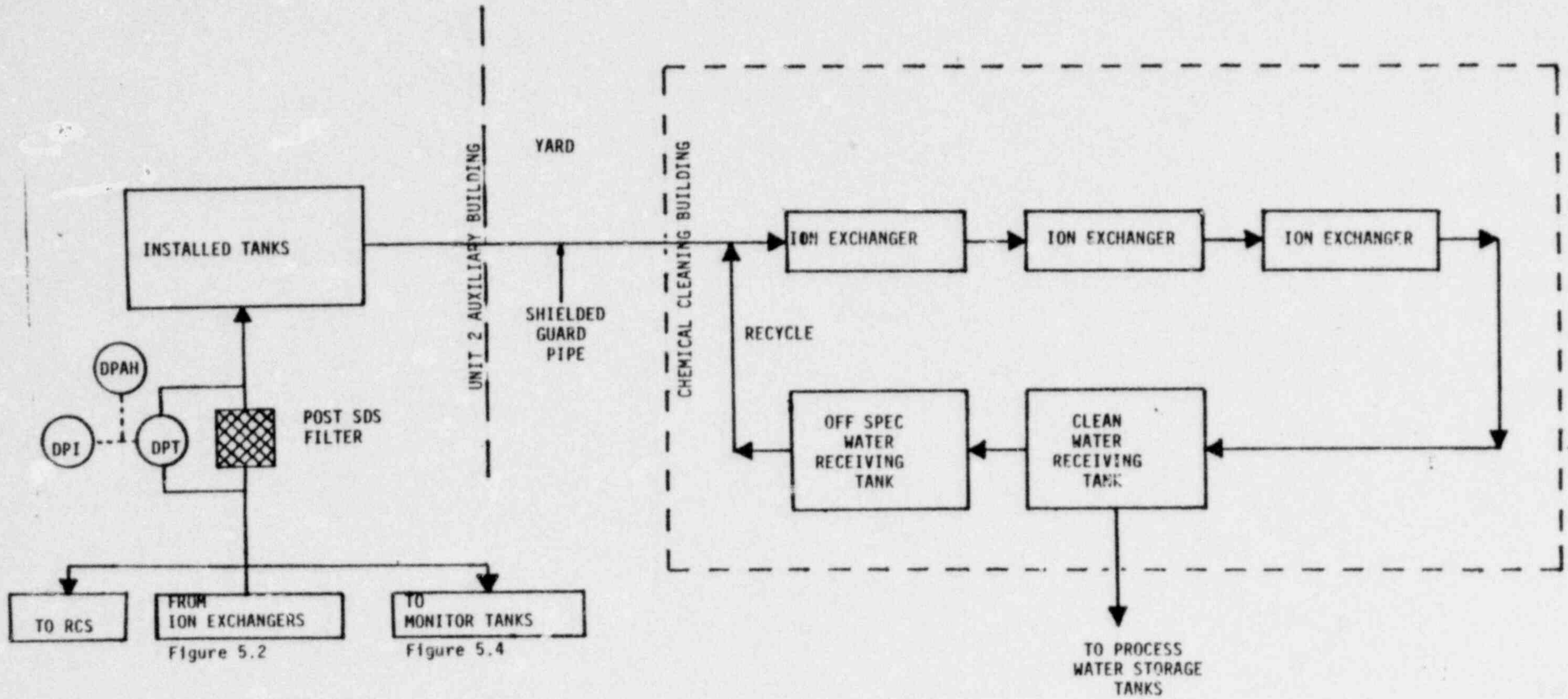


Figure 5.3 Liquid Flow Path of EPICOR II Processing System

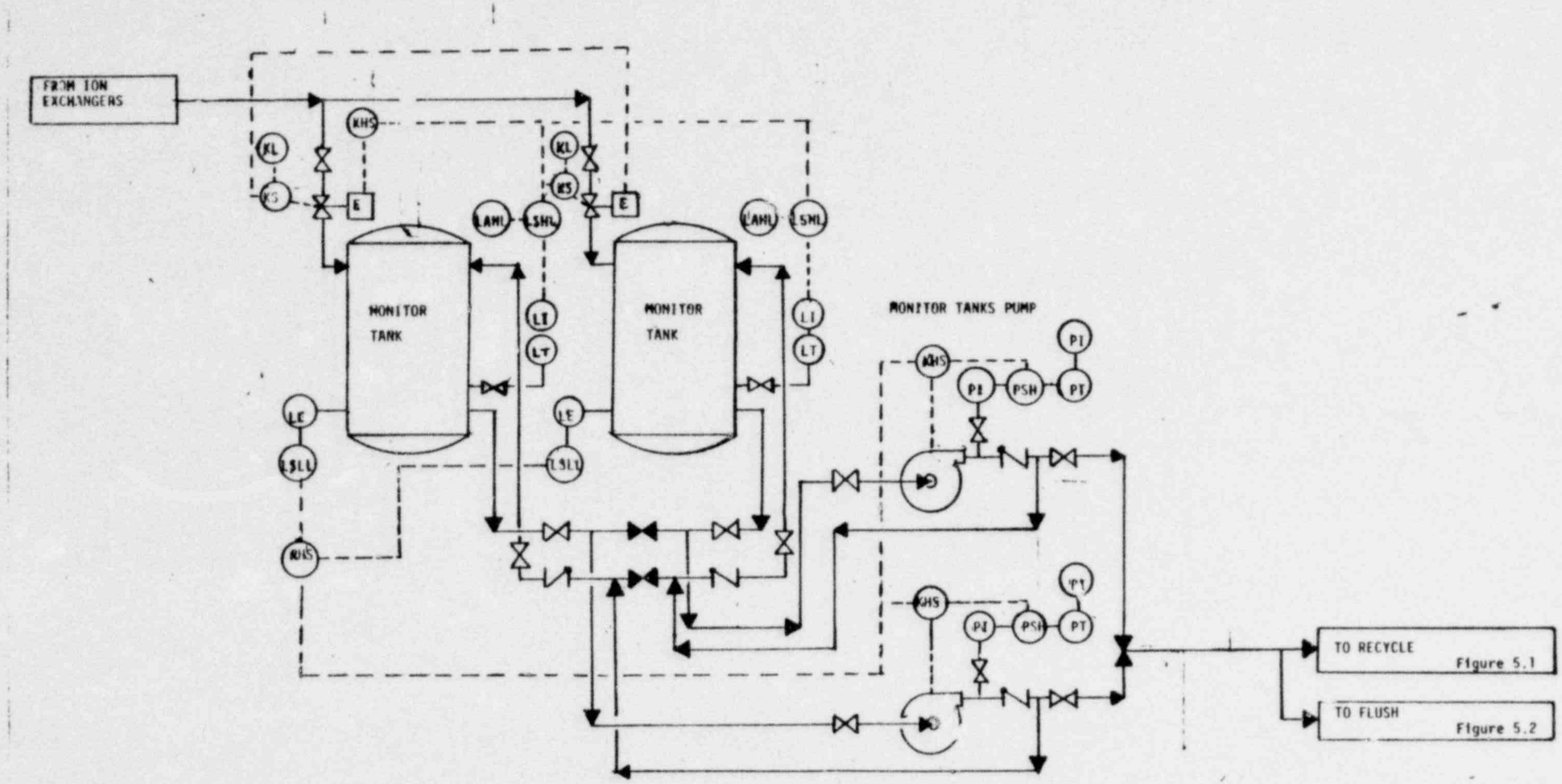


FIGURE 5.4
MONITOR TANKS

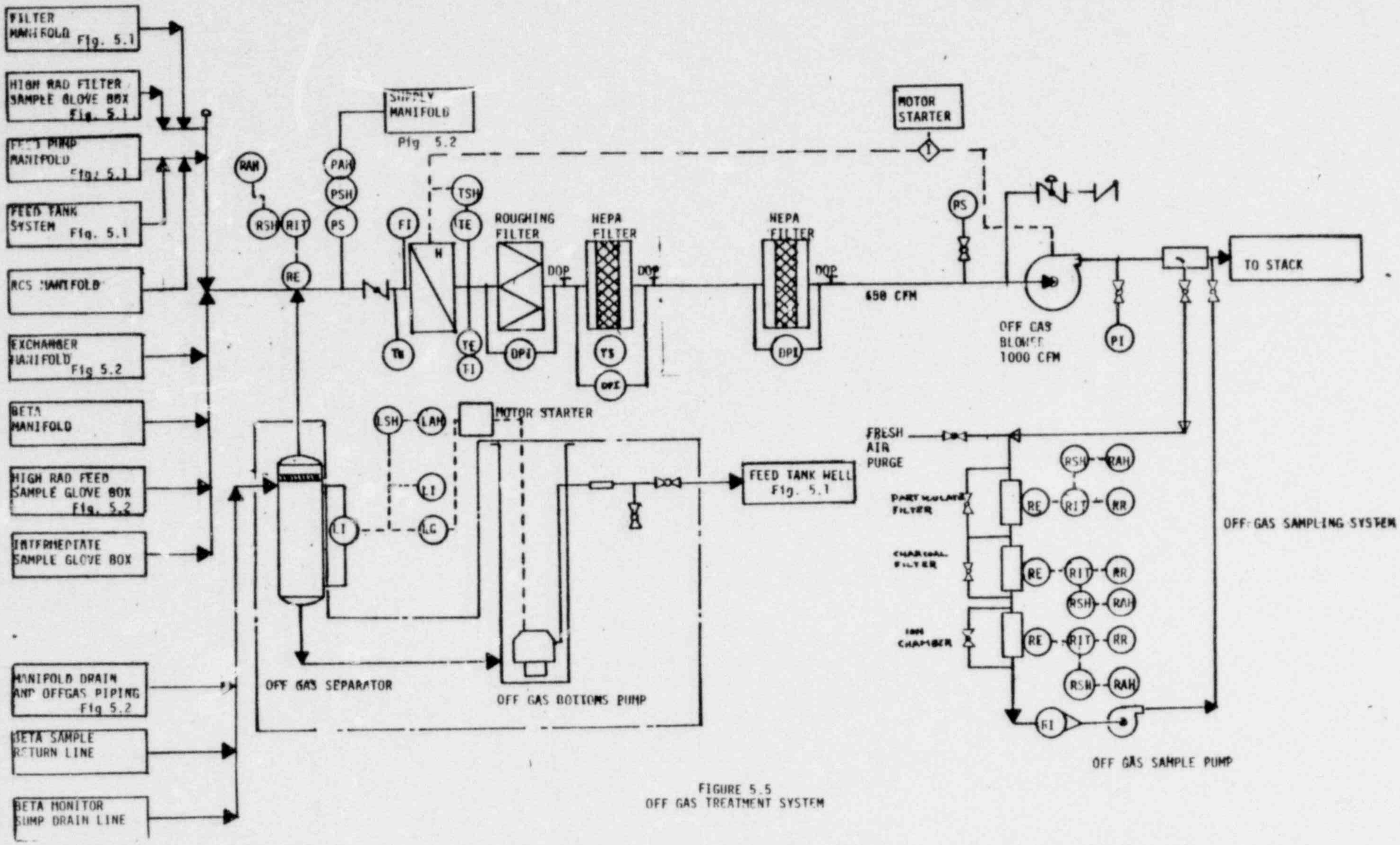
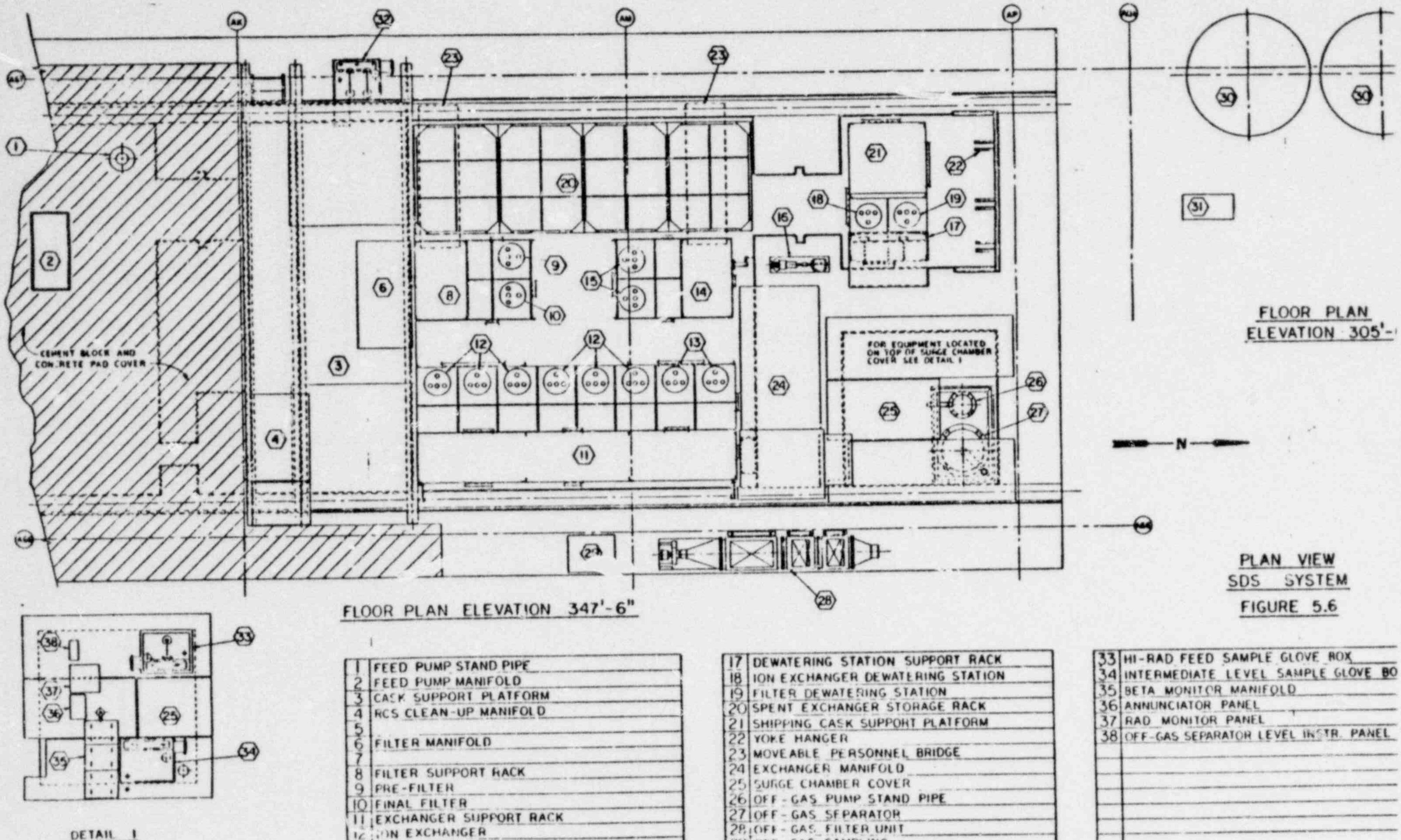


FIGURE 5.5
OFF GAS TREATMENT SYSTEM



PLAN VIEW
SDS SYSTEM
FIGURE 5.6

1	FEED PUMP STAND PIPE
2	FEED PUMP MANIFOLD
3	CASK SUPPORT PLATFORM
4	RCS CLEAN-UP MANIFOLD
5	
6	FILTER MANIFOLD
7	
8	FILTER SUPPORT RACK
9	PRE-FILTER
10	FINAL FILTER
11	EXCHANGER SUPPORT RACK
12	ION EXCHANGER
13	CATION EXCHANGER
14	LEAKAGE CONTAINMENT SUPPORT RACK
15	LEAKAGE CONTAINMENT ION EXCHANGER
16	LEAKAGE CONTAINMENT PUMP
17	DEWATERING STATION SUPPORT RACK
18	ION EXCHANGER DEWATERING STATION
19	FILTER DEWATERING STATION
20	SPENT EXCHANGER STORAGE RACK
21	SHIPPING CASK SUPPORT PLATFORM
22	YOKE HANGER
23	MOVEABLE PERSONNEL BRIDGE
24	EXCHANGER MANIFOLD
25	SURGE CHAMBER COVER
26	OFF-GAS PUMP STAND PIPE
27	OFF-GAS SEPARATOR
28	OFF-GAS FILTER UNIT
29	OFF-GAS SAMPLING
30	MONITORING TANK
31	MONITORING TANK PUMP
32	HI-RAD FILTER GLOVE BOX
33	HI-RAD FEED SAMPLE GLOVE BOX
34	INTERMEDIATE LEVEL SAMPLE GLOVE BO
35	BETA MONITOR MANIFOLD
36	ANNUNCIATOR PANEL
37	RAD MONITOR PANEL
38	OFF-GAS SEPARATOR LEVEL INSTR. PANEL

Chapter 6

Radiation Protection

6.1 Ensuring Occupational Radiation Exposures are ALARA

6.1.1 Policy Considerations

The objectives with respect to SDS operations are to ensure 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 proper radiation protection practices, including work task planning for the proper use of the appropriate equipment by qualified personnel.
3. 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 be provided with additional shielding or are submerged in the spent fuel pool. Shielding has been designed to limit whole body 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 through lead and steel shield boxes.
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. All sampling is performed in shielded glove boxes to minimize the possibility of inadvertent leakage and spread of contamination during routine operation.

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 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 occurrences, such as spills.

1. The SDS primary demineralization units are housed under approximately 16 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 Buildings which are Seismic Category I structures 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. The effluent processed water will be stored on the TMI site until final disposition has been determined.
4. The off-gas system effluent will be filtered and monitored before input to existing ventilation exhaust systems.
5. Filters, primary ion-exchange beds, cation beds, and their associated couplings are operated 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) containment enclosures (100 GPM total) is then processed through a separate ion exchange column and then discharged back to the spent fuel pool.
6. Loaded vessels will be placed in a shielded cask underwater.
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. The entire process stream is designed with appropriate pressure indicators.

11. Inlet, outlet and vent connection are made with remote operated-valved 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 or adequate shielding is provided to meet radiological exposure design limits.

All movements of the vessels 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 airborne radiological releases to the environment. 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, ion exchange and filter vessels, sample glove boxes, piping manifolds, and the dewatering station.
2. Roughing filter with differential pressure indication.
3. Two HEPA filters with differential pressure indication.
4. A centrifugal off-gas blower with flow indication.
5. Sample ports for monitoring the system and DOP test ports for HEPA testing.

6. 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 filters prior to discharge from the plant.

6.2.4 Area Radiation Monitoring Instrumentation

General area radiation monitors have been provided which will 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. Sampling operations

C. System start-up valve alignment

D. Spent vessel changeout

E. Cask removal, decontamination and survey operations

F. System maintenance

G. Vessel dewatering

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.

Source Terms for Gaseous Effluents

The plant vent system is the first off-gas stream carrying airborne radioactive material and the first potential pathway for gaseous release. The second pathway is the HVAC system vent in the Chemical Cleaning Building where the EPICOR-II system is located. 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 (in uCi/s released to the atomosphere) were developed by assuming the system operated on the principle of evaporation. For this reason, an entrainment factor of 10^{-6} is assumed for the particulate radionuclides escaping from the liquid to the vapor. An entrainment factor of 7.5×10^{-3} is assumed for ^{129}I (NUREG-0017). 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, these entrainment factors are considered to be conservative for the solution-vent system during pump transfer of water.

The release of tritium from th plant vents is calculated by assuming the air discharged from the vent was saturated with water vapor at 80°F. At this temperature, 650 cfm of air would carry 500 gm of water vapor and correlates to 2.66 x 10^{-5} uCi/cm^3 of ^3H. The release of tritium from the Chemical Cleaning Building HVAC vent is calculated from the evaporation of water at 100°F from the tank in the Chemical Cleaning Building. At this temperature the tritium concentration in the discharge of the HVAC system is 7.15 x 10^{-9} uCi/cc at 8000 cfm.

It should be noted that there are several vent systems which comprise the final off-gas stream in the Fuel Handling Building, 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 tank vents in the EPICOR-II system are the primary release point for airborne radioactive material from the Chemical Cleaning Building.

A decontamination factor (DF) of 100 is assumed for particulates for the SDS Off-gas treatment system. No effluent treatment (i.e., a DF of 1) is assumed for ^3H. The off-gas flow rate in the SDS Off-gas system is 650 ft^3/min.

Radionuclides in the off-gas of the SDS are further diluted as they are mixed with existing gaseous effluent at TMI-2, giving a total off-gas volume flow rate of 100,650 CFM (plant vent stack). It is further assumed that particulates pass through HEPA filters in place at TMI-2 to give an additional DF of 100. However, no further effluent treatment is assumed for either ^{3}H or ^{129}I.

Therefore, the total DF for particulates including both the SDS Off-Gas system and treatment previously existing at TMI-2 is 10^4. For ^{3}H and ^{129}I the DF is 1.

In the Chemical Cleaning Building, the EPICOR-II tanks and the building HVAC System are equipped with HEPA lters and charcoal adsorbers. Therefore the total DF for particulates is assumed to be 10^4. For ^{129}I a DF of 20 is assumed and for ^{3}H the DF is 1.

Table 6.1 lists the concentration of the containment sump water and influent water to the EPICOR-II system from the SDS. These data are based on the measured values given in Chapter 1 of this report and the expected performance of the SDS given in Chapter 3. The pumping rate of water through the cleanup system is assumed to be 10 GPM. 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.

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.

$$\frac{10 \text{ gpm} \times 3.785 \times 10^{3} \text{ ml/gal} \times 172 \text{ uCi/ml} \times 10^{-6} \text{ (entr. fact)}}{650 \text{ cfm} \times 2.8 \times 10^{4} \text{ ml/cf} \times 100 \text{ (DF)}} =$$

$$= 3.58 \times 10^{-9} \text{ uCi/cc}$$

In the development of the ^{129}I source term, the results from the above method yields an SDS plant vent concentration of 2.09 x 10^{-10} uCi/cc. The contribution from evaporation is added which increases this concentration to 6.15 x 10^{-10} uCi/cc.

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 ^{3}H.

Table 6.3 lists the concentration of radionuclide source terms in the Chemical Cleaning Building HVAC system following treatment. Release rates for the various nuclides are also shown. The concentrations in the effluent from the Chemical Cleaning Building are below detectable levels for all isotopes, except ^{3}H.

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 for the SDS Off-gas system. For the Chemical Cleaning Building HVAC vent, data was calculated for a ground release. 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 6.4. The data indicates that the point of maximum exposure to a hypothetical individual living near the site is 2413m away in the NNE direction since the most significant radiation release is from plant vent stack.

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.

The deposition rate from the atmosphere for radioactive material was calculated using the methods described in Regulatory Guides 1.109 and 1.111.

Scavenging of radionuclides in the plume is the process through which rain or snow washes out particles or dissolved gases and deposits them on the ground or water surfaces. In the 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 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 ad... by more than a factor of three would be unusual for all pathways except the atmosphere-pasture-cow-milk pathway for ^{129}I as it contributes to the thyroid dose. Therefore a separate estimate of the dose to the thyroid of both the infant and child has been performed.

Total dose commitments are calculated for the specified amount of each isotope released during 50 days 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

Th maximum dose to a hypothetical adult individual is calculated for the four organs and assumes the processing of 710,000 gallons of water. These estimated dose exposure levels, based on simultaneous releases from the plant vent and Chemical Cleaning Building HVAC vents, are:

Total Body	9.47×10^{-3} mrem
Bone	3.57×10^{-4} mrem
Thyroid	4.65×10^{-1} mrem
GI Tract	8.82×10^{-3} mrem

This level of exposure to the total body represents approximately 0.2% of the allowable dose exposure recommended in 10 CFR 50, Appendix I, of 5 mrem.

Table 6.5 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 78% of the dose to total body, 98% to the bone, 99% 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 6.6. Tritium contributes approximately 93% of the dose to total body, and 99% of the dose to the GI tract. Iodine-129 contributes 98% of the dose to thyroid and 58% of the bone dose.

The contribution to the individual organ doses was primarily from the SDS Off-gas releases. The contribution to the dose from releases from the Chemical Cleaning Building was less than 1%.

Because of the possible dependency of the dose to certain organs on age, a separate estimate was made of dose to the thyroid of an infant and child. This calculation yielded a dose of 1.20×10^{-1} mrem for infant thyroid and 3.36×10^{-1} mrem for the thyroid of a child.

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 for 50 days 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 radius of 50 miles from the TMI site. The population integrated dose is calculated to be 0.15 man-rem total body and 4.42 man-rem for the thyroid.

Table 6.1
Activity Level of Water
(October 1, 1980)

Isotope	Containment Sump (uCi/ml)	EPICOR-II Effluent (uCi/ml)
^{3}H	9.66×10^{-1}	9.66×10^{-1}
^{89}Sr	1.80×10^{-1}	6.79×10^{-7}
^{90}Sr	2.64×10^{0}	9.01×10^{-6}
^{125}Sb	9.10×10^{-3}	3.83×10^{-5}
^{129}I	1.36×10^{-5}	1.36×10^{-5}
^{134}Cs	2.77×10^{1}	3.17×10^{-6}
^{137}Cs	1.72×10^{2}	3.31×10^{-5}

Table 6.2
Source Terms for Gaseous Effluents
SDS OFF-Gas System

(October 1, 1980)

Radionuclide	Concentration In SDS Effluent[a] uCi/cc	Concentration In Plant Effluent[b] uCi/cc	Release Rate uCi/sec
H-3	2.66×10^{-5}	1.71×10^{-7}	8.14×10^{0}
Sr-89	3.74×10^{-12}	2.41×10^{-16}	1.15×10^{-8}
Sr-90	5.49×10^{-11}	3.55×10^{-15}	1.69×10^{-7}
Sb-125	1.89×10^{-13}	1.22×10^{-17}	5.81×10^{-10}
I-129	6.15×10^{-10}	4.00×10^{-12}	1.89×10^{-4}
Cs-134	5.76×10^{-10}	3.72×10^{-14}	1.77×10^{-6}
Cs-137	3.58×10^{-9}	2.31×10^{-13}	1.10×10^{-5}

(a) This is the radionuclide concentration in the off-gas (650 ft^3/min) following treatment, from the SDS prior to entering the existing effluent treatment system in TMI-2. A DF of 100 and an entrainment factor of 10^{-6} have been assumed for particulates. An entrainment factor of 7.5×10^{-3} has been assumed for ^{129}I. No effluent treatment is assumed for ^{3}H or ^{129}I.

(b) This is the radionuclide concentration in the off-gas (100,650 ft^3/min) from TMI-2 as it enters the atmosphere. An additional DF of 100 is assumed for particulates, however, no further treatment is assumed for either ^{129}I or ^{3}H.

Table 6.3
Source Terms for Gaseous Effluents
Chemical Cleaning Building
(As of October 1, 1980)

Radionuclide	Concentration in HVAC Vent[a] uCi/ml	Release Rate uCi/sec
^{3}H	7.15×10^{-9}	2.70×10^{-2}
^{89}Sr	4.99×10^{-17}	1.88×10^{-10}
^{90}Sr	6.62×10^{-16}	2.50×10^{-9}
^{125}Sb	2.96×10^{-15}	1.12×10^{-8}
^{129}I	5.03×10^{-15}	1.91×10^{-8}
^{134}Cs	2.34×10^{-16}	8.83×10^{-10}
^{137}Cs	2.44×10^{-15}	9.22×10^{-9}

(a) This is the concentration in the HVAC vent (8000 CFM) from the Chemical Cleaning Building as is enters the atmosphere.

TABLE 6.4

ATMOSPHERIC DISPERSION FACTORS FOR TMI (ANNUAL AVERAGE)

(Sec/m^3)

DISTANCES (Meters)

SECTION	610	2413	4022	5631	7240	12067	24135	40225	56315	72405
N	9.06-7	4.66-7	2.43-7	8.52-8	6.83-8	5.33-8	1.88-8	8.87-9	5.90.9	4.33-9
NNE	8.06-7	1.46-6	4.[illegible]5-7	2.45-7	2.05-7	8.81-8	2.78-8	1.22-8	7.99-9	5.81-9
NE	7.93-7	7.14-7	2.90-7	2.79-7	1.92-7	8.75-8	2.73-8	1.20-8	7.76-9	5.61-9
ENE	9.60-7	5.71-7	2.81-7	1.70-7	1.51-7	8.30-8	2.63-8	1.17-8	7.59-9	5.50-9
E	1.59-6	7.39-7	2.87-7	3.10-7	2.10-7	9.24-8	2.95-8	1.29-8	8.31-9	5.97-9
ESE	1.83-6	1.04-6	5.70-7	2.96-7	2.01-7	8.80-8	2.78-8	1.23-8	8.00-9	5.77-9
SE	2.27-6	1.30-6	5.94-7	3.08-7	2.09-7	9.02-8	2.82-8	1.24-8	8.00-9	5.73-9
SSE	1.42-6	4.54-7	3.82-7	2.20-7	1.59-7	1.02-7	3.17-8	1.39-8	8.99-9	6.47-9
S	7.76-7	4.66-7	5.10-7	2.33-7	1.59-7	7.02-8	2.21-8	9.79-9	6.36-9	4.60-9
SSW	3.15-7	1.77-7	1.10-7	9.85-8	7.18-8	4.31-8	1.37-8	6.07-9	3.93-9	2.83-9
SW	4.34-7	2.24-7	2.50-7	2.10-7	1.42-7	6.04-8	1.87-8	8.17-9	5.29-9	3.81-9
WSW	6.88-7	4.79-7	4.81-7	2.49-7	1.69-7	7.78-8	2.43-8	1.06-8	6.88-9	4.96-9
W	1.05-6	4.51-7	3.66-7	2.78-7	2.21-7	9.94-8	3.06-8	1.34-8	8.64-9	6.22-9
WNW	1.00-6	4.01-7	4.36-7	2.74-7	2.33-7	9.95-8	3.12-8	1.37-8	8.88-9	6.42-9
NW	8.54-7	8.08-7	8.55-7	5.99-7	4.54-7	2.48-7	6.28-8	1.11-8	7.38-9	5.42-9
NNW	7.41-7	4.06-7	2.13-7	8.62-8	9.30-8	7.20-8	2.31.8	1.03-8	6.70-9	4.88-9

NOTE: Atmospheric dispersion factors for elevated release.

Table 6.5

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	78%	98%	99%	76%
Inhalation	22%	1%	<1%	24%

(a) Includes exposure from contaminated ground surface and exposure from immersion in any plume.

Table 6.6

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

Radionuclide	Total Body	Bone	Thyroid	GI Tract
	(% Contribution to dose)			
^{3}H	93%	-	2%	99%
^{89}Sr	<1%	<1%	<1%	<1%
^{90}Sr	<1%	21%	<1%	<1%
^{125}Sb	<1%	<1%	<1%	<1%
^{129}I	6%	58%	98%	<1%
^{134}Cs	<1%	4%	<1%	<1%
^{137}Cs	<1%	17%	<1%	<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. The following accident analysis has been performed based on the assumption that zeolite beds are radiologically loaded to 40,000 Ci. Should higher radiological loadings be determined to be appropriate, the accident analysis will be reassessed using the higher radiological loadings.

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 340 curies).

It is assumed that the total activity is made up of Cesium, 47 Ci of Cs-134 and 293 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 0.39 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 (USDHEW, 1970). The depth of water in the pool is 38 feet. The calculated maximum exposure rate at six feet above the surface is 116 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.

Although the analysis of this hypothetical accident provides results that indicate radiation field of 116 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 limited.

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

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

Assumptions:

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 are drained into the pool It is further assumed that the contaminated water contains 0.77 Ci/gallon of activity, as Cs-134 and Cs-137 in the same concentration ratios that were assumed for the previous hypothetical 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 8.64 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 174 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 radiation levels, area radiation monitors would indicate the presence of high radiation. Personnel would be evacuated from the area to ensure that occupational exposures are limited.

7.3 <u>Inadvertent lifting of prefilter above pool surface</u>

<u>Assumptions</u>:

It is assumed that due to a failure in the crane control system, the over head 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 bay, 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 dose rate at a distance of 15 feet from the filter. The calculated dose rate is 21 Rem/hr and is based on an assumed filter loading of 1000 curies.

<u>Occupational Exposure Effects</u>:

As the filter assembly nears 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.

<u>Off-site Effects</u>:

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.

7.4 Inadvertent lifting of zeolite ion exchanger above pool surface

Assumptions:

It is assumed that due to multiple failures, a zeolite vessel is lifted from the pool resulting in the exposure of plant operating personnel.

Analysis of the accident is performed by modeling the zeolite ion exchanger bed in cylindrical geometry and calculating the dose rate at a distance of 20 feet from the surface of the zeolite ion exchanger. The calculated dose rate is approximately 297 Rem/hr based on an estimated zeolite ion exchange bed loading of approximately 5390 Curies of Cesium-134 and approximately 34,600 Curies of Cesium 137.

Occupational Exposure Effects:

As the zeolite vessel nears 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 reduce occupational doses. Airborne contamination would not occur since the activity is fixed on the zeolites.

Offsite Effects:

Airborne contamination as a result of this hypothetical accident would not occur since the activity is contained on the zeolites which are contained in the ion exchanger vessel. The increase in the radiation level at the site boundary would not be significant due to the shielding provided by the Fuel Handling Building walls and the distance to the site boundary.

Conclusions:

The public health and safety is not endangered as a result of this hypothetical accident. Occupational exposures are minimized by evacuation of the area.

7.5 Inadvertent Drop of SDS Shipping Cask

Assumptions:

It is assumed that due to a failure in SDS shipping cask handling equipment an SDS cask containing a zeolite ion exchanger is dropped from the Fuel Handling Building (FHB) crane to the floor at EL 305'. The SDS shipping cask is assumed to drop from the maximum crane lift height. Upon impact with the floor at EL 305', the SDS shipping cask is assumed to experience rupture as well as rupture of the zeolite vessel, thus exposing the dewatered zeolite resins to the FHB atmosphere. The radiation source is approximately 5390 Curies of Cs-134 and approximately 34,600 Curies of Cs-137 on the zeolite ion exchange media. The contribution from other isotopes on the zeolite media and residual containment building sump water (Table 1.1) in the ion exchange media is negligible; it is assumed the 10^{-4} percent of the isotopes are instantaneously released to the FHB atmosphere. This assumption is conservative because the isotopes are absorbed onto the zeolite media. The Fuel Handling Building HEPA filters are assumed to have an efficiency of 99%.

Occupational Effects:

Assuming that the SDS shipping cask ruptures completely exposing the zeolite ion exchanger containing the activity mentioned above, the calculated dose rate is approximately 297 Rem/hr at a distance of 20 feet. Upon the rupture of the cask, radiation monitors will sound announcing the presence of high radiation fields. Personnel could be evacuated from the area to reduce radiation exposures. Airborne contamination will not occur if the zeolite ion exchange vessels remains intact. With the assumption that the vessels rupture and radioactive material becomes airborne, the airborne activity will be reduced to acceptable levels by the Fuel Handling Building HVAC System prior to atmospheric release.

Operational Effects:

1. Impact on systems, structures and components has been considered which could possibly result in adversely affecting the ability to operate these Reactor Plants safely, transfer load or unload fuel safely, or maintain these Plants in a safe cold shutdown condition.

2. Analysis has been conducted which demonstrates that a postulated SDS Cask drop along the proposed travel path would not adversely affect either TMI Unit 1 or Unit 2.

Off-Site Effects:

The increase in radiation level at the site bondary would not be significant due to the shielding provided by the FHB walls and the distance to the site boundary, if the SDS cask ruptures exposing the zeolite ion exchanger. With the assumption that radioactive material escapes, the whole body dose due to the released activity at the site boundary will be less than 10^{-5} mrem for both beta and gamma radiation.

Conclusions:

The public health and safety are not compromised as a consequence of this hypothetical accident.

Chapter 8

Conduct of Operations

The SDS program for operations is divided into a phased approach. These phases are:

8.1 System Development

System development activities have been performed to assure 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 media have been in use for approximately 50 years or more, a development program was conducted at the Oak Ridge National Laboratory, the results of which are documented in ORNL TM-7448, to ensure that the media selected for use at TMI provided optimized performance characteristics of various media using samples of the waters to be processed at TMI. SDS effluent will be polished by EPICOR-II.

Additional development effort has been expended to verify that media loading and dewatering can be accomplished in the intended manner and that the remote tools, necessary for the coupling and de-coupling of the vessels, operates in the intended manner.

8.2 System Preoperational Testing

Prior to use in the SDS each 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 be pneumatically tested to assure leak-free operations. The system will be tested to 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 and, control schemes 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.

8.4 System Decommissioning

The decommissioning plan for SDS is being developed. An outline of the planned approach 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.

1) Equipment and interconnecting piping will be decontaminated. the levels to which decontamination is accomplished will depend on the intended disposition of individual items, i.e., disposal or reuse.

2) The system will be disassembled, component by component.

3) Major system components can be stored for later use or disposed of at a licensed burial facility.

4) Small components, such as valves, piping, instruments, etc. can be disposed of as radioactive waste.