NUREG-0796

Safety Evaluation Report
related to the operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit No. 2

Docket No. 50-320

Metropolitan Edison Company
Jersey Central Power and Light Company
Pennsylvania Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1981

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Abstract

A Safety Evaluation Report related to the operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit 2, located in Dauphin County, Pennsylvania, has been issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This Safety Evaluation Report presents the staff's analysis of information submitted by the licensee (Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company).

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1. INTRODUCTION

As a result of the March 28, 1979, accident at the Three Mile Island Nuclear Station, Unit 2, significant volumes of contaminated water were generated and collected in the reactor and auxiliary buildings of the plant. Several hundred thousand gallons of intermediate* level waste water collected in auxiliary building tanks. Additionally, in excess of one-half million gallons of high** activity waste water collected in the reactor building sump and the approximately 95,000 gallons of reactor coolant was contaminated to high** activity levels. Recognizing that these contaminated bodies of water must either be removed from their present locations or processed for the cleanup to proceed, Metropolitan Edison Company (the licensee) initiated the design of systems to process the intermediate and high activity waste water.

The United States Nuclear Regulatory Commission directed its staff on May 25, 1979, to prepare an environmental assessment regarding proposals to decontaminate and dispose of radioactively contaminated waste water from the Three Mile Island facility. The assessment was to be divided into several portions. The first portion dealt with the proposed decontamination of intermediate level waste water using the EPICOR-II system and was issued on October 3, 1979. The Commission approved the use of EPICOR-II on October 16, 1979, and the licensee has processed approximately 570,000 gallons of intermediate level waste water to date.

The decontamination and disposal of the high activity waste water in the containment building sump and Reactor Coolant System (RCS) was to be addressed in a subsequent assessment. Prior to issuance of that assessment, however, the Commission directed the staff, on November 21, 1979, to prepare a programmatic environmental impact statement on the decontamination and disposal of radioactive wastes resulting from the March 28, 1979, accident. Consistent with the Commission's Statement of Policy of November 21, 1979 (Reference 1), the Director, Office of Nuclear Reactor Regulation, by Order dated February 11, 1980, imposed a requirement that the processing and discharge of water in the containment building and RCS not be undertaken until approved by the NRC.

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*After the accident, intermediate level waste water was defined as water having I-131 and Cs-137 concentrations greater than 1 uCi/ml but less than 100 uCi/ml.

**High activity waste water was defined as water having I-131 and Cs-137 concentrations greater than 100 uCi/ml.

By letter dated April 10, 1980 (Reference 2), the licensee submitted a Technical Evaluation Report (TER) on the Submerged Demineralizer System (SDS) and requested NRC approval to use the SDS for decontamination of the radioactive waste water in the TMI-2 containment building sump and RCS. We advised (Reference 4) the licensee that approval of any system to process the waste water in the reactor building sump and RCS would be preceded by the NRC's evaluation of alternative techniques, as discussed in our Final Programmatic Environmental Impact Statement (PEIS), consistent with the Commission's Statement of Policy of November 21, 1979. Our Final PEIS was issued on March 9, 1981 (Reference 19). The SDS is an underwater liquid radwaste processing system, located in the "B" spent fuel pool of the TMI-2 fuel handling building. The SDS is designed to decontaminate the high activity waste water by filtration and ion exchange. In order to complete the safety and environmental review of the licensee's submittal, we requested (References 3-6) additional information to supplement the SDS TER. In response to our requests for additional information about the SDS, the licensee provided the technical information requested (References 7-9), including a revised TER (Reference 10) which was submitted on March 11, 1981. In the revised TER, the licensee proposes to use the existing EPICOR-II processing system to polish the effluent from the SDS and requests approval to operate the SDS/EPICOR-II as the method to decontaminate the containment sump water and RCS water. After reviewing the licensee's revised TER, we requested a meeting with the licensee to obtain additional information to supplement our safety review. On March 31, 1981, and April 1, 1981, we met with the licensee to discuss our request for additional information. As a result of this and subsequent meetings, the licensee submitted additional information (References 11-17) needed to complete our safety review.

The Commission issued a Statement of Policy on April 27, 1981 (Reference 20) concluding that the PEIS satisfies NRC obligations under the National Environmental Policy Act (NEPA). The Statement of Policy provides that the staff may act on each major cleanup activity without a further environmental assessment if the activity and associated environmental impacts fall within the scope of those already assessed in the PEIS (Reference 19). In addition, the Commission stated that the cleanup should be carried out expeditiously consistent with insuring protection of public health and safety and the environment.

Further, the Commission stated that the cleanup should be carried out in accordance with the criteria in Appendix R of the PEIS (Reference 19) as well as in conformance with the existing operating license (DPR-73) and with previously imposed orders. Although the criteria of Appendix R have not yet been incorporated into the facility license, the licensee has submitted a technical specification change request (Reference 24) to add these criteria to the facility operating license and has committed to implement these criteria prior to operation of the SDS. We are currently evaluating the licensee request and will act on the request in a separate action. Therefore, this safety evaluation was performed to determine the capability of the SDS to operate within the limits of Appendix R.

This report provides our evaluation of the licensee's request to decontaminate the reactor building sump water and RCS water using the SDS, followed by polishing in EPICOR-II. This evaluation only addresses the processing of the reactor building sump water and RCS water and does not consider the disposition of the processed water. The licensee has not forwarded any proposals to the NRC regarding the disposition of TMI-2 processed accident generated water and has indicated that no proposals will be made prior to 1982. Further, with the cooperation of the State of Maryland, studies are currently planned to determine the potential socioeconomic impact of the alternative of disposing of TMI-2 processed accident generated water by dilution and controlled release to the Susquehanna River. In our view, any such disposal actions taken in advance of completing these studies would be premature and without the benefit of information on societal and socioeconomic factors important to the decision-making process. Consistent with the Commission's April 27, 1981, Statement of Policy (Reference 20), any future proposals by the licensee for the disposition of processed accident-generated water will be transmitted to the Commission for approval. Any such disposal will be made in accordance with the agreements with the City of Lancaster (Reference 31). Pending decisions regarding the ultimate disposition of processed waste water, the water will be stored in available on-site tankage and portions may be cycled for re-use within the plant. Potential re-use of this water may include decontamination and flushing activities, reflood of the containment sump for shielding and control of airborne contamination, and makeup to the reactor coolant system.

2. SYSTEM DESIGN CRITERIA

2.1 Applicable Regulations

The design of the SDS is based on the criteria of 10 CFR Part 20, "Standards for Protection Against Radiation," and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

The criteria of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establish minimum requirements for the principal design criteria for light-water-cooled nuclear power plants, including the plant radioactive waste management systems. Specifically, Criterion 60, "Control of Releases of Radioactive Materials to the Environment," requires that the nuclear power unit design include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational occurrences. Criterion 61, "Fuel Storage and Handling and Radioactivity Control," requires the radioactive waste management systems to be designed with suitable shielding for radiation protection and with appropriate containment, confinement and filtration systems. Criterion 1, "Quality Standards and Records," and Criterion 2, "Design Bases for Protection Against Natural Phenomena," require design provisions relating to seismic and quality group classification and quality assurance aspects of radioactive waste management systems, structures and components. Criterion 64, "Monitoring Radioactivity Releases," requires that nuclear power plant design provide means for monitoring effluent discharge paths for radioactivity that may be released during normal operations, including anticipated operational occurrences, and from postulated accidents. Section 20.106 of the Commission's regulations establishes limits on concentrations of radioactive material in effluents to unrestricted areas. Section 20.1(c) states that the licensee should, in addition to complying with the limits set forth in Part 20, make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as far below those limits as is reasonably achievable. Section 50.36a provides that, in order to keep power reactor effluents as low as is reasonably achievable, each license authorizing operation of such a facility will include technical specifications which (1) establish operating procedures for the control of effluents, (2) require the installation, maintenance, and use of effluent control equipment, and (3) require the reporting of actual releases. The requirements of Section 50.36a are implemented in the existing Appendix B Technical Specifications.

Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50 provides numerical guidance for design objectives and technical specification requirements for limiting conditions for operation to meet the criteria "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents. It should be noted that the criteria approved by the Commission for radiological effluents from decontamination activities at TMI-2, as described in Appendix R of the PEIS, impose the design objectives of Appendix I as limits which are not to be exceeded during the cleanup of TMI-2. Imposition of 10 CFR Part 50, Appendix I criteria as mandatory limits, in lieu of design objectives, is more stringent for TMI-2 than for a normally operating nuclear power facility, in recognition of the unique situation at TMI-2.

2.2 Regulatory Guides

To implement the requirements in 10 CFR Part 20 and 10 CFR Part 50 for the design, construction, testing and operation of the SDS, the licensee utilized the guidance provided in the following regulatory guides which were prepared and approved by the NRC staff:

Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

Regulatory Guide 1.8, "Personnel Selection and Training."

Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable."

Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Reactor Plants."

We conclude that the provisions for the design, construction, testing, and operation of the SDS in accordance with the regulatory guides listed above are appropriate. Our detailed evaluation of the conformance of the SDS design and design criteria with the applicable regulations and guides referenced above is provided in Section 3 (System Description, Operation, and Evaluation) and Section 4 (Radiation Protection).

3. SYSTEM DESCRIPTION, OPERATION AND EVALUATION

3.1 Introduction

The SDS is designed specifically to provide for the controlled handling and treatment of the highly contaminated waste water in the containment sump and RCS and the radioactive gases and solid radioactive wastes which are generated during operation of the SDS. The SDS has been installed in and is designed to be operated within the TMI-2 fuel handling building. The SDS consists of a liquid waste treatment subsystem (LWTS), gaseous waste treatment subsystem (GWTS), solid waste handling subsystem (SWHS) and process and effluent radiological monitoring and sampling subsystem (PMSS). The LWTS will decontaminate the containment sump and RCS water by filtration and ion exchange. The GWTS will filter and adsorb the radioactive material that may become airborne during water processing operations, sampling operations, spent liner dewatering and spent liner storage. The SWHS will provide for dewatering, storage, and cask loading of the spent filter and resin liners generated during the processing of the high activity waste water. The PMSS will provide a direct measure of the process performance of the SDS.

In our evaluation of the SDS, we have considered (1) the capability of the system to keep the levels of radioactivity in gaseous effluents "as low as is reasonably achievable", in accordance with 10 CFR Part 20, 10 CFR Part 50.36a, and the criteria specified in Appendix R of the Final PEIS, (2) the capability of the system to maintain releases below the limits in 10 CFR Part 20, Appendix B, Table II, Column 1, (3) the quality group of the SDS and seismic design classification of the building housing the SDS, (4) the expected types, volumes, and activities of solid waste generated from waste water processing, (5) solid waste packaging and conformance to applicable Federal regulations, (6) the provisions for onsite storage prior to shipment for subsequent processing or disposal, and (7) the capability of the system to monitor the performance of process equipment and detect radioactive material leakage between systems.

The waste water to be processed consists of approximately 700,000 gallons of containment sump water and 95,000 gallons of RCS water. The radionuclide distributions in the containment sump and RCS water are listed in Tables 1 and 2, respectively. As indicated in the tables, the principal contaminants in the waste water are radionuclides of cesium and strontium. The SDS is especially designed to process high activity* waste water.

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*See definition on page1-1. It should be noted that I-131 decayed to insignificant levels within several months after the March 28, 1979, accident and the concentration of Cs-137 in the RCS is now less than 100 uCi/ml, due primarily to dilution of the RCS water with clean makeup water.

TABLE 1

Radionuclide Distribution In The
Containment Building Sump[c]
As Of March 17, 1981

Isotope	Radionuclide Concentration[a] (uCi/ml)	Radionuclide Inventory (Ci)[b]
H-3	0.8	2.1×10^{3}
Cs-134	19	5.0×10^{4}
Cs-137	140	3.7×10^{5}
Sr-90	5.0	1.3×10^{4}
Nb-95	$<1.0 \times 10^{-3}$	<3.0
Zr-95	$<1.1 \times 10^{-3}$	<3.0
Ru-106	$<1.0 \times 10^{-3}$	<3.0
Sb-125	1.5×10^{-2}	40
Ce-144	6.0×10^{-3}	16
I-129	5.7×10^{-6}	1.5×10^{-2}
	TOTAL	4.4×10^{5}

[a]Containment Building Sump volume of 700,000 gallons.

[b]All values rounded to two significant figures.

[c]Based on analysis of sump water sample of March 17, 1981 by the Oak Ridge National Laboratory.

TABLE 2

Radionuclide Distribution In The
Reactor Coolant System
As Of April 6, 1981

Isotope	Radionuclide Concentration[a] (uCi/ml)	Radionuclide Inventory Ci[d]
H-3[b]	6.0×10^{-2}	22
Cs-134[b]	3.0	1.0×10^{3}
Cs-137[b]	22	8.0×10^{3}
Sr-90[b]	19	7.0×10^{3}
Nb-95[c]	5.8×10^{-4}	2.0×10^{-1}
Zr-95[c]	3.7×10^{-4}	1.0×10^{-1}
Ru-106[c]	7.3×10^{-2}	26
Sb-125[c]	2.9×10^{-3}	1.0
Te-125m[c]	1.2×10^{-3}	4.0×10^{-1}
Te-127m[c]	2.2×10^{-2}	8.0
Te-129m[c]	4.4×10^{-5}	2.0×10^{-2}
Ce-144[c]	2.2×10^{-2}	8.0
Co-58[c]	1.5×10^{-4}	5.0×10^{-2}
	TOTAL	1.6×10^{4}

[a]RCS volume of 95,000 gallons

[b]RCS Sample Results as of 4/6/81

[c]Calculated values based on PEIS Table 7.7 and an assumed dilution rate of 0.1 gpm to 4/6/81

[d]All values rounded to two significant figues

The principal processing components of the system are submerged in the "B" spent fuel pool to utilize the shielding characteristics of the pool water. Primary waste water decontamination will take place in the LWTS of the SDS and, if necessary, the SDS effluent may be polished in the existing EPICOR-II system. The logic plan for waste water processing is shown in Figure 1.

3.2 Liquid Waste Treatment

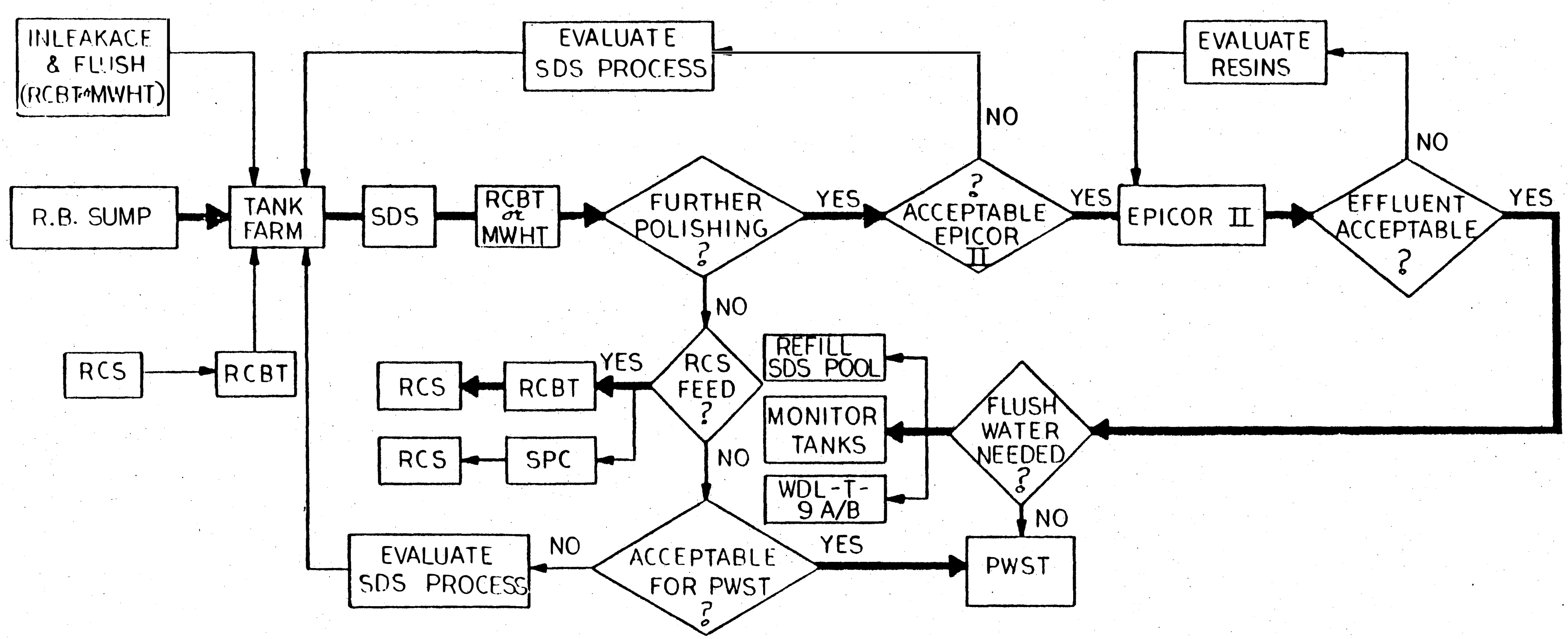
The waste water in the containment sump and RCS will be processed in batches consisting of a series of separate processing steps. For the containment sump water, a batch will consist of approximately 50,000 gallons and the processing steps will include water delivery and filtration in the LWTS, ion exchange in the LWTS, polishing in the EPICOR-II system if needed, and storage or reuse of the processed water within the plant. For the RCS water, a batch will also consist of approximately 50,000 gallons and the processing steps will include letdown of the RCS water using the existing letdown system to a holding tank (one of the three existing reactor coolant bleed tanks), water delivery and filtration (if necessary) in the LWTS, ion exchange in the LWTS, polishing in the EPICOR-II system (if necessary), and storage or re-use of the processed water within the plant. RCS letdown will be accomplished by feed and bleed (i.e., removal of contaminated water and addition of borated processed makeup water at the same flow rate) to maintain system inventory and pressure.

A schematic diagram of the flow paths for processing containment sump and RCS water through the SDS/EPICOR-II is given in Figure 2. The LWTS consists of process equipment and instrumentation necessary to collect, process, monitor, store and recycle high activity waste water. The LWTS process train consists of a prefilter, final filter, holdup tank, two parallel trains of three ion exchange vessels each, two downstream ion exchange vessels in parallel, and a post filter. The major components making up the LWTS, along with their principal design parameters, are listed in Tables 3 - 8.

The contaminated water in the containment sump will be removed by a 30 gpm surface suction pump and pumped through flexible hose to a containment penetration and connecting piping leading to the fuel handling building. A backup system for delivering water from the sump exists. It utilizes the reactor building waste pump which is located in the auxiliary building to take suction directly from the containment sump. However, the surface suction pump will be the preferred means. The surface suction pump is designed to float in the containment sump water and takes suction approximately 1 foot below the water surface to eliminate any potential interference with debris floating on the surface. The pump is operated from a control panel located on a platform over the "B" spent fuel pool. The piping in the fuel handling building directs the waste water through two filters in series which are submerged in the "B" spent fuel pool. The first filter (the prefilter) is a roughing filter designed to remove suspended solids greater than 125 microns in size and the second filter (the final filter) is designed to remove particles greater than 10 microns in size that pass through the prefilter. The filters were designed and installed to preclude plugging of the downstream ion exchange vessels. As the licensee gains operational experience with the filtration process, the filter micron sizing may be altered to optimize filtration. The filters are cartridge type and will be replaced when high differential presure monitored across either

FIGURE 1

PROCESSING LOGIC PLAN



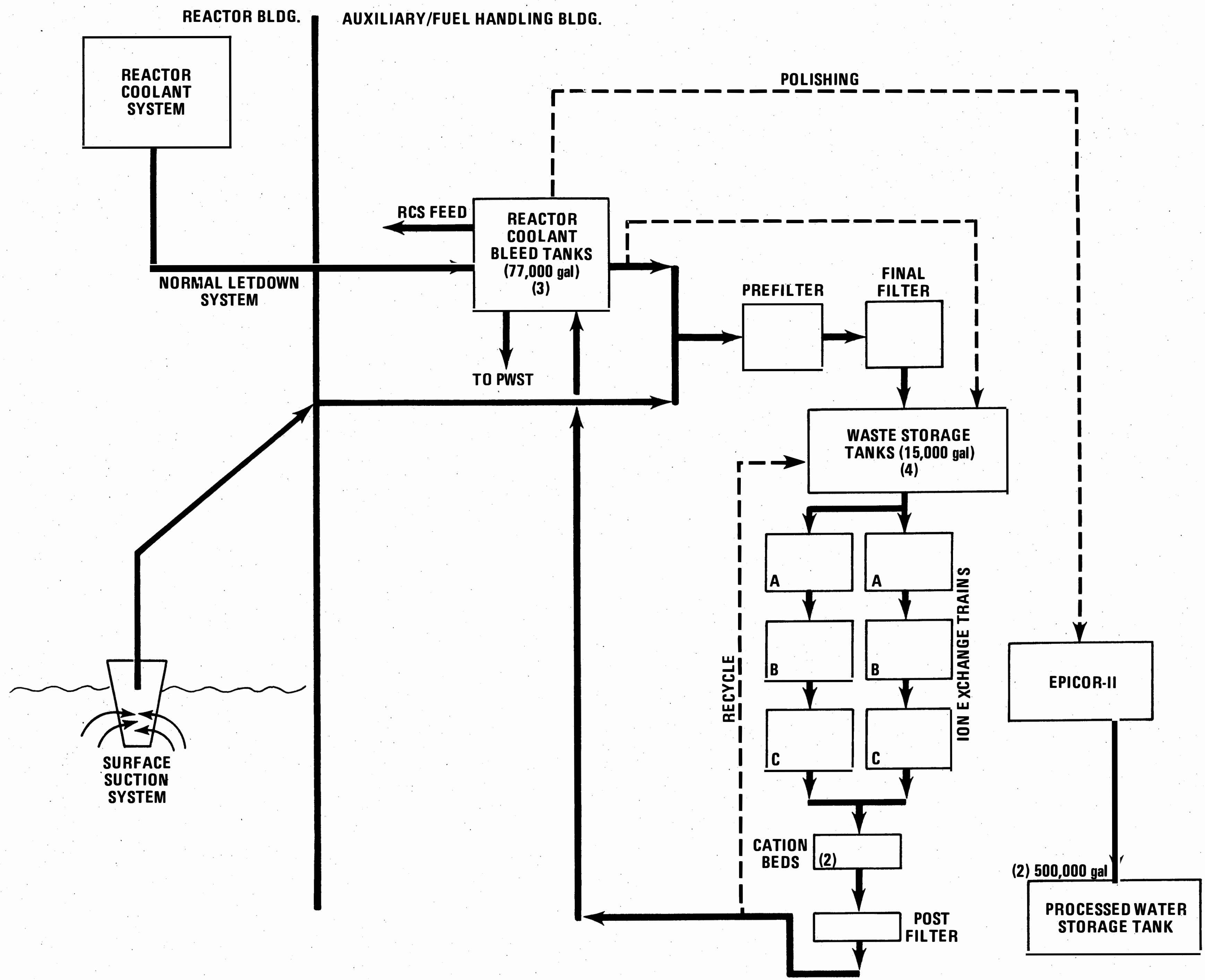


FIGURE 2 SDS LIQUID FLOW PATHS

TABLE 3

SDS Prefilter/Final Filter

Vessel Details	
Installation/Location	Vertical, "B" Spent Fuel Pool
Outside Diameter/Height, Ft., In.	2 Ft. ½ In./4 Ft. 5½ Inch
Shell Thickness	3/16 Inch
Shell Material	Stainless Steel (304)
Design Pressure/Temperature	150 PSIG/100 ^{o}F
Volume	10 Cubic Feet
Prefilter Size	3/16 Inch Roughing Screen/ 125 Micron
Final Filter Size	10 Micron

TABLE 4

SDS Waste Storage Tanks

Vessel Details	
Identification	WDL-T-2A through 2D*
Capacity - Gallons	60,000 Gal. (Total)
Installation	Horizontal
Outside Diameter/Height, Ft. In.	11 Ft. 6 In./21 Ft.
Shell Material	Stainless Steel (304)
Shell Thickness, In.	¼ Inch
Design Temperature, ^{o}F	200 ^{o}F
Design Pressure, PSIG	15 PSIG

*4 tanks 15,000 gallons each considered as one tank.

TABLE 5

SDS Ion-Exchange Vessels

Vessel Details	
Number Installed	8
Installation	Vertical
Outside Diameter/Height, Ft. In.	2 Ft. ½ in./4 Ft. 5½ in.
Shell Thickness	3/8 Inch
Shell Material	Stainless Steel (316L)
Design Pressure/ Temperature	350 PSIG/400 ^{o}F
Volume	10 Cubic Feet

TABLE 6

SDS Surface Suction Pump

Pump Details	
Name	Surface Suction
Type	Vertical 3 stage high capacity well pump
Rated Speed	3600 RPM
Rated Capacity	30 GPM
Rated Total Dynamic Head	200 Ft.
Float Material	Polystyrene Foam
Dimensions Float Material	60" x 6.5" x 18"

TABLE 7

SDS Feed Pump

Pump Details	
Name	Feed Pump
Type	Submersible/Centrifugal
Standard Material Designation	Stainless Steel
Rated Speed	3500 RPM
Rated Capacity	30 GPM
Rated Total Dynamic Head	240 Ft.
Design Temperature	100 ^{o}F

TABLE 8

Post Filter

Vessel Details	
Number Installed	One (1)
Installation	Vertical
Outside Diameter/Height, Ft., In.	10 5/8 In./3 Ft. 5 7/16 In.
Shell Thickness	0.165 Inch
Shell Material	Stainless Steel (304)
Design Pressure/Temperature	150 PSIG/100 ^{o}F
Volume	0.61 Cubic Feet
Post Filter Size	0.45 Micron

filter indicates low flow. The filtration process is controlled by valves located inside the shielded SDS filter manifold which is located on a platform over the "B" spent fuel pool, and by the instrumentation which provides flow indication from the surface suction pump and differential pressure across the filters. After filtration, the waste water is collected in the existing upper tank farm located in the "A" spent fuel pool. The upper tank farm consists of four interconnected 15,000 gallon tanks which function as a single 60,000 gallon tank. The tank system communicates directly with a standpipe, located adjacent to the tanks, which contains a submersible pump. This tank system, standpipe and pump serve as the feed system to the LWTS ion exchangers.

When processing RCS water, the reactor coolant will be withdrawn from the RCS by a feed and bleed process using the existing letdown system to a 77,000 gallon reactor coolant bleed tank. If the RCS water requires filtration, as determined from water sampling, it will be pumped from the bleed tank through the LWTS filters in a manner similar to the containment sump water treatment. The water will be recirculated thoroughly and sampled and analyzed prior to processing to completely characterize the water.

The next treatment step in the processing of containment sump or RCS water is decontamination by ion exchange. The submerged pump in the standpipe of the upper tank farm delivers waste water to the LWTS ion exchange vessels via the shielded feed and ion exchange manifold. The ion exchange portion of the LWTS consists of two submerged parallel trains of 4 zeolite ion exchange vessels (beds 1, 2, 3 and 4) each, planned to be operated as single trains or together as parallel trains. The design flow rate through each 4 vessel train is 5 gpm and the design flow through the combined 2 trains is 10 gpm, the maximum flow through the LWTS.

The processing philosophy for the SDS is to remove essentially all of the cesium and strontium radionuclides (the principal contaminants in the containment sump and RCS water) on inorganic ion exchange media in the LWTS. Each of the ion exchange vessels will be loaded with the same ion exchange media (approximately 8 cubic feet) consisting of a homogeneous mixture of two types of inorganic zeolite media. The mixture of zeolite media exhibits strong removal specificity for both cesium and strontium radionuclides. During process operations, the curie loading of the first ion exchange vessel (bed 1) in each 4 vessel train will be administratively limited to approximately 60,000 curies*, consisting

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*This limit is based on conservatively projected ion exchange capacity of the zeolite for cesium and shielding capability of the shipping cask. Higher vessel loadings, and correspondingly lower waste volumes, may be possible if: (a) the actual operations indicate additional ion exchange capacity; (b) the shielding capability of the shipping cask has been underestimated. The cask loading limit will be determined with actual spent zeolite vessels, or (c) another shipping cask with a higher loading capability is utilized.

In either case, the staff considers the 60,000 curie limit as an initial administrative limit. Higher loadings to 120,000 curies were evaluated in our PEIS (Reference 19) and are considered to be within the scope of this SER.

mostly of cesium radionuclides. Upon reaching that limit, the vessel will be flushed with processed water, removed from the system, dewatered, capped, placed in storage in submerged storage racks in the "B" spent fuel pool, and attached to a vent header in the GWTS to relieve any potential buildup of non-condensible gas. A new vessel with fresh zeolite media will be placed in the bed 1 position or, alternatively, downstream beds will be rotated forward dependent upon the degree of radionuclide breakthrough from the first bed prior to changeout. If the beds are rotated forward, zeolite bed 2 will be moved to the bed 1 position in the train and, if necessary, zeolite bed 3 will be placed in the bed 2 position and so forth. A new zeolite vessel will be placed in the open position and the SDS will resume operation until bed 1 is loaded to the administrative limit, whereupon, the bed changeout will be repeated. The licensee will operate the system to minimize vessel handling and changeout and corresponding worker exposure. Operation of the SDS in the manner described above will minimize the volume of solid waste (spent ion exchange vessels) generated from decontamination of the containment sump and RCS water.

The effluent from the LWTS process train will be collected in one of the three existing reactor coolant bleed tanks (77,000 gallons each) for sampling and analysis. The LWTS has two 12,000 gallon monitor tanks to collect processed water from the LWTS, however, these tanks will be used to store processed water as a source of flush water to the SDS during process operations. The processed water in the reactor coolant bleed tank (RCBT) can be recycled through the LWTS for further decontamination (if sampling indicates this is necessary), directed to the EPICOR-II system for polishing if needed, placed in storage in available on-site tankage, or re-used in the plant (e.g., makeup to the reactor coolant system or reflood of the containment sump for shielding or control of airborne contamination). Sampling and analysis and the planned usage of the processed water in the RCBT will determine where the water will be directed.

If the planned use of the processed water is for makeup to the RCS or reflood of the containment sump to provide shielding for sludge and plateout activity, the processed water may be placed in temporary onsite storage consistent with the limitations described in Section 7.2.4.2 of the PEIS (Reference 19) and the requirements of 10 CFR Part 20. If the water is to be polished in EPICOR-II and placed in on-site storage pending its ultimate disposal (i.e., no re-use is contemplated) the EPICOR-II system will be operated to control the specific activities of the generated solid waste (i.e., the spent ion exchange liners) for ease of waste handling, packaging and disposal.

Actual processing with the SDS may yield water that does not require further processing or, conversely, it may yield water that requires recycle through the SDS.

The description and our evaluation of the operation of the EPICOR-II system are included in NUREG-0591, Environmental Assessment - Use of EPICOR-II at Three Mile Island, Unit 2, USNRC, October 3, 1979 (Reference 21). The environmental impacts associated with the use of EPICOR-II as a polishing system to process slightly contaminated water are well bounded by the evaluation presented in NUREG-0591 which addresses the processing of water with significantly greater levels of contamination. The EPICOR-II configuration that will be employed to polish SDS effluent will consist of 3 mixed bed (i.e., organic cation and anion resin) liners in series. The first two beds will be 6 feet in diameter by 6 feet high cylindrical vessels and the last bed will be a 4 feet in diameter by 4 feet high cylindrical vessel. The first bed in the system is designed to remove nonradioactive sodium from the influent water to enhance the polishing capability of the second bed. The third bed in the system acts as a guard bed against breakthrough from the second bed. Processing through EPICOR-II will be on a batch basis and a batch will consist of approximately 25,000 gallons. Based on ion exchange column testing by the licensee of the media planned for use in EPICOR-II with samples of actual sump water, we expect the gross concentration (excluding H-3) of the SDS/EPICOR-II processed water to be approximately 7.0×10^{-5} uCi/ml. The processed water will be placed in onsite storage pending a licensee proposal, staff review and a Commission decision regarding its ultimate disposition. The licensee has constructed two 500,000 gallon processed water storage tanks to provide adequate capacity for the storage of processed accident generated water pending its disposal.

Volumes of lesser quality water may be recycled for re-use within the plant with subsequent processing in the SDS and/or polishing in EPICOR-II when this water is no longer needed in the TMI-2 cleanup.

3.3 Gaseous Waste Treatment

The gaseous waste treatment subsystem (GWTS) of the SDS is designed to process gaseous wastes based on the origin and radioactivity levels of the waste generated during water movements, process operations, spent liner dewatering, and spent liner storage. The GWTS is interfaced with the existing fuel handling building ventilation system (FHBVS) and is designed to control the release of gaseous effluents to the FHBVS and subsequently to the environment. The FHBVS is an existing filtration system which has 2 HEPA filter banks in series. A flow schematic of the GWTS is given in Figure 3. The principal components of the GWTS include the off-gas separator tank, off-gas bottoms pump and standpipe, and air filtration unit. The air filtration unit consists of an electric heater, roughing filter, a HEPA filter, a charcoal absorber, a second HEPA filter, and a blower. The major components making up the GWTS, along with their principal design parameters, are listed in Tables 9 through 11.

The off-gas separator tank, located in the shielded spent fuel pool surge pit, collects the flush liquid and moisture-containing off-gas from the spent liner dewatering system, the flush liquid and off-gas from the ion exchangers during vessel vent and fill operations, and drain liquids from the beta monitoring system and manifold, high rad feed sample glove box, ion exchanger manifold,

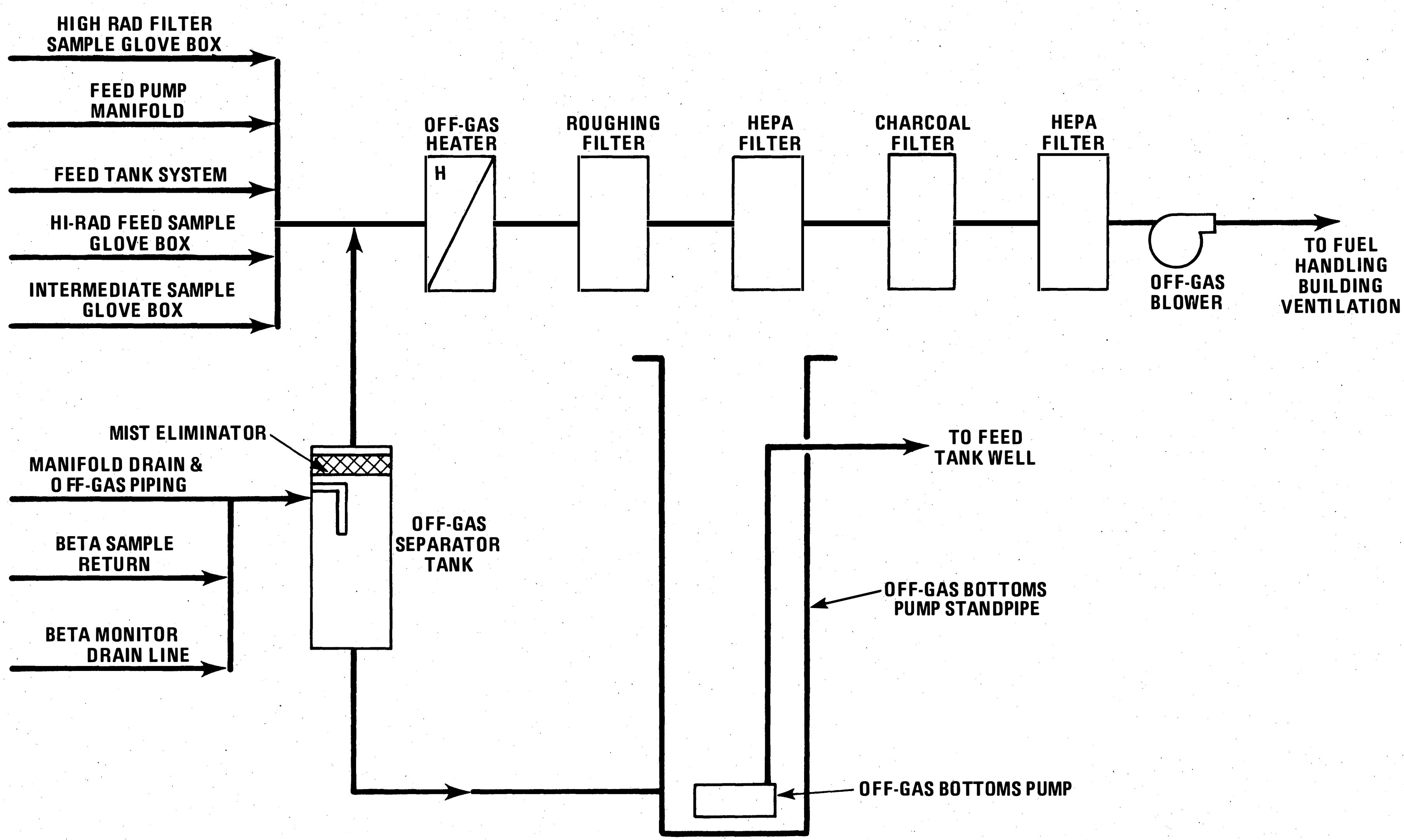

FIGURE 3 SDS OFF-GAS SYSTEM

Table 9

SDS Offgas Separator Tank and Offgas Bottoms Pump

Vessel Details	
No. Installed	One (1)
Capacity - Gallons	590 Gal.
Installation	Vertical
Outside Diameter/Height, ft. in.	36 in./10 ft.
Shell Material	Stainless Steel (304)
Shell Thickness	3/16 inch
Design Temperature, ^{o}F	100 ^{o}F
Design Pressure, PSIG	16 PSIG

Pump Details	
Name	Offgas Bottom Pump
Type	Submersible
Standard Material Designation	400 Series Stainless Steel
Rated Speed	3450 RPM
Rated Capacity	30 GPM
Rated Total Dynamic Head	65 Ft.
Design Temperature	100 ^{o}F
Lubricant	Water

TABLE 10

Offgas Blower

Blower Details	
Name	Offgas Blower
Type	Radial Flow Centrifugal Type "E"
Standard Material Designation	Sheet Steel
Rated Speed	3510 RPM
Rated Capacity	1000 CFM
Rated Static Pressure	12" W.G.
Design Temperature	104 ^{o}F

TABLE 11

GWTS Air Filtration Unit Components

Component	Description
Heater	9KW
Roughing Filter	Waterproof Fiberglass Design pressure drop 8" W.G.
HEPA Filter	2 installed Design efficiency 99.97% for .3 micron particles Size: 24 in. x 24 in. x 11½"
Charcoal Adsorber	1 in. Pleated bed Size: 24" x 24" x 11½"

and intermediate level sample glove box. The off-gas separator tank communicates with the off-gas bottom pump and standpipe and vents to the GWTS air filtration unit. The liquids collected in the off-gas separator tank are returned to the upper tank farm standpipe for subsequent processing in the SDS. The off-gas separator tank has a mist eliminator at the top of the tank to remove entrained moisture from vented effluent gas prior to entering the GWTS air filtration unit. Other gaseous waste input to the air filtration unit includes the vented effluents from the spent filter and ion exchange vessels in storage, the upper tank farm storage tanks and feed pump standpipe, high-rad filter sample glove box, intermediate level sample glove box, off-gas bottom pump standpipe and the valve containment manifolds for the prefilter and final filter, RCS cleanup, ion exchangers, feed pump, and beta monitoring system. Prior to SDS initial operation and after spent ion exchange vessel changeout and replacement, the ion exchange vessels must be vented and filled with water. The licensee intends to vent and fill these vessels with processed water from the SDS monitor tanks or water from the demineralized water system. The flush water and off-gas from these operations are directed to the off-gas separator tank. The spent filters and ion exchange vessels will be dewatered in the SDS dewatering system prior to placement in storage racks in the fuel pool. Dewatering is accomplished by blowing compressed air through the vessel until the free water is removed. The removed free water and exhaust air from the dewatering system are directed to the off-gas separator tank. Following dewatering and capping, the spent vessels are placed in the pool storage racks and the vessel vent lines are connected to a header which is piped to the GWTS air filtration unit. The vessels are continuously vented while in storage to prevent the buildup of pressure in the vessels. During process operations, the valves located in the ion exchanger manifold and beta monitoring manifold have the potential for some leakage. This leakage will collect in the sumps of these valve containment manifolds and gravity drain to the off-gas separator tank.

During waste water processing through the SDS, many samples will be drawn from various points in the process stream for analysis to support process operations. These samples will be collected and handled in the high rad feed sample glove box and intermediate level sample glove box. The purge and drain wastes from sampling operations in these glove boxes will gravity drain or be pumped to the off-gas separator tank or upper tank farm standpipe. Additionally, the effluent from the beta monitoring system will be directed to the off-gas separator tank. During water processing operations, sampling and spent vessel dewatering, there is the potential for small amounts of radioactive material to become airborne. Consequently, the upper tank farm storage tanks and feed pump standpipe, high-rad filter sample glove box, intermediate level sample glove box, off-gas bottom pump standpipe, and off-gas separator tank are vented to the GWTS air filtration unit preventing significant releases to the FHB atmosphere. The air filtration unit provides treatment of the airborne radioactive materials prior to discharge to the FHBVS and subsequently to the environment. Additionally, there is the potential for valve leakage and the generation of airborne radioactive material in the valve containment manifolds for the prefilter and final filter, RCS clean-up, ion exchangers, feed pump, and beta monitoring system. These manifolds are vented to the air filtration unit and maintained at negative pressure relative to ambient by the air filtration unit blower. The valve manifold at the containment penetration for sump water delivery to the SDS filters, located in the southwest corner of the fuel handling building at the 347 foot elevation, is enclosed within a shielded cubicle with sealant for the floor and walls. A continuous air monitor with a local alarm is also provided for this cubicle.

In order to facilitate filtration efficiency, the relative humidity of the off-gas to the air filtration unit is maintained below 70% by the unit electric heater. The off-gas then passes through a roughing filter which removes gross particulates and dust from the air stream to protect the downstream HEPA filter. The air stream passes through the downstream HEPA filter, charcoal adsorber, and a second HEPA filter prior to being discharged to the FHBVS by the off-gas blower. The charcoal adsorber is a 1 inch thick bed. The HEPA filters are in-place tested with DOP smoke to confirm a removal efficiency greater than 99.95% for the smoke particles. The charcoal adsorber is in-place tested with a gaseous halogenated hydrocarbon refrigerant to ensure that bypass leakage through the adsorber is less than 0.05%. Further, the charcoal is laboratory tested to ensure a removal efficiency greater than 95% for methyl iodide at 25° C and 95% relative humidity. The charcoal adsorber will serve to remove essentially all of any I-129 that may be present in the off-gas. The downstream HEPA filter will collect any charcoal fines that become airborne during operation of the air filtration unit. The off-gas blower is a radial flow centrifugal unit with a capacity of 1000 cfm. Pressure gages at the unit suction and discharge will be used to monitor blower performance. The GWTS off-gas will discharge to the FHBVS which has been previously evaluated and found acceptable in NUREG-0107, Safety Evaluation Report Related to Operation of Three Mile Island Unit 2, September 1976 (Reference 22).

3.4 Solid Waste Handling and Disposal

The solid waste handling system (SWHS) is designed to handle, dewater, and store spent ion exchange and filter vessels resulting from the operation of the SDS. The SWHS consists of a dewatering station and underwater storage racks in the "B" spent fuel pool with provision for storage of spent ion exchange and filter vessels. Additionally, the existing fuel handling building (FHB) crane will be used to handle and transfer spent vessels in the pool.

We have evaluated the expected types, volumes, and activities of solid waste generated from waste water processing in the SDS and subsequent polishing in the EPICOR-II system. The types of solid waste generated in the SDS will include spent cartridge filters, filter vessels and spent inorganic (zeolite media) ion exchange vessels from the LWTS, spent organic ion exchange vessels from the EPICOR-II system, spent organic ion exchange vessels from the SDS leakage containment system, and HEPA filters and charcoal adsorbers from the SDS air filtration unit. Table 12 lists these anticipated solid wastes, their forms and projected radionuclide contents and general properties, and expected ultimate disposal. It should be noted that all solid wastes will be disposed of offsite.

The most radioactive wastes, SDS prefilters, final filters, zeolite ion exchangers and leakage containment ion exchangers, will be temporarily stored in the "B" spent fuel pool under water for operator shielding protection. The off-gas system HEPA filters and charcoal adsorber are located above the pool on the off-gas system skid platform. The SDS post-filter is located above the pool on an operating platform. The EPICOR-II ion exchangers are located in the chemical cleaning building in a manner similar to the previous operation of the EPICOR-II system. During SDS process operations, the 3 liquid filters are planned to be operated based upon differential pressure across the units. When high differential pressure is encountered, the units will be flushed with processed water, disconnected, and removed from the system. The prefilter and final filter are designed to remove solids from the feed to the ion exchangers and, based on the estimated radioactivity of the sludge in the containment sump water, the radioactive loadings on these filters are expected to be low. We estimate that as many as 25 of these low activity spent filter vessels could be generated as a result of SDS operations. This estimate is based on the expected volume and radionuclide content in the sludge in the containment building sump. We estimate that there are approximately 100 ft^3 of sludge containing approximately 160 Ci of activity, primarily strontium. Based on sample results to date, there is no indication of transuranic contamination in the sludge. The specific activity of this waste would be approximately 1.6 Ci/ft^3 and is considered suitable for commercial shallow land burial.

The post-filter is provided to remove any zeolite fines that may be carried over from the upstream zeolite ion exchangers. Since the radioactive loading of the latter beds (i.e., beds 3 and 4) is expected to be low, the radioactivity of the post-filter should be correspondingly low. These filters should be suitable for disposal at commercial land burial facilities. We estimate that approximately 5 of these filters will be generated during SDS process operations.

The bulk of the radioactivity in the containment sump and RCS water will be retained in the zeolite ion exchangers. During SDS process operations, ion exchange vessels will be replaced upon reaching the administratively imposed radioactivity loading limit. Spent ion exchange vessels will be flushed with processed water, disconnected, removed from the system, and transferred to the

Table 12

SDS/EPICOR-II Solid Waste Sources For Normal Operations

Radwaste	No. Units In-Service	Waste Form	Projected Radio-Nuclide Loading	Projected Waste Generation	Container Material	Dia Ft	HT Ft	Vol. Ft	Wall Thickness in.	Expected Offsite Waste Disposition
SDS Units										
Prefilter	1	125 u cart. filters	~20 Ci	25 prefilters & final filters	304SS	2	4.5	10	3/16	CLB*
Final Filter	1	10 u cart. filters	~20 Ci		304SS	2	4.5	10	3/16	CLB
Post Filter	1	0.45 cart. filters	≪ 1 Ci	~ 5 cartridges						CLB
Zeolite Ion Exchangers	4 in. series	Mixed zeolite material	High Activity Vessels ~60,000 Ci	~10 vessels	316LSS	2	4.5	10	3/8	DOE R&D
			Low Activity Vessels <1,000 Ci Cs-137 <160 Ci Sr-90	~10 vessels						CLB
Leakage Containment ion exchangers	2 in. parallel	Organic resin	~1 Ci	~ 5	304SS	2	4.5	10	3/16	CLB
Offgas System										
Roughing filters	1	fiberglass filter media	≪1 Ci	1-3 filters						CLB
HEPA filters	2	fiberglass filter media w/frame	≪1 Ci	1-3 filters						CLB

Table 12 - Con't.

Charcoal	1	charcoal	$\ll 1$ Ci	1-3 adsorber units						CLB
EPICOR-II										
First stage	1	organic resin	<20 Ci	$\sim$20-30 vessels	coated carbon steel	6	6	120	1/4	CLB
Second stage	1	organic resin	<20 Ci	$\sim$ 5 vessels	coated carbon steel	6	6	120	1/4	CLB
Third stage	1	organic resin	<20 Ci	$\sim$5 vessels	coated carbon steel	4	4	50	1/4	CLB

*CLB = land burial at a licensed commercial disposal site.

dewatering station using the FHB crane. The crane is equipped with a yoke attached to a long shaft to mechanically prevent the inadvertent lifting of a spent vessel to a height greater than 8 feet below the pool water surface.

Based on the total inventory of activity in the containment sump and RCS water (i.e., approximately 450,000 Ci), it is theoretically possible to generate as few as 12 spent zeolite vessels. However, the SDS will be initially used to process approximately 100,000 gallons of intermediate level waste water. Additionally, the initial operation of the SDS and the corresponding loading of SDS vessels will be done slowly and cautiously and may result in initial vessel changeout prior to reaching the administrative limit. Thus, the operation of the SDS will result in the generation of high activity vessels (up to administrative limits) as well as some low activity vessels. We estimate that as many as 20 spent zeolite ion exchange vessels (10 high activity and 10 low activity) will be generated, consisting primarily of cesium and strontium radionuclides.

The high specific activity zeolite wastes are unsuitable for shallow land burial. The Department of Energy has committed to utilizing these zeolite wastes in a waste immobilization research and development program, as described in Reference 26. Immobilized wastes will not be returned to the TMI site, as DOE plans on confidence testing and extended observation of these wastes, along with possible other uses in various DOE research, development and testing programs. The DOE waste immobilization research and development program is part of the Department's fiscal year 1982 funding request to Congress.

The low activity wastes (containing less than 1000 Ci of Cs 137 and 160 Ci of Sr-90) are considered suitable for disposal at a commercial shallow land burial facility and will be buried in accordance with state and federal requirements. The operation of EPICOR-II to polish SDS effluent will generate additional low activity waste. Based on the results of ion exchange column tests using actual samples of containment sump water, we estimate that as many as 40 polisher vessels may be generated containing up to 20 Ci each. These vessels will be shipped to a commercial land burial facility for disposal. During SDS process operations, small quantities of contaminated water may leak into the process vessel containment boxes to be collected and processed in the leakage containment system ion exchangers. Based on the fact that system leakage can be quickly identified (e.g., by the system in-line radiation monitor and periodic sampling) and corrected, we estimated approximately 5 low level ion exchangers to be generated during operation of the SDS. Based on the anticipated loading of less than 1 Ci, this waste should be suitable for disposal at a commercial land burial facility.

Wastes from operation of the GWTS air filtration unit are expected to be similar to those wastes generated in other plant ventilation filtration systems. These wastes should be suitable for disposal at commercial land disposal facilities.

During process operations, spent filter and ion exchange vessels will be flushed with processed water and then disconnected, removed from the system, and transferred to the dewatering station using the FHB crane. The dewatering station consists of two underwater boxes, one for filter vessels and one for ion exchange vessels, and associated valves, piping and instrumentation to effect dewatering using station compressed air. The dewatering involves passing a constant flow of air through the vessel. Following dewatering, the spent vessels will be capped to minimize the possibility of inleakage of pool water back into the vessel. Each vessel is designed with integral check valves in the vessel nozzles to prevent leakage. From the dewatering station, the spent vessels are transferred to the pool storage racks for temporary storage pending offsite shipment for subsequent processing. Stored vessels will be continuously vented to a common header which interfaces with the GWTS air filtration unit to prevent the buildup of pressure within the vessels due to radiolysis of contained moisture. The pool storage racks can accommodate 60 spent filter and ion exchange vessels. We estimate that approximately 5 leakage containment system vessels, 25 SDS filter vessels, and 20 SDS zeolite vessels will be generated during the processing of the containment sump and RCS water. We conclude that the provisions for temporary storage of spent vessels are adequate to meet the processing needs of the SDS.

We have evaluated the potential for vessel corrosion and ion exchange media degradation while in temporary storage. This storage period may last as long as 10 years although we anticipate a much shorter time period since shipping casks for the vessels currently exist* and arrangements are currently being planned for waste immobilization at one or more offsite government facilities as part of DOE's R&D programs. The SDS ion exchange vessels are fabricated from 316L stainless steel with 3/8 inch thick walls and have a design rating of 350 psig to prevent pressurization failure of a closed vessel. The use of low-carbon austenitic stainless steel will provide corrosion resistance. Additionally, dewatering of the vessels following removal from service will reduce the means for transporting corrosive species, should any be generated, to liner walls. A number of synthetic and naturally occurring zeolites have been irradiated to exposures in the range 10^{10} to 4×10^{11} rads without any structural damage or decreased affinity of sorbed cesium or strontium (Reference 23). Based on extensive Department of Energy plant-scale experience, SDS vessels containing 60,000 Ci of cesium and strontium are expected to maintain integrity while being exposed to ionizing radiation for storage times at least as long as 10 years (Reference 23). Calculated radiation exposure of SDS liners loaded with 60,000 Ci of activity after 10 years of storage is estimated to be about 1.7×10^{11} rads which is well within existing experience for integrated exposure. While in storage, any gases that are generated will be removed by the installed vent system.

*However, NRC license has not been requested at this time, but is required for these casks. The cask planned for use by the licensee requires modification and license certification for this use.

The other much lower loaded SDS solid wastes, the prefilter and final filter and the leakage containment ion exchangers are fabricated from 304 stainless steel and they will also be dewatered. Based on the much reduced radiological loading and the use of corrosion resistant stainless steel for the vessels, we do not anticipate ion exchange media degradation or significant vessel corrosion. These units will be temporarily stored in the "B" fuel pool pending disposal offsite.

The packaging of the zeolite waste for offsite shipment involves loading the spent vessel into the shipping cask which is placed on an underwater cask platform. At no time during the operation of the SDS will a spent ion exchange or filter vessel be removed from the pool before it has been placed in a cask. When the cask is loaded with a spent vessel, the FHB crane will be used to transfer the cask to the pool surface for draining prior to shipment. The spent ion exchange vessels will be transported to one or more federal government facilities for utilization in the DOE research, development and test programs as discussed above.

All waste will be packaged in accordance with the requirements of 10 CFR Part 20, 10 CFR Part 71 and 40 CFR Parts 171-179 and shipped in accordance with Nuclear Regulatory Commission and Department of Transportation regulations. The wastes shipped to a commercial shallow land burial facility will be disposed of in accordance with federal and state requirements.

3.5 Process and Effluent Radiological Monitoring and Sampling

The process and effluent radiological monitoring and sampling system for the SDS is designed to provide information regarding radioactivity levels in process streams, indicate radioactive leakage between systems, monitor equipment performance and monitor discharges to the fuel handling building ventilation system.

Gaseous and liquid streams will be monitored. Table 13 indicates the locations of SDS process monitors and whether they perform a control function. Continuous in-line liquid monitors are located on the effluent from the LWTS guard bed (i.e. bed 4) and the effluent from the SDS leakage containment boxes. A high radiation signal from either of these monitors will automatically shut the inlet control valve to the LWTS ion exchangers stopping the further operation of the system. Additionally, the beta monitoring system can sequentially provide continuous indication of gross beta levels in process water effluent from each LWTS ion exchanger (beds 1, 2, 3, 4). Continuous gas monitors are provided for the influent and the effluent of the GWTS air filtration unit. The influent monitor provides a measure of gross gamma activity and the effluent monitor measures particulate, iodine and noble gas activity.

Table 13

SDS Process Monitors

System Radiation Monitors	Dector	Function[c]
IX Manifold General Area	G-M[a]	A,C
Off-Gas Header Effluent	G-M[a]	A
Leak Containment System Influent	G-M[b]	A,C
Train #1 A,B,C Effluents[d]	G-M[b]	A,I
Train #2 A,B,C Effluents[d]	G-M[b]	A,I
LWTS Bed 4 Effluent	G-M[b]	A,C
Off-Gas Particulates	Scintillation[b]	A,I
Off-Gas Gases	Ion Chamber[b]	A,I
Off-Gas Charcoal	Scintillation[a]	A,I

[a] Gamma Detector

[b] Beta Detector

[c] A-alarm, I-indication, C-control

[d] The effluent is monitored after passing thru each bed.

In addition to the liquid process monitor locations described above, the SDS design incorporates the capability for sampling numerous other locations in the SDS system to support process operations with detailed isotopic analyses (e.g., Sr-90 concentration) and to provide estimates of ion exchanger curie loadings. Process stream water is sampled at each stage of treatment using three shielded sample glove boxes located in the areas adjacent to the "B" spent fuel pool. These glove boxes are the high rad filter sample glove box, the high rad feed sample glove box, and the intermediate level sample glove box. The high rad filter sample glove box samples the influent to the prefilter and the effluent of the final filter. The high rad feed sample glove box samples the influent to the first zeolite ion exchanger. The intermediate level sample glove box samples the effluent from each LWTS ion exchanger in the three vessel parallel trains (beds 1, 2, and 3) and the influent and effluent of the parallel guard bed ion exchangers (bed 4). Additionally, local samples can be drawn from the influent and effluent of the leakage containment ion exchangers and from the two SDS monitor tanks. The sampling locations described above will provide representative liquid samples required to effectively monitor the operation of the SDS and control the levels of radioactivity deposited in process components.

3.6 Conformance With NRC Regulatory Guides, Criteria and Regulations

The SDS is installed and operated within the TMI-2 fuel handling building. The fuel handling building is a seismic Category I building designed to withstand the effects of natural phenomena (e.g., winds, tornadoes, floods and earthquakes) which may occur at the site. The design basis for the fuel handling building exceeds our recommended criteria, presented in Regulatory Guide 1.143, for buildings housing radwaste systems. We conclude that the fuel handling building is an acceptable structure for housing the SDS.

We have evaluated the capability of the SDS to keep the levels of radioactivity in gaseous effluents "as low as is reasonably achievable", in accordance with 10 CFR Part 20, the existing Appendix B Technical Specifications, and the criteria specified in Appendix R of the Final PEIS. There are no liquid discharges of radioactive material from the SDS and no direct gaseous discharges, however, the SDS interfaces directly with the FHBVS and is housed in the TMI-2 fuel handling building which is serviced by the FHBVS. The FHBVS discharges to the environment via the plant stack. Thus, there are two potential sources of radioactive gaseous influents to the FHBVS resulting from operation of the SDS. The first source is the direct discharge from the SDS (see Section 3.3) to the FHBVS. The second postulated source is potential leakage into the fuel handling building from SDS piping and valves which are not submerged in the fuel pool and from evaporation of tritiated water from the pool. Any leakage from SDS piping and valves which are submerged in the pool will mix with pool water in the underwater containment boxes and be processed by the leakage containment ion exchangers.

Regarding the discharge from the SDS, we conservatively estimated in the PEIS the radioactive gaseous effluents to the environment which would result from the processing of containment sump and RCS water. The principal radionuclides in these releases are listed in Table 14. The dose estimates for the maximum exposed individual which pertain to these releases are provided in Table 15. Our release model conservatively assumed that 1×10^{-4} of the curie inventory in the influent liquid waste stream to the processing system (i.e., the SDS for this case) would become aerosolized during processing. Our model further assumed that this gaseous waste stream is carried by subsequent movement of air in the process vent system, then ultimately combined with building ventilation air (i.e., the FHBVS for this case), treated by the air cleaning system for the building, monitored by the radiation detectors in the TMI-2 plant stack, and finally released to the environment by controlled discharge. Our PEIS model did not assume that there was an air filtration system in the liquid waste processing system, and, thus, no credit was given for the SDS air filtration unit in estimating the gaseous effluents to the environment.

Based on the filtration capability of the SDS and the availability of more recent information on the radionuclide distribution in the containment sump and RCS water, we have reevaluated our estimates of radioactive gaseous effluents to the environment resulting from waste water processing in the SDS. The latest estimates of the radionuclide distributions in the containment sump and RCS water are provided in Tables 1 and 2, respectively. Utilizing the PEIS model for the partitioning of radioactivity, we have assumed that 1×10^{-4} of the curie inventories listed in Tables 1 and 2 become aerosolized during SDS process operations. Further, we have assumed that this gaseous waste stream is carried by the movement of air in the GWTS of the SDS and treated by the GWTS air filtration unit. For conservatism, we have assumed an overall particulate decontamination factor (DF) of only 10 (i.e., a removal efficiency of 90%) for both of the unit HEPA filters (i.e., 2 in series) despite the fact that they are each in-place tested to confirm a removal efficiency greater than 99.95%. The gaseous effluent from the GWTS air filtration unit is directed to the FHBVS and treated by the ventilation system HEPA filters. Utilizing the PEIS model for particulate removal in ventilation air filtration systems, we have assumed an overall particulate DF of 1000 for the FHBVS. Based on the source term model, as described above, the principal radionuclides in gaseous effluents resulting from treatment of containment sump and RCS water are listed in Table 16. The dose estimates for the maximum exposed individual to these effluents are provided in Table 17 and are noted to be considerably smaller in every case than the corresponding PEIS values in Table 15.

TABLE 14

PEIS Estimates Of Gaseous Effluents
To The Environment Attributable
To Aerosol Formation From Processing of
Containment Sump And RCS Water

Nuclide	Containment Building Sump (Total Curies)	Reactor Coolant System (Total Curies)
Cs-134	6.4×10^{-3}	1.5×10^{-4}
Cs-137	4.1×10^{-2}	9.9×10^{-4}
Sr-90	6.9×10^{-4}	7.8×10^{-4}
H-3	2.5×10^{-1}	2.7×10^{-3}

TABLE 15

Dose Estimates[a] For The Maximum Exposed Individual
Due to Processing of Containment Sump and RCS Water

Location	Pathway	Dose (mrem)[b] Total-Body	Bone
Nearest garden	Inhalation	8.9×10^{-4}	1.0×10^{-2}
	Ground Shine	4.8×10^{-3}	4.8×10^{-3}
	Vegetable Use	6.2×10^{-2}	2.8×10^{-1}
	Total	6.8×10^{-2}	3.0×10^{-1}
Nearest milk goat	Inhalation	1.6×10^{-3}	4.3×10^{-3}
	Ground Shine	4.6×10^{-3}	4.6×10^{-3}
	Goat Milk Use	7.8×10^{-2}	5.4×10^{-1}
	Total	8.4×10^{-2}	5.5×10^{-1}
Nearest cow and garden	Inhalation	9.8×10^{-4}	1.1×10^{-2}
	Ground Shine	7.0×10^{-3}	7.0×10^{-3}
	Vegetable Use	9.0×10^{-2}	4.1×10^{-1}
	Cow Milk Use	2.3×10^{-2}	1.3×10^{-1}
	Total	1.2×10^{-1}	5.6×10^{-1}

[a]Based on the PEIS estimates of the gaseous effluents listed in Table 14.

[b]Total-body and maximum organ (bone) doses are listed. Doses were calculated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group considered are listed.

TABLE 16

Current Estimates of Gaseous Effluents To The Environment Resulting From Treatment of Containment Sump and RCS Water In The SDS

Nuclide	Containment Building Sump (Total Curies)	Reactor Coolant System (Total Curies)
Cs-134	5.0×10^{-4}	1.0×10^{-5}
Cs-137	3.7×10^{-3}	8.0×10^{-5}
Sr-90	1.3×10^{-4}	7.0×10^{-5}
H-3	2.0×10^{-1}	2.2×10^{-3}

TABLE 17

Dose Estimates [a] for the Maximum Exposed Individual
Due to Processing of Reactor Building Sump and RCS Water

Location	Parkway	Dose (mrem)[b] Total-Body	Bone
Nearest Garden	Inhalation	1.2×10^{-4}	3.1×10^{-4}
	Ground Shine	4.2×10^{-4}	4.2×10^{-4}
	Vegetable Use	7.7×10^{-3}	9.5×10^{-3}
	Total	8.2×10^{-3}	1.4×10^{-2}
Nearest Milk Goat	Inhalation	1.7×10^{-4}	1.0×10^{-4}
	Ground Shine	4.0×10^{-4}	4.0×10^{-4}
	Goat Milk Use	7.0×10^{-3}	5.1×10^{-2}
	Total	7.6×10^{-3}	5.2×10^{-2}
Nearest Cow and Garden	Inhalation	1.1×10^{-4}	1.4×10^{-3}
	Ground Shine	6.1×10^{-4}	6.1×10^{-4}
	Vegetable Use	1.1×10^{-2}	5.0×10^{-2}
	Cow Milk Use	2.2×10^{-3}	1.3×10^{-2}
	Total	1.4×10^{-2}	6.5×10^{-2}

[a] Based on the radionuclides listed in Table 16.

[b] Total-body and maximum organ (bone) doses are listed. Doses were calculated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group considered are listed.

We have also considered the potential for direct leakage from SDS piping and valves which are not submerged in the "B" spent fuel pool. This postulated leakage was evaluated to determine if there is a significant potential for generating airborne contamination in the fuel handling building. The SDS piping has been constructed with a design pressure of 150 psig and successfully pressure tested at 120% to 150% of its design pressure (i.e., tested at 180 to 225 psig). On the other hand, the SDS is a relatively low pressure system which operates at ambient temperature with a normal operating pressure of approximately 75 psig and the likelihood for leakage is correspondingly minimized at the low operating pressures. Further, the principal valves in the SDS system that are utilized to control process operations are housed in containment manifolds (i.e., the RCS cleanup manifold, the high-rad filter manifold, the feed pump manifold, the ion exchanger manifold, and the beta monitoring manifold) which are maintained at negative pressure by the SDS GWTS. Thus, any leakage which becomes airborne in the valve containment manifolds would be processed in the GWTS and this source of activity has been adequately bounded by our conservative assumption of the fraction of the influent activity (i.e., 1×10^{-4}) which becomes airborne during SDS process operations. We conclude that there is no significant potential for the generation of airborne activity in the atmosphere of the fuel handling building from the leakage of ex-pool SDS piping and valves with the design provisions for system integrity and leak-off collection.

We have also considered the potential impact of the evaporation of processed tritiated water from the "B" spent fuel pool. The "B" spent fuel pool has been filled with 240,000 gallons of processed water which has a tritium concentration of approximately 0.15 uCi/ml. This volume corresponds to a tritium inventory of 136 Ci. In the PEIS (Reference 19) we estimated the annual rate of evaporation of tritiated water from the plant resulting from the use of processed accident generated water to shield a submerged ion exchange processing system in the "B" spent fuel pool. Based on an average tritium concentration of 0.13 uCi/ml in the existing inventory (i.e., 740,000 gallons) of processed water in onsite storage, we estimated the rate of tritium evaporation and subsequent loss to the environment to be 50 Ci/year. Based on this release rate, we have estimated the doses to the maximum exposed individual as shown in Table 18.

More recently, we have reestimated the rate of evaporation of tritiated water from the "B" spent fuel pool based on the pool tritium concentration of 0.15 uCi/ml and a more detailed engineering model (Reference 28) for air flow induced evaporation from the pool. We have estimated the rate of tritium loss to the environment to be approximately 9 Ci/year. The estimated doses to the maximum exposed individual from the release are shown in Table 19. The doses to the maximum exposed individual resulting from waste water processing in the SDS and evaporation from the spent fuel pool are summarized in Table 20.

Table 18

Dose Estimates[a] To The Maximum Exposed Individual For The Release of 50 Ci/year Of Tritiated Water Vapor From The B Spent Fuel Pool

Location	Pathway	Dose (mrem) Total-Body
Nearest Garden	Inhalation	3.4×10^{-3}
	Ground Shine	0
	Vegetable Use	1.2×10^{-2}
	Total	1.6×10^{-2}
Nearest Milk Goat	Inhalation	1.7×10^{-3}
	Ground Shine	0
	Goat Milk Use	1.4×10^{-2}
	Total	1.5×10^{-2}
Nearest Cow and Garden	Inhalation	3.8×10^{-3}
	Ground Shine	0
	Vegetable Use	1.3×10^{-2}
	Cow Milk Use	5.3×10^{-3}
	Total	2.2×10^{-2}

[a]Based on estimates in the PEIS for evaporation of tritiated water from the spent fuel pool.

TABLE 19

Dose Estimates[a] To The Maximum Exposed Individual For The Release of 9 Ci/Year Of Tritiated Water Vapor From The B Spent Fuel Pool

Location	Pathway	Dose (mrem) Total-Body
Nearest Garden	Inhalation	6.2×10^{-4}
	Ground Shine	0
	Vegetable Use	2.2×10^{-3}
	Total	2.8×10^{-3}
Nearest Milk Goat	Inhalation	3.1×10^{-4}
	Ground Shine	0
	Goat Milk Use	2.5×10^{-3}
	Total	2.8×10^{-3}
Nearest Cow and Garden	Inhalation	6.8×10^{-4}
	Ground Shine	0
	Vegetable Use	2.4×10^{-3}
	Cow Milk Use	9.6×10^{-4}
	Total	4.0×10^{-3}

[a]Based on the releases predicted from the engineering model (Reference 28) for evaporation of tritiated water vapor from the B spent fuel pool.

TABLE 20

Dose Estimates [a] for the Maximum Exposed Individual
Due to Processing of Reactor Building Sump and
RCS Water and Associated Evaporation of Tritiated Water
from the Spent Fuel Pool

Location	Pathway	Dose (mrem) Total-Body	Bone
Nearest Garden	Inhalation	7.4×10^{-4}	3.1×10^{-4}
	Ground Shine	4.2×10^{-4}	4.2×10^{-4}
	Vegetable Use	9.9×10^{-3}	9.5×10^{-3}
	Total	1.1×10^{-2}	1.4×10^{-2}
Nearest Milk Goat	Inhalation	4.8×10^{-4}	1.0×10^{-4}
	Ground Shine	4.0×10^{-4}	4.0×10^{-4}
	Goat Milk Use	9.5×10^{-3}	5.1×10^{-2}
	Total	1.0×10^{-2}	5.2×10^{-2}
Nearest Cow and Garden	Inhalation	8.1×10^{-4}	1.4×10^{-3}
	Ground Shine	6.1×10^{-4}	6.1×10^{-4}
	Vegetable Use	1.4×10^{-2}	5.0×10^{-2}
	Cow Milk Use	3.1×10^{-3}	1.3×10^{-2}
	Total	1.8×10^{-2}	6.5×10^{-2}

[a] Based on the discharge of 9 Ci/year of tritium and the radionuclides listed in Table 16.

Based on this evaluation, we conclude that the SDS is capable of keeping the levels of radioactivity in gaseous effluents "as low as is reasonably achievable", in accordnace with 10 CFR Part 20, the existing Appendix B Technical Specifications, and the limits specified in Appendix B of the PEIS (Reference 19).

We have evaluated the capability of the SDS to maintain releases of radioactive materials in gaseous effluents to unrestricted areas below the limits in 10 CFR Part 20, Appendix B, Table II, Column 1, during periods of process operations at the maximum flow rate in the SDS. The existing Appendix B Technical Specifications implement the requirements of 10 CFR Part 20. The delivery and filtration of containment sump water to the upper tank farm will occur at a flow rate varying from 10 to 30 gpm. Based on the maximum flow rate of 30 gpm, a partitioning of 1×10^{-4} of the influent activity to the SDS gaseous waste treatment system, a GWTS particulate decontamination factor of 10, a GWTS discharge flow rate of 1000 cfm, a FHBVS particulate decontamination factor of 1000, a FHBVS flow rate of 40,000 cfm, and a total plant stack flow rate of 100,000 cfm, we calculated the instantaneous release rate (uCi/sec) of the principal radionuclides discharged to the environs for comparison with the limits in the Appendix B Technical Specifications. The calculated release rate (0.003 uCi/sec for particulates and 0.3 uCi/sec for tritium) is well within the Technical Specification limits (0.3 uCi/sec for particulates and 30,000 uCi/sec for tritium) and, therefore, we conclude that the SDS is capable of maintaining releases of radioactive materials in gaseous effluents to unrestricted areas below the limits in 10 CFR Part 20, Appendix B, Table II, Column 1.

We have evaluated the quality group classification of the SDS to ensure that the SDS has been designed, constructed, installed and will be operated on a level commensurate with the need to protect the health and safety of the public and plant operating personnel. Specifically, we have evaluated the design criteria (i.e., the codes and standards for piping, valves and vessels) for the SDS, the provisions for monitoring tank liquid levels, the provisions for controlling liquid leakage or spillage, and the quality assurance program for the design, construction, and testing of the SDS with the criteria recommended in Regulatory Guide 1.143 and conclude that the provisions in the SDS meet our regulatory criteria. There are only two tank systems in the SDS design which are used during process operations. These are the 60,000 gallon upper feed tank system and the 590 gallon off-gas separator tank system. Both tank systems are equipped with level instrumentation to monitor liquid levels and the instrumentation will actuate alarms on tank high level. Additionally, a valve in the feed line to the upper tank farm will close on a high level signal from the upper tank farm standpipe to prevent tank overflow. Similarly, the off-gas bottoms pump in the off-gas separator tank system will start to pump on high level in the off-gas separator tank in order to prevent tank overflow. The off-gas bottoms pump flow is back to the upper tank farm system.

We have also evaluated the licensee's quality assurance program for the SDS to ensure that all design, construction, and testing provisions have been met and are documented. We conclude that the licensee's quality assurance program is in accordance with the recommendations of Regulatory Guide 1.143 and the overall quality group classification for the SDS is in accordance with the criteria set forth in Regulatory Guide 1.143 and is, therefore, acceptable.

We have evaluated the licensee's onsite program for the management of radioactive solid waste to ensure that all of the waste generated from the processing of the containment sump and RCS water will be packaged and transported in accordance with Federal requirements. The licensee's program includes a detailed inspection of all solid waste which is packaged for offsite shipment for subsequent processing or disposal to ensure compliance with Federal requirements. Additionally, for all of the radioactive solid waste which is packaged onsite for shipment offsite for subsequent processing or disposal, including the waste anticipated to be generated during SDS and EPICOR-II process operations, the onsite NRC staff will inspect the solid waste containers, shipping casks, and methods of packaging to ensure that all wastes are packaged and transported in accordance with applicable Commission and Department of Transportation regulations. We conclude that all such shipments will be packaged and transported in accordance with Federal requirements.

We have evaluated the provisions for temporary onsite storage of SDS and EPICOR-II solid waste prior to shipment for subsequent processing or disposal. Spent SDS filter and ion exchange vessels will be stored in submerged racks in the "B" spent fuel pool. The storage pool is designed to accommodate 60 SDS vessels. We have estimated that up to 50 vessels total requiring pool storage will be generated as a result of SDS process operations and conclude that the provisions for onsite storage are in accordance with the criteria in Branch Technical Position-ETSB 11-3 (Revision 1). For the EPICOR-II low level solid waste, there exist onsite two engineered storage modules with 60 cells each for the storage of EPICOR-II vessels. Based on the volume of solid waste currently in storage, a single module should be available and one module is capable of accommodating all of the waste anticipated to be generated as a result of the use of EPICOR-II as a polishing system. We conclude that the provisions for storage of EPICOR-II waste are in accordance with the criteria in Branch Technical Position-ETSB 11-3 (Revision 1).

We have evaluated the capability of the SDS to monitor the performance of process equipment and detect radioactive material leakage between systems. Process monitoring in the SDS is accomplished with the system continuous monitors and with supportive sampling. Continuous in-line monitoring is provided for the effluent from the SDS (i.e., the flow from the SDS guard bed) and the beta monitoring system can sequentially provide continuous indication of gross beta levels in the process water effluent from each SDS ion exchange vessel in the process train. Additionally, sampling capability is provided for the influent to the prefilter and the effluent from the final filter to indicate the amount of suspended solids in the influent to the SDS ion exchangers. Sampling capability is also provided for the influent and effluent of each ion exchanger in the process train to indicate the degree of decontamination provided by each vessel. For gaseous effluents from the SDS, the GWTS is provided with continuous monitors for the influent and effluent of the GWTS air filtration unit to indicate the degree of particulate removal in the system. Thus, we conclude that the SDS capability for monitoring the performance of process equipment is adequate.

Potential liquid leakage from the submerged portions of the SDS during process operations and vessel changeout is collected and processed in the leakage containment system. An indication of inter-system leakage (i.e., from the SDS to the pool water) is provided by the in-line monitor on the influent to the leakage containment ion exchangers. The leakage containment system has the capability of isolating potential sources of leakage (e.g., each SDS ion exchanger vessel containment box) in the SDS so that the condition can be corrected. We conclude that the SDS has the capability of indicating radioactive material leakage so that upset conditions can be identified and corrected.

3.7 Pre-operational Testing

The Submerged Demineralizer System components, i.e., valves, pumps, piping, and vessels have been preoperationally tested according to the recommendations in Regulatory Guide 1.143 which is discussed in the design criteria section of this safety evaluation report. The piping system has been designed and fabricated as specified in the ANSI B31.1 Power Piping Code and pressure tested at 1.2 to 1.5 times its design pressure of 150 psig (180 to 225 psig) for a minimum time period of 10 minutes. Vessels were designed and fabricated in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section 8, Division 1, and were successfully hydro tested at 1.5 times their design pressure for a minimum period of 30 minutes. Flow rates and pressure differentials were also verified for instrumentation and alarm setpoint adjustments. After the system was completely assembled, it was successfully tested as a unit at 1.1 times design pressure. The "1.1" value was used to prevent exceeding the design pressure of miscellaneous pumps and valves in the system.

3.8 Operator Staffing and Training

The SDS will be staffed at a minimum with two system operators to monitor system operations, to perform required valve manipulations, spent vessel dewatering, ion exchanger changeouts, and equipment start/stop operations. One chemistry technician will be on shift to support SDS operations. He will obtain samples as required by the process control program or as directed by the process support group. One health physics technician will be dedicated to SDS operational support. He will be responsible for monitoring general area radiation levels, obtaining swipe samples, and obtaining airborne particulate and tritium samples. Engineering support will be provided by on-call engineers from the Process Support Group and Recovery Engineering. The minimum staffing as stated above will be augmented as required to support safe SDS operation. During initial SDS operations on shift engineering, chemistry technician and health physics technician support will be augmented.

Before any individual will be allowed to operate or supervise the Submerged Demineralizer System (SDS), a formal operator training program must be completed. The training program consists of self study, classroom instruction and practical hands-on experience. Prospective operators will be chosen from on-shift control room operators and auxiliary operators. These operators have completed the training and experience requirements for their respective positions as control room operators and auxiliary operators. The self study program is initiated by providing the prospective operators with a system description document, flow path diagrams, operating procedures, emergency procedures and response-to-alarm procedures. These items will allow the prospective operators to familiarize themselves with the components of the SDS and the evolutions that will be performed using the SDS. The prospective operators will be required to walk through the system to familiarize themselves with the location and features of the installed equipment. The self study program provides background information so that the classroom instruction and the hands-on instruction can be more effective in training the operators to operate the SDS.

The classroom instruction on SDS operation is provided as part of the shift's regularly scheduled training. Those required to attend include auxiliary operators and supervisors as well as control room operators. The classroom instruction consists of a minimum of eight hours of SDS instruction. The hands-on training consists of actual performance of operating evolutions under the instruction of an individual designated by the Director of Plant Operations. The evolutions that are required to be performed are system startup, operation, and shutdown; vessel changeout; process trains shifting; spent vessel tool operation; and simulation of casualty actions. Performance of the evolution will be documented by the designated individual monitoring the prospective operator. To verify that the operator has adequate knowledge and practical experience to operate the SDS, each operator is required to successfully complete a written and an oral examination. The examinations will be administered by individuals designated by the Director of Plant Operations. The written examinations will include questions to evaluate knowledge of the SDS system components, instrumentation, valves, automatic valves, operating procedures, casualty procedures and chemistry and health physics concerns as they pertain to the SDS system.

An audit of the training program will be performed by the onsite NRC staff to verify that it is carried out as planned. The audit will include monitoring some of the actual instruction and questioning the operators to evaluate the effectiveness of the training program. Subject to a satisfactory outcome of the audit, we conclude that the proposed operator training program for the SDS is acceptable.

3.9 Decommissioning

There are a number of available options for safely decommissioning the Submerged Demineralizer System. Decommissioning includes dismantling and removal of the system from its present location in the spent fuel pool. Independent of the method chosen, the SDS equipment would be flushed prior to any disassembly or removal of equipment. The decontamination factor desired as a result of the flushing would be directly dependent on which option is selected for the next decommissioning step. Underwater or above water decommissioning techniques are available. Any mode selected would incorporate ALARA considerations. A discussion of the occupational exposure resulting from decommissioning the SDS is provided in Section 7.

Disassembly Options. The options discussed in this evaluation are not all that might be available to the licensee but are those most commonly used in current industry applications.

Arc Saw Cutting is an extension of nonconsumable electrode technology and when used underwater provides a smooth uniform cut without generating significant amounts of smoke or noise. Cutting in air lacks these advantages and is therefore not as desirable in this process. Since there is no metal to metal contact, reaction forces are small, therefore the workpiece does not require rigid clamping. The arc saw is capable of cutting any electrical conducting material. High conductivity materials such as stainless steels, high alloy steels, aluminum and copper could be cut rapidly and cleanly. However, carbon steels are not as easily dismantled.

Plasma arc cutting is a process capable of cutting all metals. In air use, it will penetrate thicker sections than under water. Manually handled torch operation cannot typically be used for materials greater than 1/12 inches in thickness, although because of its adaptability to automated operation, its capabilities can be extended.

Oxygen burning (oxyacetylene cutting) is another option available. A hand held torch is the general method of usage, however, it too can be automated. Ordinarily this method is unable to cut stainless steels, aluminum, and other non-ferrous or ferrous-high percent alloy metals. The most effective use is in cutting carbon steels. This method is commonly used in dismantling beams, columns and supports. In-air or underwater application is effective. Thermite reaction lances are capable of cutting any metal and are capable of being used in air or underwater. This process is well suited for cutting irregular surfaces with minimum access; however, significant amounts of smoke are generated and its use is limited to lower level contaminated equipment and well ventilated and filtered areas. Remote cutting with power nibblers and shears operates on the same basis as a conventional pair of scissors. Heavy duty power nibblers and shears attached to long support tubes can be utilized for remote cutting mild steel and stainless steel components in the system.

Independent of the methodology selected by the licensee, portable shielding in a form appropriate for each application can be used upon removal of the equipment from the fuel pools. The equipment, if removed in a dismantled form, will be put in appropriate storage/shipping containers depending on activity levels. If the decision is made by the licensee to dismantle selected equipment out of water, appropriate ventilation tents would be installed to preclude the spread of airborne contaminants. Selected vessels may even be filled with a concrete mixture while in the pool for self shielding purposes and then removed. Whatever method, or combination of methods, is chosen by the licensee, we will review any associated procedures before the operation is begun.

4. RADIATION PROTECTION

4.1 Summary Description

As part of the licensing process for Three Mile Island, Unit 2, the original radiation protection program was described in Section 12.0 of the TMI Unit 2 Final Safety Analysis Report (Reference 25). After the accident at TMI in March 1979, several changes to the radiation protection program were made. These changes, which were incorporated to reflect the unique post-accident radiological environment at TMI-2, are described in the licensee's Radiological Controls Program Management Plan, submitted to the NRC on February 8, 1980 (Reference 29), and in the quarterly status reports on the implementation of the above program. Additional information is found in the licensee's Radiation Protection Plan of January 23, 1981 (Reference 30).

The licensee's Technical Evaluation Report (Reference 2) and System Description (Reference 27) for SDS included information on SDS layout and equipment design, operating procedures and techniques, and practices proposed for the protection of personnel against radiation. Personnel will be protected by shielding to reduce levels of radiation, ventilation arranged to control the flow of potentially contaminated air, and radiation monitoring systems employed to measure levels of radiation in potentially occupied areas and to measure airborne radioactivity throughout the plant. A health physics program will be provided for plant personnel during SDS operation, maintenance, radwaste handling, and inservice inspection.

We reviewed and evaluated the licensee's description and analysis of the radiation protection program. On the basis of our review, we conclude that the radiation protection program will provide reasonable assurance that doses to personnel will be less than those established by 10 CFR Part 20, that design features and program features are consistent with the ALARA (As Low As Is Reasonably Achievable) criteria of Section 20.1 (c) of 10 CFR Part 20, and that the program is acceptable for the planned operation of the SDS. Details are discussed in the following sections.

4.2 Shielding

The design objectives for SDS shielding are to ensure that radiation exposure rates to operating personnel will be within the required limits of 10 CFR Part 20 and that these exposure rates will also be maintained at an average level of less than one milliroentgen per hour.

The licensee has included many features in the design and layout of SDS equipment and facility shielding to maintain radiation exposures as low as is reasonably achievable. These features include the location of major sources of radiation underwater, the use of labyrinths to eliminate direct shine to accessible areas, the use of shielded valve stations, and provisions for adequate flushing of system piping. These features contribute significantly to minimizing radiation exposures to personnel who will operate the system.

Principal radiation sources in the SDS and associated systems include the feed tanks, piping from the reactor building to the feed tanks, and the ion exchange vessels. The feed tanks are shielded with three feet of concrete. The piping from the reactor building is shielded with a minimum of 2½ inches of lead. Valve containment boxes are shielded with a minimum of 20½ inches of concrete. The ion exchange vessels will be shielded by greater than 12 feet of water during processing and greater than 8 feet of water during vessel changeout.

We find that the licensee has designed the facility to keep radiation exposures within the applicable limits of 10 CFR Part 20. The design and arrangement features have been included to reduce unnecessary exposure during operations. Based on our review, we conclude that the shielding and arrangement of the facility is acceptable.

4.3 Area Monitoring

The licensee described the area radiation monitoring system in the SDS System Description and in a response letter (Reference 18). Area radiation monitors will be placed at the "A" spent fuel pool and at the north end of the "B" spent fuel pool between the cask washing pit and the surge pit. If radiation levels exceed the alarm setpoint on either of these monitors, the radiation monitor alarm is actuated. A trip signal from the alarm at the "B" pool will also close the feed isolation valve.

There are three continuous air monitors installed in the vicinity of the SDS: (1) north of the "B" spent fuel pool; (2) at the "A" spent fuel pool; and (3) at the valve containment box at southwest end of "A" pool. All continuous air monitors alarm locally.

The licensee has portable tritium samplers installed in the vicinity of the "B" spent fuel pool. Grab samples will be taken, analyzed, and results made available to the licensee's radiation protection personnel on a timely basis to permit appropriate operational analysis.

On the basis of the locations chosen, the sensitivities, the alarm settings (which are set to alarm at radiation levels which are low enough to ensure worker dose minimization), and the calibration program, we have concluded that there is reasonable assurance that radiation levels within the plant will be adequately monitored and that the area radiation monitoring system is acceptable.

4.4 Ventilation

During operation of the SDS the plant's ventilation system will be operated to maintain a suitable environment for personnel and equipment in the Fuel Handling Building. The SDS is also provided with a separate offgas system that has been interfaced with the plant's ventilation system. The objectives of the offgas and plant ventilation systems are the protection of operating personnel from possible airborne radioactivity and to provide assurance that the maximum expected airborne radioactivity concentrations will be maintained within the limits of 10 CFR Part 20 and as low as reasonably achievable. We have determined (Reference 28) that the FHBVS is capable of maintaining concentrations of airborne tritium in the FHB below the limits in 10 CFR Part 20 even with concentrations of tritiated water in the fuel pool as high as 1.0 uCi/ml. With tritiated water at a concentration 0.15 uCi/ml currently in the fuel pool, grab samples indicate airborne tritium concentrations at approximately 1×10^{-8} uCi/ml. This value is well below the Part 20 limits of 5×10^{-6} uCi/ml.

To meet these objectives, several design features are used including: (1) ventilation air flow from areas of least radioactive contamination to areas of progressively greater radioactive contamination and exhaust to ventilation filters, (2) maintenance of slight negative pressures in selected areas, and (3) selection of air-flow rates to maintain airborne concentrations of radioactive material below the limits of 10 CFR Part 20. Additionally, the ventilation system contains a quick access filter housing design feature to permit changing of the filter in a minimum amount of time and worker exposure with minimum probability for spread of radioactive contamination.

Based on our review, we conclude that the offgas ventilation system, as described in the System Description for SDS, and the Fuel Handling Building Ventilation System, as described in Sections 9.4 and 12.2 of the Final Safety Analysis Report and found acceptable by the staff in the SER (Reference 22), is acceptable because the system meets the radiation protection design objectives, and will maintain doses from airborne radioactive materials below the limits of 10 CFR Part 20 and as low as is reasonably achievable.

4.5 Health Physics

The licensee's health physics program objectives are to control radiation hazards to avoid accidential radiation exposures, to maintain exposures within the limits of 10 CFR Part 20, and to maintain individual and total man-rem occupation exposure, and exposure to the general public, as low as is reasonably achievable. The features of the health physics program are described below.

4.5.1 Radiation Protection Features

The radiation protection features at TMI-2 include a health physics counting room, several radiochemistry and counting areas, personnel decontamination and emergency treatment areas, access control points, calibration facility, respirator testing facility, respirator maintenance facility, contaminated clothing laundry, whole body counting facility and radiological controls offices. Operation of the SDS will utilize these systems and facilities as appropriate.

There are additional radiation protection features which have been added to support the SDS. These include (1) three shielded glove boxes with filtered ventilation, (2) use of shielded and ventilated valve manifold boxes, and (3) the use of shielded, all-welded piping runs. The glove boxes and valve manifold boxes are drained and have flush connections to aid in minimizing radiation fields by enabling decontamination without opening the manifold.

An offgas system maintains a negative pressure on the glove boxes, valve manifolds, feed standpipe, offgas separator tank, beta monitoring manifold, and feed tanks. The offgas system will continuously sweep those components. If the offgas system is not operating, neither filling of the tanks nor processing of water by SDS will be permitted.

We conclude that these radiation protection features are appropriate, and we find that they are acceptable for limiting occupational exposure during SDS operation.

4.5.2 Radiation Protection Equipment

Radiation protection equipment at TMI-2 includes portable survey instruments, personnel monitoring equipment, fixed and portable area and airborne radioactivity monitors, laboratory equipment, air samplers, respiratory protective equipment, protective clothing and contamination control equipment. This equipment will be used by SDS workers during SDS operation.

Personnel working in the SDS areas are required to use a combination of portable friskers, hand and foot monitors, and portable monitors prior to leaving the licensee's protected area. Since the SDS was installed in a portion of the licensee's restricted area the above contamination control devices are part of the existing plant equipment.

Continuous air monitors, tritium samplers, and area radiation monitors are located on the 347 foot elevation of the fuel handling building, in the vicinity of the SDS.

We conclude that the number and types of equipment to be used are adequate and provide reasonable assurance that the licensee will be able to maintain occupational radiation exposures as low as is reasonably achievable.

4.5.3 Personnel Dosimetry

All permanent and temporary plant personnel will be assigned beta-gamma thermoluminescent dosimeter (TLD) badges when working in the SDS areas and any other restricted (radiologically controlled) areas. These badges will be processed monthly, or more frequently if a significant exposure is expected or if required by the licensee's administrative control procedures. All personnel are also required to wear self-reading dosimeters when working in radiation work permit (RWP) areas. The readings from these dosimeters are used to keep a cumulative and easily obtainable total of an individual's dose prior to TLD badge processing. Visitors who enter radiologically controlled areas are issued TLD's. As a minimum, whole body counting is performed annually on personnel who enter RWP required areas or who wear respirators. Whole body counting and other bioassays are also performed when required by the radiological controls procedure manual or when deemed necessary by radiological controls personnel. All radiation exposure information will be processed and recorded in accordance with 10 CFR Part 20. We conclude that the licensee's personnel dosimetry program is acceptable for SDS operation.

4.5.4 Plant Procedures and Practices

Maintenance, repair and surveillance activities and methods used by the licensee are reviewed by the licensee's ALARA group to assure that all plant radiation protection procedures, practices and criteria have been considered and to assure that occupational radiation exposures will be ALARA and in accordance with Regulatory Guide 8.8. Procedures are in effect to assure that exposure limits are not exceeded by plant or visitor personnel onsite, to administer and control conditions of radiation work permits, to establish survey frequencies, to post radiation areas to control access to various categories of radiologically controlled areas, to control all radioactive material entering or leaving the plant site, to train plant and contractor personnel in radiation protection policies and procedures, and to meet the recommendations of Regulatory Guide 1.33.

The radiological controls organization, the qualifications of the health physics personnel, the objectives of the radiological controls program and the ways in which it will be implemented are in accordance with Regulatory Guides 8.8, 8.10 and 1.8 and are acceptable.

Radiation protection personnel who are assigned to work supporting SDS system operations will receive training on the SDS system as described in Section 3.8. Topics covered include system description, system design bases, and SDS radiological controls responsibilities. The training includes a classroom phase and a system walkthrough; at the end of the training a written examination is given. As a minimum, a radiological controls technician will be assigned to the SDS system at all times when filling the feed tanks or processing water.

4.6 Conclusion

Based on the information presented in the SDS System Description, the licensee's Radiation Protection Plan, the licensee's Radiological Controls Program Management Plan, and the licensee's responses to our questions, we conclude that the licensee's radiation protection program is capable of maintaining occupational radiation exposures within the applicable limits of 10 CFR Parts 20 and 50 and as low as is reasonably achievable regarding the operation of the Submerged Demineralizer System.

5. ACCIDENT ANALYSIS

5.1 Introduction

The quantities of radioactive materials that may be released during postulated accidents associated with SDS operations are based on the SDS as described in the licensee's TER of March 11, 1981 (Reference 10) and Final System Description (Reference 27). After reviewing these documents we considered a cask drop, HEPA filter failure during processing of containment sump or RCS water, and the failure of the 60,000 gallon feed tank system to the SDS as the bounding accidents. As described in the following subsections, we have analyzed the accidents which could occur during the operation of the SDS. Our analyses show that the offsite radiological consequences of each are within the limits of 10 CFR Part 20 or guidelines of Part 100, as appropriate.

5.2 Cask Drop Accident

We stated in section 3.4 of this safety evaluation that loading on the first stage ion exchanger resins would be administratively limited to 60,000 Ci. For the purpose of this evaluation, the accident scenario is based on a loading of 120,000 Ci (maximum curie loading ever expected). The worst-case accident that could arise would occur during the transfer of a first-stage ion-exchange vessel in a shipping cask within the fuel handling building. The cask is designed to retain its integrity during a 30 feet drop, however, at one point during the cask movement, it could fall a distance of 60 feet. For this 60 feet drop accident it was assumed that the ion-exchange vessel in the cask, as well as the cask seal, would be breached. This accident was evaluated in the PEIS (Reference 19) where we estimated that approximately 12 Ci of mostly cesium would be released to the fuel handling building atmosphere. The 12 Ci released to the fuel handling building atmosphere is based on a curie loading of 120,000 Ci of dewatered ion-exchange media and a partition factor of 10^{-4}. Since the cask drop was assumed to occur in the fuel handling building, the contaminated air would be treated by the FHBVS HEPA filters and the estimated release to the environment is 0.012 Ci. The resulting dose estimates for the maximum exposed individual as a result of this accident are presented in Table 21 and we conclude that the offsite radiological consequences are well within the guidelines of 10 CFR Part 100.

5.3 HEPA Filter Failure During Processing of Highly Contaminated Reactor Building Sump or RCS Water

In the PEIS, we evaluated the consequences of a HEPA filter failure during the processing of the highly contaminated containment sump water and RCS. The estimated doses for the maximum exposed individual to this postulated accident are listed in Table 22. However, in our PEIS evaluation, the radioactivity which was estimated to become aerosolized during the processing of the containment sump or RCS water was assumed to be deposited on the building ventilation system HEPA filter. The HEPA filter was subsequently assumed to fail releasing a fraction (0.001) of its contents to the environment. The PEIS evaluation did not consider the HEPA filters (i.e., two filters in series) incorporated in the

SDS design, and, thus, did not assume activity deposition in the SDS air filtration unit. In actuality, any radioactive material which becomes aerosolized during SDS process operations would collect on the first HEPA filter in the SDS GWTS. For any significant radioactivity to reach the environment during a postulated HEPA filter failure, the downstream HEPA filter in the GWTS and the two HEPA filters in the FHBVS would also have to fail. We conclude that this postulated event is well bounded by the PEIS evaluation and the dose consequences in Table 22 which are well within the guidelines of 10 CFR Part 100.

5.4 Failure of the 60,000 Gallon Feed Tank System to the SDS or of a 77,000 Gallon Reactor Coolant Bleed Tank

In the final PEIS, we evaluated the consequences of the leakage of the entire 700,000 gallons of water in the bottom of the reactor building into the ground water. This leakage would ultimately reach the Susequehanna River and then the public through the drinking water dose transport pathway. This is the bounding accident involving a release of liquid radioactivity from TMI-2. Our evaluation indicates that the concentrations of the principal radionuclides (i.e., Sr-90, Cs-137, and H-3) at the nearest drinking water intake are well below the concentration limits in 10 CFR Part 20 as shown in Table 23. In the case of the SDS, a similar type accident would involve the rupture of the four interconnected SDS feed tanks (15,000 gallon capacity each, total capacity 60,000 gallons) in the "A" spent fuel pool containing reactor building sump water or a 77,000 gallon reactor coolant bleed tank containing RCS water with subsequent leakage of the released water into the ground and eventually to the public via the water transport pathway. However, since the potential volume to be released is considerably less than that analyzed in the Final PEIS, the consequences of such an accident would be correspondingly lower, and we conclude that this postulated event is well bounded by the results of our PEIS evaluation of containment sump water leakage and within the limits of 10 CFR Part 20.

5.5 Conclusion

Based on the results of the above accident evaluations, we conclude that the postulated failure of the feed tanks to the SDS will not result in radionuclide concentrations in the Susquehanna River which exceed the limits in 10 CFR Part 20. Further, the consequences resulting from the postulated drop of a high specific activity zeolite vessel or the failure of a HEPA filter during SDS process operations are well within the guidelines of 10 CFR Part 100.

TABLE 21: Dose Estimates for the Maximum Exposed Individual as a Result of a Cask Drop Accident

Location	Pathway	Dose (mrem)[a] Total-Body	Bone	Liver
Nearest Garden	Inhalation	5.7×10^{-5}	2.0×10^{-5}	9.1×10^{-6}
	Ground Shine	6.8×10^{-5}	6.8×10^{-5}	6.8×10^{-5}
	Vegetable Use	4.7×10^{-4}	2.4×10^{-3}	1.5×10^{-3}
	Total	5.4×10^{-4}	2.5×10^{-3}	1.6×10^{-3}
Nearest Milk Goat	Inhalation	5.7×10^{-6}	1.0×10^{-5}	6.7×10^{-6}
	Ground Shine	6.8×10^{-5}	6.8×10^{-5}	6.8×10^{-5}
	Goat Milk Use	1.1×10^{-3}	7.6×10^{-3}	9.3×10^{-3}
	Total	1.2×10^{-3}	7.7×10^{-3}	9.4×10^{-3}
Nearest Cow and Garden	Inhalation	3.9×10^{-6}	1.4×10^{-5}	6.3×10^{-6}
	Ground Shine	6.8×10^{-5}	6.8×10^{-5}	6.8×10^{-5}
	Vegetable Use	4.7×10^{-4}	2.4×10^{-3}	1.5×10^{-3}
	Cow Milk Use	2.7×10^{-4}	1.2×10^{-3}	1.2×10^{-3}
	Total	8.1×10^{-4}	3.7×10^{-3}	2.8×10^{-3}

[a] Total-body and maximum organ (bone and liver) doses are listed. Doses were calculated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group considered are listed.

TABLE 22 : Dose Estimates for the Maximum Exposed Individual Caused by HEPA Filter Failure

Processing Operation	Dose (mrem)[a]		
	Total-Body	Bone	Liver
Containment Sump Water	3.3	16	12
Reactor Coolant System Water	1.5	6.0	0.28

[a] Total-body and maximum organ (bone and liver) doses are listed. Doses were calculated for four age groups: adults, teenagers, children, and infants. The highest dose estimates for each age group considered are listed.

TABLE 23: Peak Concentrations[a] of Principal Radionuclides in the Susquehanna River Resulting From Postulated Leakage of Sump Water From the Reactor Building

Radionuclide	Peak Concentration (uCi/ml) in Susquehanna River	Maximum Permissible Concentration (MPC) From 10 CFR Part 20
H-3	5.2×10^{-7}	3.0×10^{-3}
Cs-137	5.1×10^{-10}	2.0×10^{-5}
Sr-90	5.1×10^{-8}	3.0×10^{-7}

[a]Based on leakage of 700,000 gallons containing approximately 500,000 Ci of radionuclides.

6. TECHNICAL SPECIFICATIONS

Currently incorporated into the TMI-2 proposed Technical Specifications* are criteria that, in conjunction with the surveillance requirements, protect the health and safety of the public for the operation of the SDS. Proposed Technical Specification 6.8.1.g.4 of the Administrative Controls requires NRC approval of all procedures related to the operation of the SDS. Furthermore, proposed Technical Specification 6.8.2 prevents the licensee from altering the intent of any of these procedures without the NRC's re-approval of the procedure. This ensures that once the procedure is determined by the NRC to be adequate, this protection will not be lost through procedural or technical alterations.

The proposed Technical Specifications and associated Surveillance Requirements ensure that gaseous effluents from the fuel handling building, including those from the SDS, are monitored in accordance with General Design Criterion 64 of Appendix A to 10 CFR Part 50. Also all airborne releases are treated by the fuel handling building air cleanup system prior to discharge to the plant stack.

The staff has therefore determined that the present proposed Technical Specifications and associated Surveillance Requirements are adequate and no further additions or changes are required for the operation of the Submerged Demineralizer System.

*Reference to proposed Technical Specifications is to those Technical Specifications proposed by the Order of February 11, 1980.

7. ENVIRONMENTAL CONSIDERATIONS

7.1 Introduction

We have evaluated the licensee's proposal to decontaminate the containment sump and RCS water to determine if the associated environmental impacts fall within the scope of those already assessed in the PEIS. Specifically, we have evaluated the potential impacts resulting from the discharge of radioactive materials in gaseous effluents to the environment during SDS process operations for comparison with the PEIS estimates. We have evaluated the environmental impacts associated with the offsite transportation of the radioactive solid waste which is generated during process operations. Lastly, we have evaluated the occupational exposure associated with process operations, vessel handling, packaging, and storage, and the decommissioning of the SDS.

7.2 Radioactive Effluents to the Environment

In Section 3.6 of this report, we provided updated estimates of the releases of radioactive gaseous effluents to the environment resulting from SDS process operations and concomitant evaporation of tritiated water from the "B" spent fuel pool. The evaporation of tritiated water is expected to total approximately 9 Ci per year and the releases of the principal radionuclides to the environment due to processing of containment sump and RCS water are listed in Table 16.

Based on these effluent releases, we have estimated the cumulative total body dose to the population residing within a 50 mile radius of TMI-2. As shown in Table 24, the cumulative total body population dose is estimated to be 0.16 person-rem and compares favorably with the cumulative population dose estimate (i.e., 2.1 person-rem) which is based on gaseous effluents predicted in the PEIS. The cumulative population dose derived from the PEIS source term is based on the discharge of 50 Ci per year of tritiated water from the "B" spent fuel pool and the radionuclides listed in Table 14. We conclude that the environmental impacts associated with the discharge of radioactive materials to the environment due to SDS process operations fall within the scope of those already assessed in the PEIS.

7.3 Transportation of Radioactive Solid Waste

In Section 3.4 of this evaluation, we provided estimates of the quantity and type of solid radioactive waste anticipated to be generated during the processing of the containment sump and RCS water. These estimates are based on the licensee's plans for administrative control of curie loadings in both the SDS and EPICOR-II and the latest estimates of the radionuclide distribution in the containment sump and RCS water (see Tables 1 and 2). As shown in Table 25, we have compared these estimates for the generation of solid waste with the corresponding waste quantities and types estimated in the PEIS. The current estimates for spent cartridge filters, SDS prefilters and final filters, and spent zeolite vessels are well within the corresponding estimates in the PEIS for these wastes. However, the current estimate for spent organic vessels exceeds the corresponding numbers in the PEIS, primarily due to the licensee's plan to utilize EPICOR-II to polish the SDS effluent. Further, we anticipate

TABLE 24

Dose Estimates for the Projected Population in the Year 2010 Residing Within the 50-Mile Radius of TMI Resulting from SDS Process Operations and Associated Evaporation of Tritiated Water from the Spent Fuel Pool

Source Term		50-Mile Total Body Cumulative Population Dose (person-rem)
Updated Evaluation		
SDS Operation[a]		8.0×10^{-2}
H-3 Evaporation (9 Ci)		7.6×10^{-2}
	TOTAL	1.6×10^{-1}
Evaluation Based on PEIS		
SDS Operation[b]		1.7
H-3 Evaporation (50 Ci)		4.2×10^{-1}
	TOTAL	2.1

[a]Based on the source term in Table 16.

[b]Based on the source term in Table 14.

TABLE 25

Estimates Of The Generation Of Solid Radioactive Waste Resulting From The Processing Of Containment Sump And RCS Water

Type of Package	Number of Packages	
	Current Estimate	PEIS Estimate
Spent Cartridge Filters	5	21
SDS Prefilters and Final Filters	25	32-48*
Spent Zeolite Vessels	20	8-60
Spent Organic Vessels	40	4-16

*Includes the number of packages estimated for containment sludge since our current estimate for the number of prefilters and final filters is based on sludge removal in the sump.

several additional packages of miscellaneous waste (valve containment boxes, piping, tools, etc.) to be generated from the decommissioning of the SDS. In view of these additional waste packages, we have estimated in Table 26 the number of shipments of SDS wastes for comparison with estimates in the PEIS since environmental impact is a function of the total number of offsite shipments. In terms of required offsite shipments of decommissioning wastes, the packages of miscellaneous SDS material should not constitute more than two shipments. Our evaluation shows that the current estimates for required shipments fall within the range of shipments estimated in the PEIS for corresponding wastes and we conclude that the environmental impacts associated with the offsite shipment of the solid waste generated in SDS/EPICOR-II fall within the scope of those impacts previously assessed in the PEIS.

7.4 Occupational Doses

We have reviewed the licensee's estimates for occupational doses (Reference 16) and we have developed our own estimated cumulative occupational doses which will be associated with process operations, vessel handling, packaging, storage, and the decommissioning of the SDS. In the PEIS, we estimated the occupational dose associated with the processing of the containment sump and RCS water in an SDS type system. Further, we estimated the occupational dose associated with vessel handling, packaging and storage. As discussed in Section 7 and 8 of the PEIS, the total for these estimates ranged from approximately 11 to 15 person-rem. In the preparation of the PEIS, we did not quantify the cumulative occupational dose associated with decommissioning of the SDS; however, based on the design and construction of the SDS (i.e., the use of small bore stainless steel piping, flushing provisions, ease of system access, etc.) we expected the corresponding occupational dose to be insignificant (i.e., a small fraction of the dose anticipated for the entire cleanup of TMI-2).

Based on the now completed construction, layout and design of the SDS and on more detailed information regarding staffing requirements, process operations, sampling, vessel handling, packaging and disposition, we have re-evaluated the occupational dose associated with process operations, vessel handling and packaging. We estimate that these SDS operations will result in cumulative occupational doses totaling 21 person-rem. Table 27 presents a more detailed breakdown of this estimate. Additionally, we have estimated that the cumulative occupational dose associated with decommissiong the SDS will total about 35 person-rem. This quantitative estimate of the occupational dose resulting from decommissioning yields a value which is a small fraction of the cumulative dose estimated in the PEIS for the entire cleanup up to a decision point as to whether the plant will be refurbished or decommissioned (2000-8000 person-rem). In addition, our estimate of occupational dose resulting from decommissioning the SDS is a small fraction of the dose estimated in the PEIS for decommissioning the entire plant (1800 person-rem). Details of our dose estimate for decommissioning are also presented in Table 27.

Based on this most recent estimate of the cumulative occupational radiation doses associated with SDS activities, the staff concludes that the likely occupational doses resulting from use of the SDS fall within the scope of those previously considered in the PEIS.

TABLE 26

Estimates Of The Number Of Shipments Of Solid
Radioactive Waste Resulting From The Processing Of Containment
Sump And RCS Water And Decommissioning Of The SDS

Type of Waste	Number of Shipments	
	Current Estimate	PEIS Estimate
Spent Cartridge Filters	1	3
SDS Prefilters and Final Filters	8	13-16
Spent Zeolite Vessels	13	7-30
Spent Organic Vessels	40	4-16
Miscellaneous Decommissioning Wastes	2	*
TOTAL	64	24-65

*Not estimated in the PEIS.

TABLE 27

Cumulative Occupational Radiation Dose Estimates for SDS Related Activities

SDS Activity	Dose Estimate (person-rem)
1. System Operations (Total)	21.2
a. Filling Feed Tank	0.9
b. Batch Processing	3.4
c. Filter Vessel Replacement	0.03
d. Process Ion Exchange Vessel Replacement	0.03
e. Leakage Containment Vessel Replacement	0.01
f. Cask Handling	0.7
g. Vessel Dewatering	0.3
h. System Non-operational	2.5
i. Supervision	2.5
j. Health Physics Surveys	5.5
K. Health Physics General Coverage	5.3
2. Decommissioning Operations (Total)	35.3
a. Removal of Equipment in Pool	18.0
b. Removal of Equipment Above Pool	6.0
c. Removal of Off-gas Separator Tank and Stand Pipe	0.7
d. Removal of Ventilation Equipment and Electrical Cabinets	0.4
e. Waste Handling	2.0
f. Health Physics and Decon Support	5.0
g. Supervision/Engineering Support	3.2

7.5 Summary of Environmental Considerations

We have evaluated the environmental impacts associated with the processing of the containment sump and RCS water in the combined SDS/EPICOR-II system. Specifically, we have evaluated the impacts resulting from the discharge of radioactive materials in gaseous effluents to the environment, the impacts associated with the offsite transportation of the solid radioactive waste, and the impacts to the occupational workforce during SDS process operations and subsequent system decommissioning. We conclude that these environmental impacts fall within the scope of those already assessed in the PEIS and are therefore acceptable.

8. NEED FOR PROMPT ACTION

The Commission in its April 27, 1981 Statement of Policy on the PEIS concluded that "...this statement [the final PEIS] satisfies our obligation under the National Environmental Policy Act (NEPA)" and further stated "...the licensee should accelerate the pace of the cleanup to complete expeditiously all decontamination activities consistent with ensuring protection of public health and safety and the environment" (46 F.R. 24764, May 1, 1981). However, the high* radiation levels within the reactor building (particularly in the lower portions of the building) due to the continued presence of the highly contaminated water in the building's sump and in the RCS significantly hinders the performance of major decontamination activities within the reactor building. In fact, in keeping with the ALARA requirements of 10 CFR Part 20, we have concluded that this water must be decontaminated prior to the initiation of major decontamination activities within the reactor building to substantially reduce the occupational exposure levels within the building. Furthermore, as long as the highly contaminated water remains in the reactor building, a possibility exists for the building to begin leaking and releasing the highly contaminated water to the environment. Decontamination of this water with the corresponding immobilization of this activity on the SDS zeolite ion exchangers and their subsequent removal from TMI-2 to one or more Federal government facilities for DOE research, development, and testing purposes will significantly decrease the possibility of releasing this activity to the environment. Therefore, we have concluded that consideration of the public's health, safety and interest requires that processing of the reactor building sump and RCS water be initiated and completed expeditously. Since the SDS is available and capable of safely performing this decontamination activity and since the solid radioactive wastes to be generated during its operation can be safely disposed of offsite, this system should be used to effect waste water cleanup.

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*Presently the exposure rate at the 305 foot elevation level (entry level) is approximately 200-400 mR/hr, of which the majority is due to shine from the sump water. The exposure rate over the unshielded stair well is approximately 20 R/hr, primarily due to direct shine from the sump water.

9. CONCLUSIONS

Based on the foregoing considerations, we have concluded that: (1) the SDS can be operated within the limits of the existing Facility License, (2) the SDS will immobilize and concentrate the radioactive materials dispersed throughout the reactor building sump water and reactor coolant system water, (3) the solid radioactive wastes to be generated by operation of the SDS can be safely disposed of, (4) the environmental impacts of operating the SDS fall within the scope of those already assessed in the Final PEIS, (5) the SDS has been designed and fabricated and will be operated in accordance with the applicable criteria (including ALARA) and guidance provided in the Commission's Regulations and Regulatory Guides, (6) the public's health, safety and interest requires that processing of the reactor building sump and RCS water be initiated and completed expeditiously using the SDS, (7) there is reasonable assurance that the health and safety of the public will not be endangered by operation of the SDS, and (8) such activities will be conducted in compliance with the Commission's regulations.

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*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and/or the National Technical Information Service, Springfield, VA 22161.

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A Safety Evaluation Report related to the operation of the Submerged Demineralizer System at Three Mile Island Nuclear Station, Unit 2, located in Dauphin County, Pennsylvania, has been issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This Safety Evaluation Report presents the staff's analysis of information submitted by the licensee (Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company).

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